INTERNATIONAL RADIATION PROTECTION ASSOCIATION

6th INTERNATIONAL CONGRESS
Organized by the Fachverband für Strahlenschutz e.V.
Berlin (West) May 7 – 12, 1984

RADIATION – RISK – PROTECTION

Edited by A. Kaul, R. Neider, J. Peńsko,
F.-E. Stieve and H. Brunner

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RADIATION – RISK – PROTECTION

COMPACTS VOLUME I

Topics

1 Sources of Human Exposure
2 Environmental Transfer and Modelling
3 Dose Assessment
4 Biological Effects
5 Internal Emitters
6 Risk and Detriment Assessment

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P R E F A C E

The motto of the 6th International IRPA Congress at Berlin (West) is

RADIATION - RISK - PROTECTION.

This motto, as the basis of a full week of scientific discussion in Berlin (West) with colleagues from home and abroad, reflects our mutual spheres of interest and is focused on:

- detecting RADIATION, ionizing or non-ionizing, by measurement,
- quantifying the RISK to man from irradiation, and
- assuring the PROTECTION of man and his environment by reducing the radiation risk as far as reasonably achievable and in relation to the benefits from the application of radiation.

The response to the First Announcement of the Congress was unexpectedly large:
nearly 500 preliminary abstracts were received. Even after the deadline for registering scientific papers had expired, the world-wide interest was not diminished.

The members of the International Meeting Programme Committee and the International Meeting Commission had to take care not to overload the scientific programme and, at the same time, not to greatly limit the scientific diversity offered. Using the radiation protection principles of ICRP as an analogy in obtaining an optimization through justification and limitation, it was not an easy task to decide which papers should be included in the programme and which should be omitted. I sincerely hope that we were succesful in structuring a scientifically balanced programme that will allow the Berlin Congress to be remembered as something more than a social event.

As President of the Congress and on behalf of the editors of the Compacts, I would like to take this opportunity to thank all other members of the International Meeting Programme Committee and the International Meeting Commission for their valuable contribution in creating the scientific structure of this Congress.

All papers deemed acceptable by the Programme Committee are included in these Compacts. Independent from the type of presentation, either oral or poster, each author had four pages at his or her disposal. Not all authors took advantage of presenting their results. Therefore, in these cases, the abstracts which were submitted have been published in lieu of a compact. The plenary speakers were given eight pages, since their papers required more space due to their introductory nature concerning the main topics.

The editors classified the papers according to topics, well knowing that this sequence might render a study of the written version of the compacts more difficult. Nevertheless, this solution was adopted assuming that these Compacts are of lasting value and may serve as a theme-oriented reference work also for those unable to attend the Congress, but wishing to be informed on the scientific and technological state of the art as presented at the Berlin Congress in 1984.

In conclusion, I would like to express the hope that these contributions may be useful to all readers. Again, may I repeat my thanks to all those who contributed to the organization of the Congress and to the preparation of the volumes of Compacts which, of course, would not be available without the efforts of the authors!

For the editors,

Alexander Kaul
Congress President

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LEVELS AND EFFECTS OF HUMAN EXPOSURE TO IONIZING RADIATION

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The paper intends to outline the activities of the UNSCEAR and to summarize the content of its most recent report to the General Assembly [1]. This purpose is made difficult by the very large amount of information it contains: only the most general conclusions will therefore be presented and the reader is referred to the original publication for a detailed description of the conditions under which such conclusions were derived.

RADIATION SOURCES AND DOSES

The Committee systematically analysed all major sources of ionizing radiation which may give rise to exposure of the public or of the workers. For each source the results were expressed as individual doses, which are meant to give an indication of the possible risk level under the various exposure conditions; and also as collective doses, which are related to the present and future overall health impact of the sources. The new radiological units were employed and the organ and whole-body doses were combined by the use of effective dose equivalent. As a consequence, the relative weight of the various sources may have changed by comparison with previous analyses.

Natural Sources

The highest contribution to the annual average doses received by the human species comes from the natural sources, both external (cosmic rays, radionuclides in the soil or in building materials) and internal (ingested or inhaled radionuclides). The most recent estimates of the Committee in respect to natural sources are given in Table I. They show that the annual effective dose equivalent from natural background is now set at around 2 mSv, about one-half of which is due to the presence of radon in dwellings. The contribution of other natural or technologically modified sources and of consumer products does not substantially depart from previous assessments.

Man-made Sources

In contrast with the relative stability of natural sources, man-made ones vary significantly as a function of time and location and the doses received by the exposed population groups may therefore change substantially.

Medical irradiation ranks highest among the artificial sources of human exposure. Individual doses are extremely variable and therefore the use of average doses is of little significance. Collective doses, on the contrary, may provide some indication of the importance of medical exposures. The concept of collective dose equivalent cannot easily be applied in this case,

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* Opinions expressed in this presentation do not necessarily reflect the views of the UNSCEAR.

however, the tentative evaluation of the Committee is that the annual effective dose equivalent for diagnostic irradiation in developed countries may be of the order of 1 mSv. Incomplete information from developing countries point to a frequency of radiological examinations which is lower by an order of magnitude than in developed areas of the world. Therefore, the value of annual collective effective dose equivalent globally applicable all over the world for diagnostic radiology may tentatively be set at about 0.4 mSv.

TABLE I. ESTIMATED ANNUAL EFFECTIVE DOSE EQUIVALENTS FROM NATURAL SOURCES OF RADIATION IN AREAS OF "NORMAL" BACKGROUND

Source	Annual effective dose equivalent (millisievert)		
	External irradiation	Internal irradiation	Total
Cosmic rays			
Ionizing component	0.28		0.28
Neutron component	0.02		0.02
Cosmogenic nuclides		0.015	0.015
Primordial nuclides			
Potassium-40	0.12	0.18	0.30
Rubidium-87		0.006	0.006
Uranium-238 series	0.09	0.95	1.04
Thorium-232 series	0.14	0.19	0.33
TOTAL (rounded)	0.65	1.34	2.0

The exposure of mankind as a result of atmospheric nuclear explosions has always been a subject of special interest by the Committee. A study of the annual collective doses received from fallout exposure shows (Figure 1, (a)) that there was a considerable increase during the early 1960's to a peak corresponding to about 7% of the average background dose. Following the partial test ban treaty a steady decrease was observed down to values of about 1% of background at present. The annual collective doses received at any given time are the result of all explosions up to that time. However, it may be of interest to consider the collective doses committed up to the complete decay of the radionuclides involved in the explosions carried out in any given year. The data show (Figure, 1 (b)) that the explosions in the years 1961-1962 for their overall health impact were by far the most important of all explosions conducted so far.

The Committee assessed the doses to the public and the workers due to the production of nuclear power. On the basis of models and of reasonable approximations, it was estimated that the short-term collective dose equivalent received by members of the public through this source of radiation may have increased by two orders of magnitude between 1960 and 1980, to be at present about 0.01% of the corresponding value from natural sources, as a global average. There are, of course, very large variations as a function of the distance from the plants. Workers involved in the production of nuclear power receive effective dose equivalents which are, as a gross average, of the same order of magnitude as that from natural sources. The long-term component of the doses from nuclear power plants has also been evaluated. The results have emphasized the role of radon emanating from mine and mill tailings over the extremely long periods of time. However, the uncertainties involved in these assessments are very large.

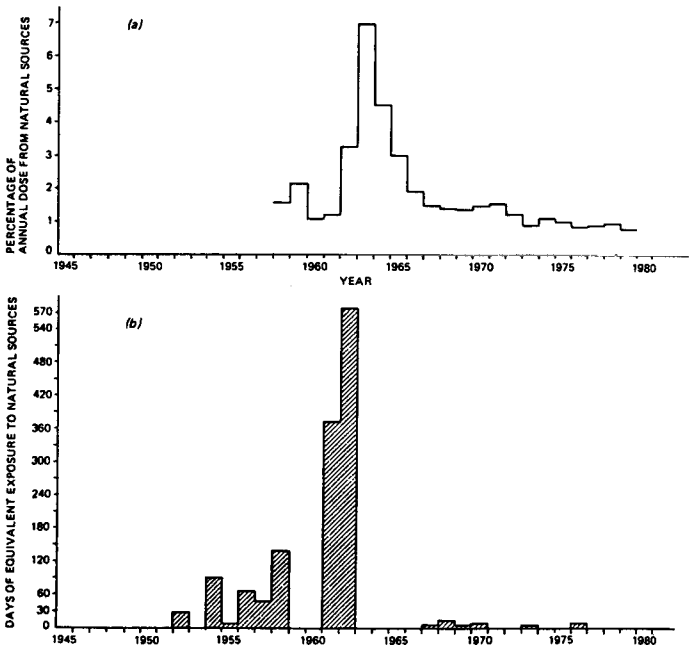

FIGURE 1. TRENDS WITH TIME OF COLLECTIVE DOSES FROM ATMOSPHERIC NUCLEAR EXPLOSIONS. (a) AVERAGE ANNUAL COLLECTIVE DOSES RECEIVED BY THE WORLD POPULATION BETWEEN 1958-1979; (b) COLLECTIVE DOSES COMMITTED FOR THE FUTURE BY EXPLOSIONS CARRIED OUT BETWEEN 1952 AND 1976.

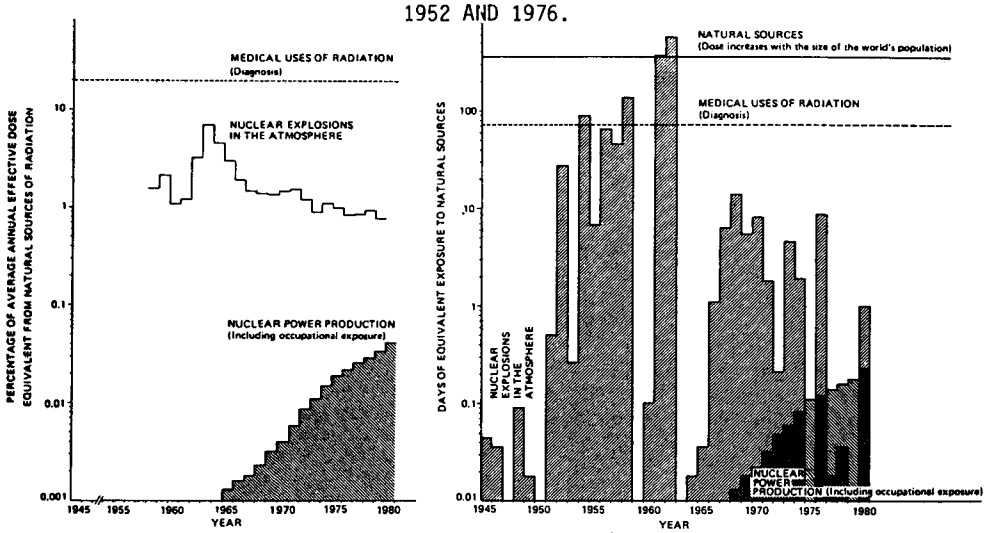

FIGURE 2. TRENDS WITH TIME OF DOSES FROM DIFFERENT SOURCES OF RADIATION. (a) ANNUAL EFFECTIVE DOSE EQUIVALENTS, EXPRESSED AS PERCENTAGE OF THE AVERAGE EXPOSURE TO NATURAL SOURCES; (b) COLLECTIVE EFFECTIVE DOSE EQUIVALENT COMMITMENTS PER YEAR OF PRACTICE, EXPRESSED AS DAYS OF EQUIVALENT EXPOSURE TO NATURAL SOURCES.

Through the use of effective dose equivalent it has been possible to compare the contribution of the various sources, as in Figure 2, (a). Keeping in mind that grouping of the various sources in the Figure is simply a way of representing their respective health impact, it is shown that the collective effective dose equivalent due to natural exposures is the most important, followed by that due to medical exposure at about 20% of the natural level. The contribution of medical exposure is estimated to have been roughly constant in time. Doses due to nuclear explosions were discontinuous, but they mostly decreased since 1963. The collective effective dose equivalent from the production of nuclear power increased gradually in relation with the expansion of nuclear programmes, although its contribution is still at a substantially lower order of magnitude. As an alternative way of presentation, which takes into account all present and future exposures due to the above practices, Figure 2, (b) shows the collective effective dose equivalent commitments in each year from 1945 to 1980, expressed as days of exposure to natural sources that would give the same dose.

BIOLOGICAL EFFECTS AND RISKS

In the field of hereditary effects the Committee assessed the most recent information and derived new estimates of risk. On the contrary, the review of somatic effects only covered some of the non-neoplastic consequences of the exposure of the whole body or of specific organs and tissues to relatively high doses.

Hereditary Effects

The newest scientific data qualitatively reinforced previous working hypotheses underlying the genetic risk estimates. Quantitatively, they led to values of the risk which were in good agreement with those derived in 1977 [2]. These risks were evaluated in the light of current understanding of human genetics, which shows that spontaneous genetic defects in man arise in about 10% of live-born children, considering conditions of variable clinical severity.

A detailed analysis of dose-effect relationships for chromosomal aberrations and gene mutations obtained in a variety of mammalian species led to the validation in principle of the hypothesis that there is some proportionality between the spontaneous and the radiation-induced mutability of specific genes. On this hypothesis rests the indirect method of genetic risk assessment. By the use of such method it is presently estimated that a human population exposed to small doses of radiation of low LET at the rate of 1 Gy per generation (30 years) would express of the order of 2200 new cases of genetic diseases per million live-born in the course of the first generation. If such an exposure would continue indefinitely, the corresponding number at the equilibrium would be expected to reach 15000 cases. Table II shows minor differences with respect to the 1977 estimates, but they can hardly be seen as significant. These predictions were confirmed by an independent and direct method of assessment which relies on the mutability of single genes.

TABLE II. NUMBERS OF GENETIC DISEASES PER MILLION PROGENY FOR POPULATION EXPOSURE TO LOW-LET, LOW DOSE IRRADIATION AT THE RATE OF 1 GY PER GENERATION, ESTIMATED BY THE DOUBLING DOSE METHOD (UNSCEAR, 1977 AND 1982).

TYPE OF DISEASE	FIRST GENERATION		AT EQUILIBRIUM	
	1977	1982	1977	1982
Dominant and x-linked	2000	1500	10000	10000
Structural aberrations	3800	240	4000	400
Complex aetiology	500	450	4500	4500
	~ 6300	~ 2200	~ 18500	~ 15000

There has been an initial attempt to evaluate the genetic risk not simply as the number of expected diseases, but also in terms of their clinical importance. The new approach could eventually result in a better appreciation of the socio-economic aspects of radiation-induced hereditary effects. However, given the virtual absence of direct information in man, extrapolation from other species with appropriate corrections will have to be continued in the future to assess the human risk.

Somatic Effects

There was a very large amount of data available in man on the effects of irradiating single organs and tissues, mainly derived from radiotherapy experience, and there was the need to review these data, also in the light of a similarly large experience in experimental animals, in order to derive conclusions of general validity. The Committee carried out such an exercise on the basis of two unifying concepts: that cell survival, on the one hand, and tissue kinetics, on the other, determine to a large extent the amount and timing of tissue reactions. The difficulties of the exercise were to be found in the need to extrapolate data obtained at high doses to a much lower dose range and in the need to adapt data obtained from diseased tissues to the response of normal tissues. A systematic analysis of all tissues was useful to point out their relative sensitivity and the dose levels at which different types of early or late effects may be expected to arise. However, beyond such detailed results, there emerged the confirmation of the general notion that all non-stochastic effects on tissues are characterized by threshold-type sigmoid dose relationships. The values of the thresholds may be different for each given tissue and effect, but the mechanisms underlying the threshold make it impossible that it may be abolished at low doses and dose rates. Consequently, if one assumes non-threshold linear relationships to apply to the stochastic effects, it follows that these might still be seen at doses which are too low for the non-stochastic damage to be expressed. For this reason the Committee considers that at the low doses and dose rates the induction of tumours could be the only somatic consequence of irradiation in man.

This conclusion is more straightforward in the case of single tissue and organ irradiation than in the case of whole-body exposure where the effects may be more diffuse and their pathogenesis uncertain. The Committee investigated an effect of irradiation which goes under the name of non-specific life span shortening. After a general discussion and an extensive analysis of data obtained in experimental animals and in man, the Committee concluded that at low doses and dose rates the shortening of life seen in irradiated populations may be attributed essentially to the appearance of tumours above the natural level. Non-specific damage may certainly be seen at doses around

and above the LD_{50}, and this is brought about by lesions to the vascular and connective tissues. However, the notion that non-specific effects, in addition to the neoplastic ones, should be taken into account (for radiation protection purposes, for example) appears unjustified in the light of present evidence.

It has often been suggested that the effects of a ubiquitous agent such as radiation might be modified through the interaction with other physical, chemical or biological agents widely present in the environment. This has obvious implications for the establishment of precise estimates of risk. Numerous difficulties had to be faced in examining this issue. In fact, the available evidence refers to levels and exposure modalities which are very different from those existing in the environment; there is a lack of any systematic analysis of the possible interacting agents, their dose relationships and mechanisms of action; the statistical significance of many observations is doubtful and the effects investigated marginal; concepts such as additivity, synergism and antagonism are not easily defined conceptually. The study revealed some conditions of interaction which are of importance under specific, mostly occupational, situations such as the synergism between radon and tobacco exposure leading to pulmonary tumours in uranium miners. It was impossible, however, to document interactions of wide significance at the population level, which might require special attention in the context of present radiation protection philosophy. In this field there is ample scope for further research to explore the nature of the interacting agents, their mechanisms of action, the dose dependencies, the type and sequence of the treatments.

REFERENCES

[1] United Nations. Ionizing Radiation: Sources and Biological Effects. United Nations Scientific Committee on the Effects of Atomic Radiation 1982 report to the General Assembly, with annexes. United Nations sales publication No. E.82.IX.8. New York, 1982.

[2] United Nations. Sources and Effects of Ionizing Radiation. United Nations Scientific Committee on the Effects of Atomic Radiation 1977 report to the General Assembly, with annexes. United Nations sales publication No. E.77.IX.I. New York, 1977.

RISK AS A BASIS FOR RADIATION PROTECTION

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Discussions on risk in radiation protection go back more than 30 years (1,2) but it was not until Report 26 by the ICRP in 1977 (3) that risk estimates became closely associated with a protection system. Since future protection systems will become more and more dependent on numerical estimates of risk, these estimates and their uncertainties become of critical importance and must constantly be updated as new information becomes available. I shall discuss here some features of the complex subject of somatic risk from ionizing radiation in relation to problems in radiation protection. Others in later plenary lectures will develop other aspects of this question.

SOMATIC RISKS

Somatic risks are distributed in time after each exposure and, for example, when an individual receives a dose of 1 rad of low-LET radiation, he or she is initially at risk of dying from leukemia or later from any one of a wide variety of solid tissue tumors, especially thyroid, breast, lung or bone. A model which attempts to describe the situation is shown in Figure 1.

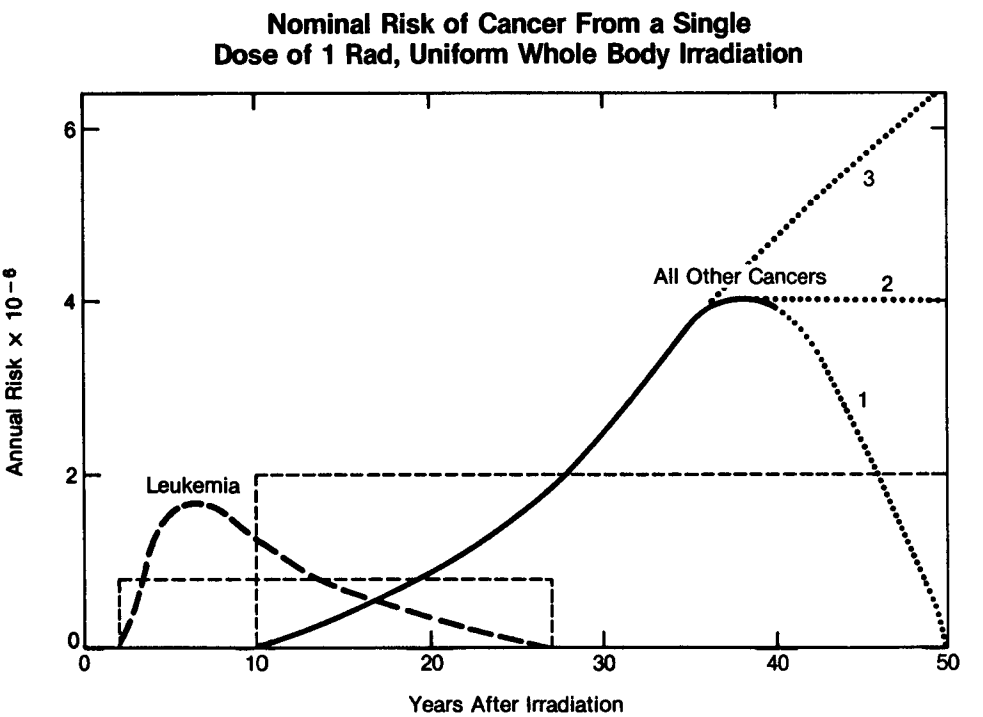

Figure 1

At first, for 2 years, there is no risk. Then the individual is at increasing risk of leukemia up to about 6-8 years, the risk falling slowly thereafter to essentially zero at 27 years. The average annual risk over the period shown dashed is 0.8×10^{-6}/year and the period of risk is 25 years, so that the lifetime risk of mortality from leukemia is 2×10^{-5} per rad. In the case of the solid tumors, we know the exact time relationships less well, especially at later times which are shown dotted. There is no risk in the first 10 years (latent period); then the risk rises slowly. Thereafter the risk may decline (like leukemia) to be over by about the 50th year, curve 1, continue at about the same level, curve 2, or rise steadily, like the natural risk of cancer, curve 3. For simplicity, we shall consider curve 1; the average annual risk is taken to be 2×10^{-6}/year for a period of risk of 40 years (dashed line) or a lifetime risk of solid tissue cancers per rad of 8×10^{-5}. (The actual values of the annual risk, 0.8×10^{-6}/yr for leukemia, 2.0×10^{-6}/yr for solid tumors can easily be adjusted in the model if and when this should be warranted.) The risk of solid tumors, while slower to develop, is 4 times greater than that of leukemia, and the total risk of leukemia and other cancers is 10^{-4} per rad lifetime. Note that for individual solid tumors, e.g., lung and breast, the actual latent periods and periods at risk may differ and latent periods also vary with age.

It is evident (4) that after continuous irradiation, the annual risk will increase year by year in a somewhat complex way, reaching about 10^{-4}/yr after 50 years at 1 rad/yr, and an accumulated risk of 0.24% eventually rising to 0.5%. If the risk from each dose is actually over after 50 years, all risk will not cease until 50 years after exposures are terminated. Risk will never cease if the risk from a single dose continues indefinitely.

Recently (4), nine different estimates of risk for individual organs made between 1957 and 1982 were compared; they varied by not much more than a factor of 2. Table 1, for example, compares the "best" estimates made by UNSCEAR 1977 (5), ICRP 1977 (3) and BEIR 1980 (6) with a more recent NCRP appraisal (7) which includes the latest Japanese data (8). Actually, each of the reports gave ranges

Table 1
ABSOLUTE RISK ESTIMATES (LIFETIME)
(MORTALITY X 10^{-6}/RAD)

	UNSCEAR 1977	ICRP 1977	BEIR 1980	NCRP 1983
Leukemia	20	20	25	20
Thyroid	10	5	~ 5	5
Breast	(50)	AVG 25	20	20
Lung	25	20	25	25
All Others	40	50	35	50
	95-145	125	110	120

NCRP Figures are Based on Appraisal of All Earlier Data plus Kato and Schull 1982.

and in particular the BEIR committee gave various other estimates based on other projection and extrapolation models.

In addition to the uncertainties associated with the small numbers involved in each tumor endpoint, the estimates depend on many other factors, some of which are not well known or understood. I have already published some comments about them (9,10,11), and I will make only a few points here.

Japanese Data, Leukemia and Solid Tumors

The total risk was taken to be 5 x leukemia risk (5). The lifetime leukemia risk at 2 x 10^{-3} Sv^{-1} is probably known to within about a factor of 3. The ratio 5, once thought to be only about 2 (in the 1950's) was first cited as 5 by NCRP in 1971 (2). Even as the data from Japan continues to accumulate more and more solid tumors (but no more leukemias), the projected number of 5 still seems to be a conservative ratio since to date there are ~ 90 leukemias and about 160 solid tumors, a ratio of about 1.8. However, the solid tumors in the last 4-year period rose sharply (8), and the next periods of data accumulation in the Japanese will be critical. The value of the ratio is supported by other sources, not useful for quantitative risk estimates because of poor dosimetry, but useful for ratios, such as the U.S. and the British radiologists, studies of pelvic irradiation, and the treatment of spondylitis with x rays.

Japanese Dosimetry

Estimates of the doses received by the Japanese survivors of the A-bombs were made in 1950, 1957 and 1965 (12). Recently, as a result mainly of new spectral information on the radiations believed to have issued from the weapons and some new transport calculation techniques, revisions in the doses have been proposed (13). The proposed changes at Nagasaki are minor but those at Hiroshima may decrease the neutron dose by about x 10 and increase the gamma dose by about x 4. A multilaboratory program now in progress is likely to confirm the LLNL estimates, at least approximately. Some changes in structural shielding and organ shielding estimates must still be taken into account. An extensive program of actual measurement of the dose--by activation techniques for the neutrons, and by thermoluminescence in roof tiles for the gamma rays is planned in both Japanese and U.S. laboratories (14). This may provide a real measurement estimate of the doses to confirm or challenge the calculations. The effect of the changes on risk estimates is not expected to be large, the effects at Hiroshima previously thought to be due to neutrons now being attributed to the increased γ rays. It will be surprising if, in the end, our estimates of γ risk change by as much as x 2 (15). (If risk estimates are based on sources other than the bombs only, the estimates rise by about 50-100% (4,16), well within the uncertainties expected.) The new proposed dosimetry brings the data for the two cities into better agreement for some biological endpoints, such as leukemia (17,15).

Absolute and Relative Risk

The risks quoted above are based on the absolute risk model. BEIR 1980 gave estimates based on each model and in some cases relative risks are 2 to 3 times greater than the corresponding absolute risk. Relative risk seems to have been a more useful predictor of

effects in the Japanese than absolute risk, at least for two age groups (18). Relative risk may not be so useful for comparing organ risks or for constructing protection systems based on comparative organ risk such as that of ICRP (3).

The respective merits of absolute and relative risk models will undoubtedly be better understood when the accumulation of data in the Japanese is more mature. Obviously, the two models have different mechanistic implications.

Incidence vs. Mortality

Mortality information has until recently been more reliable than incidence data. As incidence data improve, they may assume greater importance because of their more direct applicability for dose-effect models and their possible significance philosophically. The relationship between induced and fatal cancers varies with site but may be about x 2 higher overall.

In spite of these and many other uncertainties, the estimates of risk given in Table 1 seem sound, though surprises are always possible because our data at low doses is still so unfirm. However, until the cancers other than leukemia in the Japanese become clear (one or two more data cycles at least) and the dosimetry re-evaluation is complete, higher total risk estimates even for low-LET radiation cannot be discounted.

RISK AND RADIATION PROTECTION LEVELS

Radiation protection levels for workers, for the public and for emergencies have been in existence for some time; they were developed without risk as a base and have served our society well, so far. How then, does one relate a knowledge of risk to judgments about protection levels? One way is to simply consider the risks associated with the levels and compare these with other appropriate circumstances.

Occupational

We can derive from the model of Figure 1 (4) that exposure continuously at 1 rad/yr will result after 50 years in an annual risk of 10^{-4}/yr and an accumulated risk of 0.24% (eventually 0.5%). Also, comparison with fatal accident rates in industry indicates that we should aim to maintain the average risk to workers below 10^{-4}/yr in order to compare favorably with the "safer" industries (4,7).

Actual exposures in occupational circumstances in the USA for different worker groups are shown in Table 2. They involve nominally 1,357,000 workers to an average exposure of 110 mrem. For those actually exposed, 610,000, the average is 250 mrem, corresponding to an annual risk eventually of 0.25 x 10^{-4}/yr and a lifetime risk (eventually) of about 0.125%.

While the average level may be satisfactory, some few workers could be consistently close to the maximum annual limit. At 5 rem/yr for 50 years (age 18-68), 250 rems is theoretically possible, an annual risk of 5 x10^{-4}/yr and a total risk after 50 years of 1.2%

Table 2
OCCUPATIONAL EXPOSURE SUMMARY
(1980)

	No. of Workers		Mean Exposure mrem		Collective Dose
	Total	Exposed	Total	Exposed	(person-rem)
Medicine	477,500	218,600	70	150	32,600
Industry	371,800	196,700	120	230	46,200
Nuclear Fuel Cycle	148,100	82,900	390	700	57,700
Government	178,600	56,300	60	200	11,100
Misc.	181,100	53,400	40	140	7,900
All Workers	1,357,100	612,900	110	250	155,500

rising to a maximum of 2.5% lifetime. This is high compared with a natural rate of dying of cancer of 16-20%. This risk could be avoided by either a lifetime limit of say 100 rems or perhaps better 2(N-18) rems where N is the age in years. This would keep the flexibility of the NCRP age proration formula, limit the exposure of the youngest workers, limit the overall risk to no more than 1% and reduce the magnitude of the changes in organ dose levels permitted by the ICRP system. NCRP is giving consideration to this at present. In other respects, the exposure of workers seems reasonable and on the average decreasing with time, so that ALARA seems to be working.

High LET Radiation

An important subset of the occupationally exposed are those relatively few individuals, potentially or actually exposed to high LET radiations such as neutrons. NCRP issued a statement in February 1980 (19) warning that neutrons may be somewhat more hazardous relative to x and γ rays than previously thought. This view was based mainly on the effects attributed to neutrons at Hiroshima, based on the T65 dosimetry, but also on the increasing awareness of laboratory studies indicating RBE's higher than the quality factor (of 10) for fission neutrons. Subsequent dosimetric information has essentially removed the neutrons from the scene at Hiroshima, but laboratory studies have continued to indicate RBE values higher than 10 at low doses. The RBE values may depend on endpoint and an average may be in the 30-50 range (20). The situation, as indicated in the NCRP statement, may not be as drastic as these RBE's imply because measurement on the body may tend to overestimate the actual whole-body dose substantially. Nevertheless, it seems that revisions in the quality factor should be in prospect when adequate data is available and a comprehensive study of the problem can be completed.

Public

Some of the sources of exposure to the public yielding the largest population detriment are shown in Table 3. The limit for individual members of the public, 0.5 rem/yr, results in an average

Table 3
POPULATION DETRIMENT FROM DIFFERENT RADIATION EXPOSURE CIRCUMSTANCES

SOURCE	AVERAGE ANNUAL EFFECTIVE DOSE EQUIVALENT	# PEOPLE	ANNUAL COLLECTIVE EFFECTIVE DOSE EQUIVALENT
BACKGROUND OTHER THAN RADON	1 mSv	240×10^6	240×10^3 Person-Sv
RADON - AVERAGE BACKGROUND (0.2 WLM/YR)	2 mSv	240×10^6	480×10^3 Person-Sv
- MOST HIGHLY EXPOSED AT >2 WLM/YR	20 mSv+	0.33×10^6	6.6×10^3 Person-Sv
MEDICAL EXPOSURES	0.9 mSv	240×10^6	220×10^3 Person-Sv
OCCUPATIONAL	2.5 mSv	1.36×10^6	1.55×10^3 Person-Sv

"expected" level to the public (3) of 0.05 rem/yr. Associated with this is a maximum risk of 0.5×10^{-5}/yr and an accumulated risk of 1.2×10^{-4}. Background to the whole body (excluding radon) gives rise to an annual risk, at the average level of 0.1 rem/yr, of 10^{-5}/yr after 50 years and a total accumulated risk of 0.024% in 50 years or possibly up to about 0.05% lifetime. Radon exposure to the lungs is an additional factor. The effective dose equivalent from the average radon background level of 0.2 WLM/yr in the USA is estimated to be 0.2 rem/yr, i.e., the risk of lung tumors from the radon background is twice as great as the risk of all cancers from other background sources. Furthermore, the maximum levels of radon can reach above 10 times the average, i.e., 2 WLM/yr, which is the NCRP's proposed limit for remedial action, and at this level the annual risk reaches 2×10^{-4}/yr and the total risk about 1% lifetime.

While these levels are experienced by relatively few people, estimated at about 330,000 persons above 2 WLM/yr, the collective dose is substantially more than that involved in exposures to the radiation work force. Furthermore, the information we have is based on a limited data base and more data are needed to determine whether the problem is more or less serious than we currently believe.

Emergency Levels

Formerly, levels were promulgated for working circumstances involving, for example, emergency situations arising from accidents. Recommended levels (2) for emergencies were 100 rem lifesaving and 25 rem non-lifesaving. Neither NCRP nor ICRP now recommend an emergency level, and they suggest the use of volunteers. The former numbers are useful as guides, perhaps in public as well as occupational circumstances. At 100 rem, the risk of acute effects is small but the cancer risk will follow Figure 1 and continue for 50 years, reaching a maximum annual risk of perhaps 10×10^{-4}/yr and a 50-year risk (also the lifetime risk) of 2.5×10^{-2}. (The risk coefficient to be associated with 100 rem is that associated with higher doses and dose rates *viz* 5×10^{-5} for leukemia x 5 for all cancers, not the 2×10^{-5} for leukemia used at low doses.)

For 25 rems, the situation is equivocal; the dose is neither high nor low, thus the base leukemia risk coefficient is between 2 and 5 x 10^{-5}, and the total risk is between 2.5 and 6 x 10^{-3} lifetime.

Space

Since 1970 radiation exposures to individuals on missions in space has been limited by the recommendations of the Space Science Board (13). These include a lifetime limit of 400 rems which would probably be received in smaller amounts per mission. Thus, the risk is probably between 4 and 10 x 10^{-2}, i.e. 4-10% lifetime, a rather considerable addition to the normal risk of cancer. Even so, the annual risk, which may be about 0.5% over a 20-year period, may not be large compared with other risks the astronauts face. Further study of space radiation hazards seems warranted (21).

CONCLUSIONS

An examination of the risks associated with current protection levels seems to indicate that occupationally, the average exposures are satisfactory, but that some additional limitation may be necessary for exposures close to the limit and NCRP is considering this. Further consideration needs to be given to exposures to high LET radiation since RBEs may require adjustments in the quality factor. Public: Exposures to man-made sources are generally small compared with those from natural background, only medical procedures being close to the same range. The most important source of exposure in many countries is radon and both ICRP and NCRP have drawn attention to this and proposed similar action levels. Emergency situations and space radiation activities are special circumstances involving relatively few people and higher risks of other kinds, so different judgments need to be employed.

References

1. NCRP. NCRP Report 17. National Council on Radiation Protection and Measurements, Bethesda, MD, 1954.

2. NCRP. NCRP Report 39. National Council on Radiation Protection and Measurements, Bethesda, MD, 1971.

3. ICRP. ICRP Publication 26. Annals of the ICRP 1, #3, Pergamon Press, N.Y., 1977.

4. W.K. Sinclair. In Proceedings of the 16th Midyear Topical Meeting: Epidemiology Applied to Health Physics, pp. 181-198, Albuquerque, NM, January 1983.

5. UNSCEAR. United Nations Scientific Committee on the Effects of Atomic Radiation, United Nations, N.Y., 1977.

6. NAS-BEIR. BEIR III Report. National Academy of Sciences, Washington, DC, 1980.

7. NCRP, Basic Radiation Protection Criteria. Report to be published. National Council on Radiation Protection and Measurements, Bethesda, MD.

8. H. Kato and W.J. Schull. Radiation Research 90, 395-432, 1982.

9. W.K. Sinclair. Yale Journal of Biol. and Med. 54, 471-484, 1981.

10. W.K. Sinclair. Radiology 138, 1-9, 1981.

11. W.K. Sinclair. In Proceedings of the 16th Annual Meeting of the National Council on Radiation Protection and Measurements. 1980.

12. J. Auxier. TID-27080, USERDA Technical Informational Center, Oak Ridge, TN, 1977.

13. W. Loewe and E. Mendelsohn. Health Physics, 41, 663-666, 1981.

14. U.S.-Japan Joint Workshop for Reassessment of Atomic Bomb Radiation Dosimetry in Hiroshima and Nagasaki. Feb. 16, 1983, Radiation Effects Research Foundation, Hiroshima, Japan.

15. W.K. Sinclair. In Proceedings of the 7th International Congress of Radiation Research, Amsterdam. July 3-8, 1983.

16. M.W. Charles and P.J. Lindop. Journal of the Society for Radiological Protection 1, 15-19, 1981.

17. T. Straume and R. L. Dobson, Health Physics, 41, pp. 666-671, 1981.

18. S. Jablon. In Proceedings of the 16th Midyear Topical Meeting: Epidemiology Applied to Health Physics, pp. 167-180. Albuquerque, NM, January 1983.

19. NCRP Statement on Dose Limit for Neutrons. National Council on Radiation Protection and Measurements, Bethesda, MD, Feb. 1980.

20. W.K. Sinclair. In 8th Symposium on Microdosimetry, Commission of the European Community, Brussels, EUR 8395, pp. 1-37, 1983.

21. W.K. Sinclair. In Workshop on Radiation Safety Standards, Proceedings of the 24th COSPAR Meeting, Ottawa, Canada, May 1982, in Advances in Space Research, 3, 151-159, 1983.

THE EPIDEMIOLOGY OF RADIATION CARCINOGENESIS IN MAN

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INTRODUCTION

The opportunities and the difficulties of human epidemiology are well illustrated in the studies that have been made on the effects of ionizing radiation in man. Moreover, the development of our knowledge on this subject shows a characteristic evolution: firstly, an indication that such effects occur; then, rigorous evidence of the frequency with which they have occurred following radiation exposure; and finally, quantitative data relating the frequency of their occurrence to the size of the absorbed dose in the tissues in which they occurred.

The course of this evolution differed somewhat for different types of radiation effect. In some, the nature of the effect indicated the radiation exposure as its likely cause; as when characteristic skin changes were limited to a heavily irradiated area, or when posterior polar cataracts developed following high exposures of the lens in therapy. Here the problem was essentially one of recording the frequency of such typical changes, and the levels of dose after which they occurred.

For other effects, such as the induction of malignancies where types of cancer induced were identical in behaviour and microscopic appearance with those occurring "naturally", the major problem was different. It was not simply to determine the number that followed a given radiation exposure, but to distinguish this number from that which would have incurred in the absence of the radiation. At high dose, there was no difficulty: cancer induction in skin and deeper tissues was evident from early days, since the numbers of such cancers arising in tissues that had been heavily irradiated in the course of therapy very obviously exceeded the numbers to be expected by chance (1, 2). The detection of cancer induction at lower dose, however, had to wait for a further 30 years until statistics were obtained to compare cancer frequencies in large irradiated and unirradiated populations (3, 4, 5); and it was 10 years later still before reliable data were obtained relating frequency of malignancies with size of dose (6, 7).

Basic problems in the epidemiological study of radiation carcinogenesis arise, therefore, in identifying two populations of people: one of which has been exposed at a known time to a known dose of radiation of appropriate size, quality, and distribution; and another population which can be regarded as identical in all relevant respects except for not having been so exposed.

A major additional problem is caused, however, by the variable but long intervals of time between the radiation exposure and the detectability of any resulting cancer. If the question were only whether radiation induces cancer, or a particular type of cancer, this might not cause difficulty: some excess of cancer might already become detectable within 10 or 15 years. When, however, the need is to determine the total risk of cancer following a given exposure, the surveillance of the exposed population needs to be both long and comprehensive: ideally extending for more than 30 years, and quite possibly for the whole remaining lifetime of the exposed individuals. The difficulties of complete identification and recording of causes of deaths, or, worse, of serious illnesses, for these periods of time in the necessarily large population groups, are obvious.

In view of these problems, it is remarkable how much reliable information has in fact been obtained on the frequency with which different forms of cancer follow exposure of the various body organs at moderate dose (8, 9). In a variety of populations exposed under widely differing circumstances, broadly consistent estimates of these risks have been obtained. The different requirements for rigorous epidemiological analysis have, however, been satisfied in different degrees in the various population groups exposed for radiotherapeutic or diagnostic reasons, or from occupational or natural sources, or by direct exposure to nuclear weapons or to fallout from them (8). It is therefore necessary to review the extent to which it has been possible, in the study of such populations, to satisfy these numerous requirements: the need for prolonged surveillance; the complete and accurate determination of all causes of death or major disease; and the correct ascertainment of the time at which doses were given, their body distribution, their size and radiation quality; and, above all, the suitability of the control group with which the frequency of effects is compared.

LENGTH OF SURVEILLANCE

In only two types of induced malignancy has the increased mortality apparently ceased, or almost ceased, within 30 years of the radiation exposure. In the extended life span study group in Hiroshima and Nagasaki, about 85 deaths from leukaemia had occurred in irradiated survivors, in excess of the number expected, within 25 years of the bombs. A further excess of 5 deaths occurred in the following 8 years (10). Similarly, in 898 patients treated for various diseases by intravenous injection of the shortlived radium 222, an excess mortality above expectation of 53 from bone sarcoma had occurred within 25 years of treatment (11), with only one further such death in the following 8 years. In both surveys, the intervals between exposure and deaths are log-normally distributed, the statistics of the two distributions being closely similar; with log. mean doses and standard deviations of about 11.5 years and 1.85 for the deaths from leukaemia, and of 9.5 years and 1.65 for those from sarcoma. Bone sarcomas continue to develop for longer periods after incorporation of radium 226 (12), but here of course the prolonged retention of this long-lived isotope in the skeleton results in continuing irradiation of bone cells.

For other forms of cancer, with the possible exception of those following irradiation in utero, no estimate can yet be made of the mean "latent period" between irradiation and the development of induced cancers. In published surveys of cancers induced by therapeutic radiation, this mean value has commonly lain between 20 and 30 years (8). Such figures are likely to underestimate the true mean latency, however, since late developing cancers are less likely to be detected during periods of clinical follow-up. This is particularly likely to be the case when radiotherapy has been given for malignant disease or in the elderly. Even in studies of patients treated for benign diseases with long survival after treatment, and where actuarial corrections are properly made for other causes of death, such patients may not be followed up for long periods if there is no clinical need to do so, and perhaps also if a life-long follow-up to detect or exclude a very low incidence of induced cancer might cause or maintain an unwarranted anxiety.

In Hiroshima and Nagasaki, however, an excess mortality for malignancies other than leukaemia continues at more than 30 years since exposure (10), and shows no significant decrease in rate when expressed in relation to the surviving number of irradiated people. A decrease in the excess mortality of individual types of cancer cannot, however, be excluded.

The maintenance of an accurate long surveillance is important, therefore, if cancers are thought of as developing after a fixed distribution of latent intervals, as implied by a so-called absolute predictive model of cancer induction. It is even more important according to the relative predictive model (9), which postulates that, following exposure, induced cancers develop mainly at the ages at which naturally occurring cancers of the same types mainly develop. In this case the long-term follow-up is crucial, since so many types of cancer have an increasing incidence in old age. Few studies have been made in which exposed populations have been followed for long periods into old age. It remains unclear, therefore, whether exposure at younger ages will cause increasing numbers of cancers to appear in old age, or whether - as apparently with leukaemia and bone sarcoma - induced cancers will appear at a decreasing rate as old age approaches. The continuing study of survivors at Hiroshima and Nagasaki is of particular importance for this reason, as it is for the estimation of any variation of latency or risk with age at exposure, sex, and the size of the dose delivered.

ASCERTAINMENT OF CAUSES OF DEATH OR DISEASE

Mortality

In determing the frequency with which cancer develops in irradiated populations, records of mortality are likely to be more reliable than those of incidence. When death results from cancer, this fact is likely to be correctly recorded, at least in countries in which death certification recording the cause of death is obligatory and where such evidence is available to epidemiologists. It is unimportant that the epidemiologist shall have access to the death certificate itself, provided that his information on the cause of death in each individual can be linked to the radiation exposure of that individual. In countries or districts in which death certificates are treated as confidential documents, this linkage could and should be made by the registering authority, without the causes of death of named individuals necessarily being released outside that authority.

While death certificates are likely to be reasonably reliable in determining the frequency with which death is due to cancer, they are less reliable in determining the organ or tissue in which the cancer developed, unless an autopsy was carried out, or surgical or biopsy tissue had been obtained. In particular, cancers which have caused death following spread to bone, liver or lungs from their primary site or origin, may be recorded as cancers of these secondarily metastasised organs. This may be unimportant in estimating the total carcinogenic effect of uniform whole body radiation. It is critical, however, in determining organ sensitivity to radiation, or the dose response relationship for individual types of cancers.

Incidence

The recording of all cancers which are successfully treated and which do not cause death depends on continued clinical surveillance of the exposed population, or retrospective enquiries about all surgical or other treatments that they have received. Both these sources of information are difficult to achieve comprehensively over long time periods. Only the former method is likely to be reliable, unless efficient cancer registries have operated over the whole of the relevant periods, and are accessible to enquiry. Since rather few cancer registries were operating comprehensively before about 1960 (13), adequate incidence data are likely to be available only from clinical surveys with full follow-up of individuals, or from the exposed populations in Hiroshima and Nagasaki, or of the Marshall Islands, or in workers under medical supervision in the same employment over long periods.

The amount of "detriment" due to the induction of curable cancers, relative to that from cancers which prove to be fatal, is potentially most important for readily treatable types of cancer such as basal and squamous cell tumours of skin, and the follicular epithelial cell cancers of the thyroid which are induced by radiation. For most of the skin cancers of these types, the surgical or other treatment is usually so simple that their occurrence is not consistently reported in tumour registries, and may be overlooked even during follow-up of exposed individuals. Also, of the follicular and papillary thyroid cancers, many may persist without causing symptoms or ill health for long periods, so that estimates of their frequency will depend critically upon how regularly an exposed population is examined, or is subjected to operation if minor nodularities are felt in the gland.

In view of the problems in ensuring a full record of cancer incidence, rather than of mortality, in exposed populations, it is arguable that the frequency of curable cancer induction in different organs may be less reliably estimated by studies of incidence than by assuming that radiation-induced malignancies of any histological type have the same behaviour and curability as naturally occurring cancers of the same type, and relying on the frequency with which such types of cancer can normally be cured by efficient forms of treatment. If this assumption is warranted, the induction of curable cancers can be inferred from the more reliable data on the induction of fatal cancers of each type.

ESTIMATION OF DOSE

It is obviously necessary to determine the dose at which an organ has been exposed, in any survey of the risks of inducing cancer in that organ. It is important also for determining whether the cancer induction is due to the radiation, since a correlation between cancer incidence and dose in different sections of the exposed population supports the hypothesis that the cancers are due to the radiation rather than to some other carcinogen to which the population may have been exposed.

Merely the assessment of the dose to the relevant tissues, however, has frequently presented problems. In Hiroshima and Nagasaki, the assessment of weapon yield, the energy of radiations at ground level, and the transmission through buildings are under review. In some studies of irradiation in utero (14), the probable dose to the foetus can only be estimated according to the criteria of normal radiological practise at the relevant times (15). The estimates of dose to breast tissues from multiple fluoroscopies, to which an increased breast cancer incidence is attributed, depend upon memories of the likely average duration of the exposures, and of the frequency with which patients faced the x ray tube (16). The thyroid doses in the Rongelap islanders are assessed, as regards internal exposure, from the urinary excretion of radioiodines at 15 days after the fallout occurred, and from an estimated relationship between excretion rate and thyroid content (17). Similarly the dose from radium 226 incorporated in bone, to the endosteal cells from which bone sarcomas are believed to arise, is estimated from the likely distribution of radium in bone at different times after intake, and the assumed proximity of endosteal cells to the bone surface: a proximity which may be affected by inflammatory reactions and fibrosis at the surface of radioactive bone (18).

Such examples could be multiplied, and we must recognise the difficulties in estimating the dose to the bronchial or lung alveolar cells of uranium miners from sampling of the radon concentration at fixed positions in mine galleries; the doses to various organs from the different areas of spine irradiated in patients with ankylosing spondylitis; the need to discount "wasted radiation" during the continuing irradiation of bone cells after a sarcoma has been initiated

by radium 226; the "self-absorbtion" of alpha radiation in Thorotrast particulates or in surrounding areas of tissue necrosis (19); or for the fact that today's incidence of lung cancer in uranium miners is attributable to their exposure to radon daughter concentrations of 20 or 30 years ago.

It is also important to emphasise that some cancers arise from types of cell which are not uniformly distributed within the organ in which they are located. If the absorbed dose also is non-uniformly distributed within the organ, it is essential to estimate the dose to the relevant cells, and not simply the mean dose to the organ. This led previously to confusion from using mean bone dose in relation to bone sarcoma induction, and is causing uncertainty if mean lung dose is used in estimating the induction rate of cancers of lung arising from bronchial and alveolar cells (20). We must think of cancers of cell types and not merely of organs.

COMPARISON POPULATIONS

It can rarely have been possible to compare cancer rates in an exposed population with those in a control population which was truly identical except by not having been equally exposed. Considerable care has therefore been taken, in most surveys, to ensure that both populations are similar in all respects that are likely to be relevant.

Thus, in populations exposed for medical reasons, either in therapy or in diagnosis, cancer frequencies can legitimately be compared with those in the healthy general population if the disease treated or diagnosed is very unlikely to be associated with increased cancer incidence in any organs studied, either by a direct effect, an observed clinical association, or a linkage of inherited predispositions. Thus, tenia capitis seems unlikely to be associated with thyroid or brain cancer, but Modan et al. compared the results of scalp irradiation in tenia patients with cancer incidences in sex and age matched members of the same families (21). On the other hand, ankylosing spondylitis is known to be associated with ulcerative disease of the colon, and in this condition there is an increased incidence of cancer of the colon. Court Brown and Doll therefore excluded colonic cancers from their totals of cancers of heavily irradiated sites (22). A linkage between diseases, at the low frequencies of association which would be sufficient to falsify the interpretation of effects of low doses, may however be unexpected. In studies of the effects of the radioiodine treatment of thyroid overactivity, there was some suggestion of a slightly increased frequency of leukaemia in patients with this disease, whether treated by radioiodine or by surgery (23).

In several other surveys, the cancer frequencies of radiation treated patients have been compared, not with the expected frequencies in healthy people, or in people from the same families and localities, but with those in patients with the same disease but untreated by radiation (24). This clearly represents a stricter control. Even here, however, it is necessary to be sure that the decision to use radiation, rather than an alternative treatment, was not determined by the severity of the disease, the locality in which the treatment was given, or even the economic status of the patient - to ensure that the comparison does not still contain significant sources of bias: either in likelihood of cancer, or in efficiency of its subsequent ascertainment if it does occur.

Similar problems arise in other connections. Studies of the cancer incidence in areas of high natural background radiation are rendered difficult because of differing medical facilities, or of cultural, economic or dietary conditions, in areas of high altitude or of volcanic origin. Or the effects of occupational exposure to radiation may be confounded, not only by the "healthy worker" effect, but by an associated exposure to carcinogenic chemicals, as - in the case of

uranium miners - to dust, arsenic and diesel fumes.

It is for such reasons that radiation is more securely identified as the cause of an increased cancer frequency if the amount of the excess can be related to the mean size of dose in subgroups of the exposed population. In these circumstances the least exposed members of the population afford a basis for comparison with the most exposed, and may form a more valid comparison group than populations living in different localities or circumstances, or than those of the nation as a whole. Thus it has been concluded for some years that the survivors in Hiroshima and Nagasaki who had been exposed at low dose, are likely to form a more reliable comparison group for those exposed at high dose, than the total Japanese population (25), in which differences in cancer incidence are seen in different town and country areas and at different times. And when risk estimates are obtainable from medical, occupational or atomic bomb exposures, the required information in any case depends on a linear or other regression of cancer mortality or incidence upon dose.

STATISTICAL CRITERIA

Overall, however, the greatest difficulty in many epidemiological surveys of radiation, as of other, effects has been in obtaining even approximate estimates of the size of a small excess of induced cancers, above a large expected number of naturally occurring ones. The commoner a form of cancer is normally, the greater in the difficulty in detecting a given increase. More work needs to be done on this "signal to noise" ratio, between the risk per unit dose of inducing different types of cancer, and the frequency of their natural occurrence. As our attention focuses down from the total excess of all types of cancer induced by whole body radiation, to the excess of one particular type such as leukaemia, we may gain by examining a type which is more readily induced than is cancer as a whole. We lose, however, by dealing with smaller numbers, with the inevitably greater statistical difficulties that this involves. But, broadly speaking, the type of cancer in which an increase will be most easily detected, will be that type with the lowest ratio between the natural incidence and the square of the induction rate per unit dose (26).

The difficulties of a reliable detection of an excess of cancers following radiation are, however, very severe at low dose. If an excess was just detectable in a survey in which the mean dose to the irradiated organ was one sievert, to detect the excess following a mean dose of one millisievert with the same statistical reliability would require a survey, not a thousand times larger, but a million times larger in person-years, if the induction rate per unit dose was the same. Even if the induction rate per unit dose was ten times larger at low dose, the survey would still need to be 10,000 times larger. And even minor sources of bias could vitiate the results of any more limited surveys. Both the interpretation, and the planning, of studies at low dose need to bear these points in mind.

And finally, the epidemiological study of radiation carcinogenesis contains another trap for the unwary. If a survey is undertaken to detect whether a particular type of cancer is increased in frequency following radiation, the numbers of that type of cancer can be compared in the exposed and unexposed populations, and the significance of any excess assessed by conventional methods in terms of the probability that the excess might have occurred by chance. If however the exposure involved many or all body organs, and an excess of cancer was observed in one of them, the assessment of its significance must obviously be different, and requires to be more stringent. Specifically, a value of $p = 0.02$ may be acceptable if the survey was planned to detect an excess of cancer in a specified organ or tissue under review. If, however, a value of $p = 0.02$ is calculated for the organ in which the greatest cancer excess was observed, in a

survey in which the cancer incidence in 15 organs was reviewed, the probability that this result may have been due to chance has a value of p which is given (27) by

$$p = 1 - (1 - 0.02)^{15} = 0.26$$

which is less than convincing. The result would require repeated survey to test the reproducibility of this finding.

The lower the dose, the greater must in any case be the uncertainties and risks of minor bias in any purely epidemiological surveys; and the greater must inevitably be our reliance upon inference, on all established radiobiological grounds, from the extensive data that have now been obtained from epidemiological surveys at moderate dose levels in man.

REFERENCES

(1) Jagić, N.von, Schwarz, G. and Siebenrock, L. von, Blutbefunde bei Röntgenologen. Berl. klin. Wochenschr. 48 1220-1222 (1911).

(2) Hesse, O., Symptomatologie, Pathogenese und Therapie des Röntgenkarzinomes. (Barth, Leipzig: 1911).

(3) Martland, H.S., The occurrence of malignancy in radioactive persons. Amer. J. Cancer, 15 2435-2516 (1931)

(4) March, H.C., Leukaemia in radiologists. Radiology, 43 275-278 (1944).

(5) Dublin, L.I. and Spiegelmann, M., "Safety criteria in atomic energy", in Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Vol. 13, pp 315-318. (United Nations, New York: 1948).

(6) Simpson, C.L., Hempelmann, L.H. and Fuller, L.M., Neoplasia in children treated with x-rays in infancy for thymic enlargement. Radiology, 64 840-845 (1955).

(7) Court Brown, W.M. and Doll, R., Leukaemia and aplastic anaemia in patients irradiated for ankylosing spondylitis. Medical Research Council Special Report 295 (HMSO, London: 1957).

(8) United Nations Scientific Committee on the Effects of Atomic Radiation: Report to the General Assembly, 1977. (United Nations, New York: 1977).

(9) BEIR Committee (on the Biological Effects of Ionizing Radiation). The effects on populations of exposure to low levels of ionizing radiation. National Research Council, 1980. (National Academy Press, Washington DC: 1980).

(10) Kato, H. and Schull, W.J., Studies of the mortality of A-bomb survivors. 7, Mortality, 1950-1978: Part 1. Cancer Mortality. Radiation Research, 90 395-432 (1982).

(11) Mays, C.W. and Spiess, H., "Bone sarcoma in ^{224}Ra patients", in Radiation Carcinogenesis: Epidemiology and Biological Significance. Boice, J.D. and Fraumeni, J.F. eds. (Raven Press, New York: in press, 1983).

(12) Rowland, R.E., Stehney, A.F. and Lucas, H.F., Dose-response relationships for radium induced bone sarcomas. Health Physics, 44 (Supp.1) 15-31 (1983).

(13) Cancer incidence in 5 continents, Vol. 1 (Doll, R., Payne, P. and Waterhouse, J. eds.) International Union against Cancer (Springer-Verlag, Berlin: 1966).

(14) Stewart, A. and Kneale, G.W., Radiation dose effects in relation to obstetric x-rays and childhood cancers. Lancet 1 1085-1088 (1970).

(15) United Nations Scientific Committee on the Effects of Atomic Radiation: Report to the General Assembly, 1972 (United Nations, New York: 1972).

(16) Boice, J.D., Monson, R.R. and Rosenstein, M., Cancer mortality in women after repeated fluoroscopic examinations of the chest. J. Nat. Cancer Inst. 66 863-868 (1981).

(17) Conard, R.A. et al. A twenty-year review of medical findings in a Marshallese population accidentally exposed to radioactive fallout. Brookhaven National Laboratory report BNL50424 (Upton NY: 1975).

(18) Lloyd, E.L. and Henning, C.B., Cells at risk for the production of bone tumours in radium exposed individuals: an electron microscope study. Health Physics 44 (Supp.1) 135-148 (1983).

(19) Kaul, A. and Muth, H. Thorotrast kinetics and radiation dose. Rad. and Environm. Biophys. 15 241-259 (1978).

(20) Jacobi, W. Relations between the inhaled potential α-energy of ^{222}Rn and ^{220}Rn daughters and the absorbed α-energy in the bronchial and pulmonary region. Health Physics 23 3-11 (1972).

(21) Modan, B., Mart, H., Baidatz, D., Steinitz, R. and Levin, S.G., Radiation-induced head and neck tumours. Lancet 1 277-279 (1974).

(22) Court Brown, W.M. and Doll, R., Mortality from cancer and other causes after radiotherapy for ankylosing spondylitis. Brit. med. J., 2 1327-1332 (1965).

(23) Saenger, E.L., Thoma, G.E. and Tompkins, E.A., Incidence of leukaemia following treatment of hyperthyroidism. J. Amer. med. Assoc., 205 855-862 (1968).

(24) Smith, P.G., Doll, R. and Radford, E.P. Cancer mortality among patients with ankylosing spondylitis not given x-ray therapy, Brit. J. Radiol. 50, 728-734 (1977).

(25) Beebe, G.W., Kato, H. and Land, C.E. Studies of the mortality of A-bomb survivors. 4. Mortality and radiation dose, 1950-1966. Radiation Research, 48 613-649 (1971).

(26) Pochin, E.E., "Need for future epidemiological studies", in Radiation Carcinogenesis: Epidemiology and Biological Significance. Boice, J.D. and Fraumeni, J.F. eds. (Raven Press, New York: in press, 1983).

(27) Tippett, L.H.C. The methods of statistics, 4th Edn. at p92 (John Wiley, New York: 1952).

RADIOLOGICAL PROTECTION CRITERIA FOR RADIOACTIVE WASTE REPOSITORIES

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INTRODUCTION

This paper explores the possibility of extending the use of the "system of dose limitation" to the design and safety evaluation of repositories for high level radioactive waste. Individual dose - limitation and optimization of protection cannot be directly used for judging the safety level of systems intended either to prevent or to limit potential future exposures. Nevertheless, it appears that the basic principles can conceptually be extended to the limitation of such uncertain exposures.

The third component of the system of dose limitation, the justification of the practice, relates to the larger matter of energy production by nuclear means. The net possitive benefit must be associated with that practice "in toto" and not with an unavoidable part of it, such as the disposal of the wastes.

Disposal of radioactive waste involves isolation of the radioactive material from the human environment over a sufficient period of time. Exposures of man that could still occur would be due to mechanisms that may be described as either (1) "Normal" release mechanisms, leading to a predictable exposure pattern in space and time, or (2) Random disruptive event, for each of which a probability of occurrence and an exposure pattern can be postulated. The main difference is that the normal release is expected to occur first after a retention period which is long enough to permit the activity to decay to clearly acceptable values, while disruptive events may cause releases before there is sufficient decay to make the exposures acceptable.

The main objectives in radiation protection are to keep individual exposures below recommended limits and to reduce them further as far as it is reasonably achievable. The purpose of the individual dose limits is to limit each individual's probability of severe harm at an acceptable level. The extent to which doses are as low as reasonably achievable is determined by optimization of protective efforts related to a given radiation source or practice. Radiation protection is therefore concerned with both individual-related efforts (risk limitation) and source-related efforts (optimization of protection).

INDIVIDUAL RELATED REQUIREMENTS

The dose limits recommended by ICRP are intended for the total dose from all sources (excluding natural background and medical exposures of patients). For the contribution from any one particular source, competent authorities are expected to set an upper bound which should only be some fraction of the ICRP dose limit if exposure from other sources is possible (1)(2).

The use of the concepts of limit and upper bound in the case of high-level waste repositories involves three main conceptual difficulties (3): a) the question of the level of ambition for protecting individuals in the far future; b) the global contribution of each repository to the "per caput" dose; and c) the problem of potential exposures of probabilistic nature.

The first is a policy question which obviously can have several answers. The crucial question is whether we wish to protect future individuals at the same level of ambition as we protect present individuals. Some feel that we have more obligation to the present than to an uncertain future. Others feel that we have a special obligation to the future when we cause risks for our own benefit. Others, again, would state that present benefits are indirectly prerequisites for future benefits and well being. In this paper it is assumed that the ambition is to protect all individuals alike, irrespective of time and location. A different policy would not change the following principles, only the quantitative results.

The second conceptual problem relates to the selection of an upper bound. It is possible to postulate a generic upper bound for the annual dose contribution from the practice of radioactive waste isolation, which would limit the dose to any critical group resulting from all repositories. Such generic upper bound would be of little use in practice. What is required is an upper bound for the annual dose due to a given repository. In this case, however, it is necessary to take account of the "per caput" global contribution of all the repositories which would add to the local dose contribution of the repository under consideration.

The "normal" release situation and its resulting exposure pattern, by definition, have a probability which may be considered equal to one. The design requirement would therefore be that at all times

$$H_{L,k} + \Sigma_i H_{g,i} \leq f H_{lim}$$

where $H_{L,k}$ is the annual dose contributed locally to its critical group by repository k, $H_{g,i}$ is the global "per caput" annual dose contribution of repository i (the summation being carried over all repositories), f is the fraction of the ICRP dose limit considered appropriate for waste isolation in general, and H_{lim} is the dose limit.

The establishment of "upper bound" conditions, therefore, must take account of the interplay of the local dose contribution and the "per caput" global dose contribution. This is a particular case of a problem that was discussed in another paper related to limitation of releases of radioactive materials to the environment (4).

The relation presented above could also be expressed as

$$H_{L,k} + \sum_i r_i \, H_{L,i} \leq f \, H_{lim}$$

where r_i is the ratio of the "per caput" dose to the local critical group dose (both due to repository i), $H_{L,i}$ is the local critical group annual dose, and the other symbols have been defined previously.

It seems reasonable to postulate that the "per caput" contribution of a repository is proportional to the amount of uranium utilized to produce the waste in the repository (3). This implies that

$$\frac{r_i \, H_{L,i}}{\sum_i r_i \, H_{L,i}} = \frac{W_i}{W_t}$$

where W_i is the amount of uranium originating the waste in repository i and W_t is the total amount of uranium which is predicted to be used for nuclear power generation.

The limiting equation for doses in the critical group for repository k can therefore be written as

$$H_{L,k} \left(1 + r_k \frac{W_t}{W_k} \right) \leq f \, H_{lim} \quad ,$$

and the upper bound, U_k, for the annual dose in the critical group is given by

$$U_k = \frac{f \, H_{lim}}{1 + r_k \frac{W_t}{W_k}}$$

It must be indicated that the upper bound will differ significantly from the simple value $f \, H_{lim}$ only in the cases where $r_k(W_t/W_k)$ is not small. For planning purposes, and similar to the case of upper bounds used at present for some nuclear installations, the value of $f \, H_{lim}$ can be taken to be in the order of 10^{-1} mSv in a year.

The third conceptual problem in applying individual dose limitation to radioactive waste repositories relates to the potential exposures resulting from probabilistic disruptive events.

It should be recognized that even in "normal" release conditions, the occurrence of detrimental biological effects is probabilistic. For radiation protection purposes, the probability of stochastic effects, namely fatal cancer and serious genetic effects, is taken to be approximately $2 \, 10^{-2}$ per sievert of effective dose equivalent(1). The probability of non-stochastic effects, above the threshold increases from cero to one over a usually small range of doses. For example, for mortality due to the accute radiation syndrome resulting from whole-body irradiation, the probability at

less than 2 gray is practically cero, while it is nearly one above 6 gray.

In the case of random disruptive events one deals with the probability of the event and with the resulting doses in the group of individuals receiving the highest doses. Regarding radiation effects one deals, therefore, with effects of second order stochasticity. For example, the probability of dying (risk) of a member of the public due to a given potential disruptive event can be expressed as $R = P_1 P_2$, where R is the "risk", P_1 is the probability of occurrence of the event which would result in a dose H to the member of the public under consideration, and P_2 is the conditional probability of death given dose H.

If several potential situations are taken to be possible, the risk R defined in the previous paragraph becomes

$$R = \sum_i P_{1i} P_{2i}$$

It should be realized that P_2 is proportional to dose at the lower doses (where only stochastic effects are possible), but increases steeply with whole-body dose at values above 2 Gy due to acute non-stochastic effects.

Probability values must be assigned to the few disruptive events that are expected to cause significant exposures if they occur. Event-tree assessments will yield probability values p_i for a number of exposure situations (i) due to various sequences of events. In each exposure situation a maximum annual dose (H_i) in time and space may be identified. The simplest approach would be to assess these maximum doses as if they occurred simultaneously in one and the same critical group. This will overestimate the probability of harm to any given person but will grossly simplify the assessment(3).

On the assumption that the maximum probability of harm committed in a year, given an exposure situation (i), is a function of the maximum annual dose that would occur in that situation (H_i), the maximum value of the total probability of harm, R_{max}, that is committed for one year can never exceed the direct summation of the maximum values for each exposure situation:

$$R_{max} < \sum_{i=0} p(H_i)\ p(\text{harm}\ \ h_i)$$

It is proposed that the basic objective is to ensure that the maximum individual risk of a member of the public due to potential disruptive events in a repository does not exceed the risk due to "normal" radioactive releases. The order of magnitude of the upper bound suggested in previous paragraphs (0.1 mSv in a year) corresponds to a risk of lethality of the order of 10^{-6} in a year. Therefore, the proposed objective can be stated as

$$R_{max} < 10^{-6}$$

where R_{max} is the maximum value of the probability of lethality committed for one year.

If several disruptive event sequences are considered (the order of magnitude of the number being ten), then on the average their individual risk contribution should not exceed a value $R_i = 10^{-7}$ in a year. The "acceptable" probability of occurrence of a sequence giving dose H_i would be $(\leq 10^{-7}/10^{-2}\,Sv^{-1}\cdot H_i)$ in the region of doses H_i where only stochastic effects can occur.

At higher dose values the "acceptable" probability of occurrence would be $\leq 10^{-7}/f(H_i)$, where $f(H_i)$ is a non-linear function of dose which becomes 1 at high doses. The annual probability of disruptive sequences yielding non-stochastic lethal doses should not exceed therefore the order of 10^{-7}.

The procedure described above for applying individual dose limitation to probabilistic disruptive events has been expressed in a "criterion curve"(3), similar to curves used in the probabilistic nuclear safety philosophy(5).

SOURCE-RELATED REQUIREMENT

A source of both "normal" and potential exposures corresponding to only low individual risks might still cause a substantial collective consequence. A source-related requirement is therefore to optimize the level of protection to be applied to the source in order to keep the consequence as low as reasonably achievable.

ICRP Publication 37(6) presents the basic principles of optimization, explains its use as a decision tool, discusses the criteria involved in the requirement and describes the conceptual basis and the techniques of cost-benefit procedures for optimization.

The application of such techniques of optimization to the engineering of repositories (barriers, containers, backfillings, etc.) involves conceptual problems, again related to the time scale and to the possibility of disruptive events.

Optimization of protection may be based on the total collective doses and on the probabilities of the various events, as long as no high individual doses causing non-stochatic effects are deemed to be possible. It is possible to calculated the expectation value of the total collective dose in addition to the collective dose that can be assessed from the "normal" exposure situation. If a possible sequence with a probability p_i causes a collective dose S_i, the expectation value is $\bar{S}_i = p_i S_i$. However, the inherent uncertainty of the expectation is large, the standard deviation being $\bar{S}_i/\sqrt{p_i}$. The use of the expectation value in protection optimization may therefore be invalid in the case of contribution of low probability-high consequence situations(7). It would be even more so if high doses, beyond the range of stochastic linearity, are possible.

A further problem is the time scale involved. Due to the presence of very-long life nuclides, an assessment of the complete collective dose commitment is highly speculative. In optimization, however, one deals with differences of collective doses between different protection options. The time period of interest is there-

fore the period in which the alternative protection levels (engineering solutions) have an influence on the exposure pattern, because the longer exposure tails would cancel when substracting collective doses of different options. The period of interest for the assessment of the relevant incomplete collective dose commitments (a few thousand years) would make such assessments more reasonable that it would appear from the half lifes of the nuclides involved. For these assessments, procedures similar to those used by UNSCEAR(8) can be applied.

REFERENCES

(1) ICRP Publication 26. 3. ed. Oxford, Pergamon Press, Annals of the ICRP v. 1: N° 3, 1977.

(2) International Atomic Energy Agency. Safety Series N° 9. STI/PUB/607. Vienna 1982. 172 p.

(3) Beninson, D. and Lindell, B. IAEA-CN-43/43. International Conference on Radioactive Waste Management, Seattle, Wa., USA, 16-20 May 1983.

(4) Beninson D. IAEA-SMA-258/52.

(5) González, A. International Meeting on Thermal Nuclear Reactor Safety, Chicago, Aug. 29 - Sept. 2, 1982. NUREG/CP0027.

(6) ICRP Publication 37. Oxford, Pergamon Press, Annals of the ICRP v. 10: N° 2/3, 1983.

(7) Beninson, D. and Lindell, B. IAEA-CN-39/4. STI/PUB/566.

(8) United Nations Scientific Committee on the Effects of Atomic Radiation. 1982 Report to the General Assembly, with annexes. New York, United Nations, 1982. 773 p.

RECOMMANDATIONS SUR LES PRINCIPES GENERAUX DE PROTECTION CONTRE LES RAYONNEMENTS NON-IONISANTS

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I - INTRODUCTION

Pour répondre à l'intérêt croissant manifesté par ses membres à propos des problèmes de protection posés par les rayonnements non-ionisants, l'Association Internationale de Radioprotection (IRPA) a créé en avril 1977 le Comité international sur les rayonnements non-ionisants (INIRC) en lui confiant les missions suivantes :

- faire le point des connaissances sur les effets biologiques des rayonnements non-ionisants (RNI)
- élaborer des documents de base et des recommandations qui pourraient être adoptées au niveau international,
- collaborer avec les autres organisations internationales intéressées à la promotion de la protection contre les RNI.

C'est dans ce cadre que l'INIRC a préparé des recommandations sur les limites d'exposition aux principaux types de rayonnements non-ionisants, qui sont présentées plus en détail dans une autre session de ce Congrès.

Les limites d'exposition proposées par le Comité s'appuient sur l'analyse de l'ensemble des données scientifiques actuellement disponibles en ce qui concerne l'interaction de ces différentes formes d'énergie et des organismes vivants. La collecte et le dépouillement de ces données ont été effectués, dans le cadre d'un travail conjoint de l'Organisation Mondiale de la Santé (OMS) et de l'IRPA/INIRC, grâce à la subvention allouée par le Programme des Nations-Unies pour l'Environnement (UNEP) pour la préparation de documents sur les critères d'hygiène à prendre en considération pour les différentes nuisances rencontrées dans l'environnement. Quatre rapports traitant respectivement des critères d'hygiène relatifs au rayonnement ultraviolet, aux micro-ondes et aux radiofréquences, aux ultrasons, aux rayonnements optiques et aux lasers ont ainsi été établis. Un cinquième rapport sur les rayonnements de fréquence extrêmement faible (ELF) est en cours d'achèvement. De nombreux spécialistes appartenant aux pays et aux institutions les plus divers ont contribué à ce travail par leurs commentaires avant la mise au point finale du texte par un groupe pluridisciplinaire d'experts représentant les différentes tendances scientifiques.

Après le rappel de quelques généralités, nous analyserons brièvement les recommandations proposées par l'IRPA/INIRC, en particulier en ce qui concerne les limites d'exposition aux champs électromagnétiques, et considérerons les lignes directrices d'une politique générale de protection contre les rayonnements non-ionisants.

II - GENERALITES

D'une manière générale, le terme de rayonnements non-ionisants se réfère à tous les types de rayonnements qui, lors de leur interaction avec la matière, ne peuvent céder une énergie suffisante pour produire une ionisation. Ils comprennent tous les rayonnements électromagnétiques ayant une longueur d'onde dans le vide égale ou supérieure à 10^{-7} m. Ces rayonnements peuvent être émis de façon continue ou de façon intermittente et les modulations affectant la fréquence, l'amplitude ou l'impulsion peuvent modifier certaines de leurs propriétés. Les rayonnements optiques (UV, visible,IR)peuvent se présenter sous la forme de faisceaux cohérents et monochromatiques (lasers) ; ils sont alors porteurs d'une énergie dont la concentration est considérable.

A des fins de protection le domaine des RNI est généralement étendu aux champs électrostatiques et magnétostatiques. D'autre part, l'INIRC a inclus dans son champ d'action les ultrasons dont les problèmes de protection très similaires à ceux des rayonnements électromagnétiques n'ont été pris en charge, jusqu'à présent, par aucun organisme international.

Rappelons que dans le domaine des rayonnements non-ionisants comme dans celui des rayonnements ionisants, il importe de faire une distinction entre deux notions que très souvent on confond et que l'on peut désigner par les mots : *effet et dommage*. A partir du moment où il y a interaction entre les rayonnements et la matière vivante, il y a toujours un effet. Cet *effet* peut consister en phénomènes physiques ou physico-chimiques sans conséquences biologiques: phénomènes transitoires suivis d'un retour presque immédiat à l'état initial. Il peut aussi entraîner des processus biologiques plus ou moins réversibles dont les conséquences peuvent être favorables, neutres ou nocives pour la santé. Le *dommage* désignera un effet nocif comportant une altération décelable de la santé pour l'individu exposé lui-même ou pour ses descendants.

La nature des effets produits par les différents types de RNI et leurs conséquences biologiques éventuelles sont extrêmement variables et dépendent d'un grand nombre de facteurs que l'on peut sommairement regrouper en trois catégories :

- *Les paramètres liés aux caractéristiques de la source*, tels que : les dimensions de la source (ponctuelle ou étendue) ; la fréquence du rayonnement émis (ou encore sa longueur d'onde ou l'énergie des photons) qui conditionnera en particulier la profondeur de pénétration du rayonnement dans les tissus et peut, dans certains cas, entraîner des phénomènes spéciaux (comme la résonance); l'intensité ou la densité de puissance du champ ou du faisceau ; le mode d'émission du rayonnement (cohérent ou non-cohérent, continu ou pulsé) et, le cas échéant, la durée des impulsions et de l'intervalle qui les sépare ;

- *Les paramètres liés aux modalités d'exposition* : durée et répartition de l'exposition dans le temps (exposition continue ou intermittente); distance de la source (sauf dans le cas des lasers émis par une source considérée comme ponctuelle); orientation du sujet exposé

dans le champ de rayonnement, distribution spatiale de l'exposition (exposition totale ou partielle de l'organisme) ;

- *Les paramètres liés aux caractéristiques biologiques* : en particulier la nature, la constitution cellulaire et même moléculaire des tissus exposés ; leurs propriétés électriques ; certaines caractéristiques physiologiques comme l'irrigation sanguine lorsqu'il s'agit d'effets thermiques ; les dimensions de l'organe ou du sujet ; l'importance fonctionnelle des tissus ou organes atteints,etc...

C'est grâce aux progrès faits dans la connaissance du rôle joué par ces paramètres qu'une certaine évolution a eu lieu au cours des dernières années dans l'établissement des normes de sécurité relatives, en particulier, aux champs électromagnétiques de 10 MHz à 300 GHz.

L'objectif essentiel des travaux de l'INIRC étant de définir des normes relatives à l'exposition des tissus ou de l'organisme entier aux rayonnements non-ionisants afin de prévenir les dommages pour la santé, on peut qualifier ces normes de *"sanitaires"* par opposition aux normes *"techniques"* qui sont des spécifications imposées aux caractéristiques des appareils afin d'en garantir la sécurité, sous tous ses aspects, dès le stade de la construction. Les prescriptions techniques relatives à l'émission ou à la fuite des rayonnements sont dérivées des normes sanitaires. L'harmonisation des normes techniques au niveau international relève de la Commission Electrotechnique Internationale.

Les normes sanitaires, par contre, s'appliquent soit à l'échange d'énergie entre rayonnement et tissus de l'organisme (ce sont alors des limites d'exposition fondamentales), soit plus fréquemment, dans le cas des rayonnements non-ionisants, aux paramètres caractéristiques du champ de rayonnement au point de l'espace où l'organisme peut être exposé et elles constituent alors des limites opérationnelles.

III - GRANDEURS ET UNITES UTILISEES EN PROTECTION CONTRE LES RNI

Sans entrer dans le détail des grandeurs et unités utilisées pour les différents types de RNI, qui font l'objet d'une présentation dans une autre session de ce Congrès et d'un rapport en cours de préparation par le Comité, je voudrais simplement donner ici quelques idées générales à ce sujet.

La première remarque est qu'on peut regretter qu'il n'y ait pas de grandeur unique, analogue à la dose absorbée, qui permette d'exprimer quantitativement l'échange d'énergie entre rayonnement et matière pour l'ensemble du spectre des rayonnements électromagnétiques non-ionisants et, encore moins, une grandeur qui, comme l'équivalent de dose, permette d'apprécier les risques d'une exposition compte tenu des caractéristiques du rayonnement en cause. Le premier pas en ce sens a été fait, il y a à peine quelques années, dans le domaine des radiofréquences par introduction de la notion de taux d'absorption spécifique (TAS), grandeur analogue à celle appelée débit de dose absorbée pour les rayonnements ionisants. Cette grandeur rend compte de l'énergie transférée aux tissus par unité de temps et de masse et s'exprime en watts par kilogramme. Elle peut être évaluée par calcul et, dans certains cas, mesurée, mais la détermination

exacte de sa distribution dans l'organisme humain irradié pose encore de nombreux problèmes.

Dans tous les autres cas, les normes sanitaires s'appliquent à des paramètres qui caractérisent le champ en un lieu de l'espace sans récepteur où le sujet peut ensuite être exposé. Les grandeurs utilisées, la terminologie qui les désigne et les unités dans lesquelles elles sont exprimées, varient selon le type de rayonnement considéré. C'est ainsi que les principales grandeurs rencontrées en protection contre les rayonnements non-ionisants sont :

-*La puissance traversant l'unité de surface* (en l'absence du sujet). Elle est désignée par de nombreux termes et il est regrettable que bien des obstacles s'opposent encore à l'adoption d'une terminologie unique. On parle de : densité de puissance, débit de fluence énergétique, irradiance ou éclairement énergétique pour les rayonnements optiques, intensité acoustique pour les ultrasons. L'unité utilisée dans le Système international est le watt par mètre carré ($W.m^{-2}$) ou, dans de nombreux cas, le $mW.cm^{-2}$ et le $\mu W.cm^{-2}$.

-*L'énergie traversant l'unité de surface* (en l'absence du sujet), appelée exposition radiante, qui est égale au produit de la densité de puissance moyenne par le temps d'exposition. Elle est utilisée plus particulièrement pour l'exposition de la peau et des yeux aux rayonnements optiques et s'exprime en joules par mètre carré ($J.m^{-2}$).

-*L'intensité du champ électrique* (E) et *du champ magnétique* (H) qui, d'une façon générale, caractérisent le rayonnement électromagnétique. E et H s'expriment respectivement en volts par mètre et ampères par mètre.

A une distance suffisamment éloignée de la source, de l'ordre de quelques longueurs d'onde, les deux champs sont en phase et la densité de puissance est égale au produit de E et de H. Compte-tenu des relations existant alors entre ces trois grandeurs, la limitation de l'une d'entre elles entraînera automatiquement celle des deux autres.

En champ proche, cette relation n'est plus valable et les limites d'exposition devront être appliquées directement aux intensités du champ électrique et du champ magnétique. C'est pourquoi, aux fréquences inférieures à 10 MHz, où les longueurs d'onde sont très grandes comparées aux dimensions du corps humain, les limites seront toujours exprimées en volts par mètre ($V.m^{-1}$) pour le champ électrique et en ampères par mètre ($A.m^{-1}$) pour le champ magnétique.

-*Pour les ultrasons*, les limites s'appliquent au *niveau de pression acoustique* (NPA), grandeur sans dimension qui est une fonction du rapport entre la pression acoustique étudiée et une pression de référence, $NPA = 20 \log_{10} (p/p_r)$. La pression de référence correspond au plus faible niveau de son audible perçu par l'oreille humaine et sa valeur numérique est normalement de 20 micropascals.

Puisque l'intensité acoustique est proportionnelle au carré de la pression acoustique, le niveau de pression acoustique peut également être défini par rapport à l'intensité acoustique, $NPA = 10 \log_{10} (I/Ir)$.

Le niveau de pression acoustique se mesure en décibels, unité exprimant le rapport entre deux grandeurs semblables.

La diversité des grandeurs utilisées pour évaluer et limiter l'exposition à ces différents rayonnements provient certes de leur interaction quelque peu différente avec les systèmes biologiques selon la longueur d'onde considérée, mais également pour une grande part des traditions auxquelles radioélectriciens, opticiens et acousticiens sont attachés dans leurs domaines respectifs. C'est pour ces raisons, parmi d'autres, qu'à la suite d'une analyse détaillée, le Comité a conclu qu'il paraissait difficile d'aboutir dans un proche avenir à une harmonisation générale des concepts, grandeurs et unités utilisés en dosimétrie et en protection

contre les RNI.

IV - NIVEAUX ET MODALITES D'EXPOSITION DE L'HOMME

A l'exception de l'exposition au rayonnement ultraviolet d'origine solaire, il ne semble pas que l'homme ait été exposé dans le passé à des niveaux notables de RNI.

En ce qui concerne les ondes radioélectriques, pour lesquelles on a des données chiffrées suffisamment abondantes, on sait qu'il a toujours existé un léger bruit de fond naturel. L'électricité atmosphérique, les émissions d'ondes électromagnétiques par le soleil et les étoiles créent des champs faibles présentant une densité de puissance moyenne de l'ordre de 10^{-8} mW.cm^{-2}. Les sources artificielles qui se sont multipliées depuis quelques décades (radiodiffusion, télévision, radars, utilisations industrielles, médicales, domestiques...) entraînent non seulement une exposition professionnelle importante, mais encore une exposition générale du public qui se situe à plusieurs ordres de grandeur au-dessus du bruit de fond naturel. On estime que, dans certaines zones fortement industrialisées, la densité de puissance ambiante est de l'ordre de 10^{-3} mW.cm^{-2}, soit un facteur 10^5 par rapport au bruit de fond naturel. Bien qu'une telle élévation au-dessus du niveau naturel constitue un élément nouveau dans l'environnement de l'être humain, il est universellement accepté qu'elle n'entraîne pas de risques pour la santé de l'homme.

D'autre part, le rayonnement ultraviolet, les lasers et les ultrasons font l'objet d'applications de plus en plus nombreuses dans la recherche, l'industrie, la médecine, parfois même dans les pratiques de la vie courante. Il en résulte une multiplication considérable des sources et, par suite, une obligation plus pressante de promouvoir une politique générale de protection pour l'exposition des travailleurs et du public.

V - RECOMMANDATIONS DE L'IRPA/INIRC SUR LES LIMITES D'EXPOSITION

C'est pour répondre à ce souci que le Comité international sur les Rayonnements non-ionisants de l'IRPA vient de publier des recommandations sur les limites à respecter lors de l'exposition à des champs électromagnétiques de 100 kHz à 300 GHz et aux ultrasons se propageant dans l'air. Des recommandations analogues viennent d'être approuvées en ce qui concerne le rayonnement ultraviolet et les lasers et devraient être publiées avant la fin de cette année.

Il faut souligner qu'il s'agit là de la première tentative faite par un organisme international pour promouvoir une harmonisation des normes sanitaires utilisées dans les divers pays du monde. Ces recommandations ont été élaborées avec l'aide des experts les plus connus dans chaque domaine concerné et soumises également pour commentaires à toutes les Sociétés affiliées à l'IRPA.

Je n'insisterai pas ici sur les normes relatives aux ultrasons et aux rayonnements optiques qui sont exposées par ailleurs et pour lesquelles il n'existe pas de divergences fondamentales. D'une manière générale, les limites d'exposition recommandées ne s'appliquent pas lorsqu'il s'agit de l'exposition des patients à des fins médicales (diagnostiques ou thérapeutiques). Aussi, les recommandations faites pour les ultrasons ne s'appliquent-elles qu'au cas de leur propagation dans l'air qui constitue le principal risque d'exposition résultant des autres utilisations. Dans le domaine des rayonnements lasers, il convient, en outre, de mentionner le travail important fait par la Commission Electrotechnique Internationale (CEI) au cours des dernières années pour aboutir à un accord international sur les normes techniques comportant, notamment, une classification des appareils à laser en fonction de leurs risques et des limites d'émission pour chacune de ces classes. Cet accord est maintenant réalisé et la norme de la CEI devrait être publiée prochainement. Malgré

cela et en dépit d'une demande accrue de la part des cercles intéressés, peu de pays ont jusqu'à présent prévu une réglementation ou un code de pratique pour la protection des travailleurs et du public lors de l'utilisation de ces rayonnements. C'est pourquoi, semble-t-il, les projets élaborés par l'INIRC ont été accueillis avec intérêt puisqu'ils ont déjà fait l'objet d'un certain nombre de demandes de renseignements.

Il existe cependant un domaine, à savoir celui des ondes radioélectriques, où l'harmonisation sur le plan international présente une importance toute particulière du fait de leur emploi généralisé dans les télécommunications et où, paradoxalement, les divergences étaient jusqu'à récemment les plus profondes. Toutefois au cours des dernières années, faisant suite à un effort accru dans la recherche sur les effets biologiques de ces rayonnements, des tendances nouvelles sont apparues laissant percevoir quelques possibilités de rapprochement. Les propositions faites par l'INIRC dans ce but, qui sont brièvement résumées ci-dessous, s'appuient sur les données scientifiques les plus récentes.

Rappelons que, pendant longtemps, l'intérêt en ce domaine était concentré sur les micro-ondes, c'est-à-dire la partie du spectre des radiofréquences comprise entre 300 MHz et 300 GHz. La limite de 10 $mW.cm^{-2}$ utilisée aux Etats-Unis et par un certain nombre de pays européens jusqu'à ces dernières années était fondée sur la charge calorifique additionnelle considérée comme tolérable par l'organisme humain. En URSS, par contre, la limite mille fois plus faible prescrite pour l'exposition professionnelle était déduite de certaines modifications fonctionnelles des systèmes nerveux et cardiovasculaire observées chez l'animal et chez l'homme à des densités de puissance de l'ordre de 0,1 $mW.cm^{-2}$. Il semble toutefois que les incertitudes sur les expositions subies et les mesures effectuées à cette époque ne permettent pas d'en tirer des conclusions précises sur la relation dose-effet. Par la suite, tenant compte de données plus récentes, un certain nombre de pays avaient adopté des limites intermédiaires entre ces deux extrêmes.

Cependant, le développement des applications, de nouvelles données biologiques et surtout une meilleure connaissance des processus d'interaction biophysique grâce à l'emploi de techniques modernes, ont conduit, ces derniers temps, non seulement à étendre les limites d'exposition vers des fréquences plus basses, mais à en réviser complètement les fondements scientifiques. C'est sur ces nouvelles bases que s'appuient les limites qui viennent d'être proposées par l'IRPA/INIRC, ainsi que par deux organismes américains très influents, l'American National Standards Institute (ANSI) et l'American Conference of Governmental Industrial Hygienists (ACGIH). Toutefois, bien qu'admettant la même limite fondamentale de 0,4 $W.kg^{-1}$ pour le taux d'absorption spécifique moyen (TAS) de l'énergie électromagnétique dans l'ensemble de l'organisme, les limites opérationnelles recommandées par les trois organismes présentent des différences en raison de certaines divergences d'interprétation et de l'adoption de facteurs de sécurité différents.

L'effet biologique le plus sensible qui a servi de base à l'établissement de la limite fondamentale mentionnée ci-dessus est une modification du comportement chez les animaux soumis à des irradiations de durée inférieure à 1 heure (par exemple convulsions, arrêt ou diminution du travail, endurance réduite, réactions d'aversion). Pour tenir compte de certaines incertitudes et, en particulier, du risque éventuel que pourrait impliquer une exposition prolongée, on a appliqué un facteur de sécurité de 10 à la valeur de 4 $W.kg^{-1}$ au-dessous de laquelle aucun dommage sur la santé des animaux n'avait été constaté.

Ce n'est que grâce à des méthodes mises au point très récemment que l'on peut évaluer l'énergie électromagnétique absorbée dans des conditions d'exposition proches de la réalité. Le TAS, qui dépend d'un grand nombre de facteurs, est en général obtenu par calcul ou, dans quelques cas, déterminé expérimentalement. Pour la

pratique, il a donc fallu en déduire des limites plus accessibles à la mesure qui sont exprimées en termes d'intensités du champ électrique ($V.m^{-1}$) et du champ magnétique ($A.m^{-1}$) ou de densité de puissance ($W.m^{-2}$). Celles-ci varient avec la fréquence pour tenir compte des variations de l'énergie absorbée en fonction du domaine de fréquence considéré et, en particulier, du phénomène de résonance entre 30 MHz et environ 400 MHz qui peut amplifier considérablement l'énergie absorbée. Cependant, compte-tenu des résultats expérimentaux limités dont on dispose pour le moment, la détermination des limites pratiques implique certaines extrapolations et comporte de ce fait des incertitudes. De plus, le taux d'absorption spécifique sur lequel elles sont fondées ne rend compte que des effets dits thermiques, alors que certaines observations n'excluent pas actuellement la possibilité de production de certains effets spécifiques liés à d'autres mécanismes. Mais les données dont on dispose à ce sujet sont encore insuffisantes pour déterminer si ces effets présentent un risque pour la santé.

Aux fréquences inférieures à 10 MHz, compte-tenu des longueurs d'onde mises en jeu, les conditions d'exposition sont telles que les limites doivent être fixées en termes d'intensité du champ électrique et du champ magnétique et l'exposition devra respecter la plus faible de ces limites. Les densités de puissance correspondantes ne sont alors données qu'à titre comparatif. Dans cette région où l'énergie absorbée par l'organisme est très faible, les limites proposées sont fondées sur des considérations différentes et l'INIRC, contrairement à certains autres organismes, a tenu compte, en particulier, des chocs électriques et brûlures que l'individu peut ressentir par contact avec des objets métalliques non mis à la terre lorsque le champ électrique dépasse 200 V/m.

En ce qui concerne l'exposition des *personnes du public*, l'INIRC a estimé qu'il fallait incorporer un *facteur de sécurité supplémentaire de* 5 aux limites fixées pour les travailleurs, afin de tenir compte de la présence possible d'individus plus sensibles, de la difficulté de détecter d'éventuels effets sur la santé et du fait que le public peut être exposé 24 heures par jour pendant des périodes prolongées.

Toutefois certains appareils de très faible puissance assez couramment utilisés, tels que tous émetteurs-récepteurs dont la puissance est inférieure à 7 watts, sont exclus de l'application de ces limites, car ils n'engendrent que des champs extrêmement localisés.

VI - PRINCIPES GENERAUX DE PROTECTION CONTRE LES RNI

L'INIRC a, d'autre part, établi l'ébauche d'une doctrine générale de protection contre les RNI qui repose sur les principes fondamentaux suivants :

1- *L'établissement de normes sanitaires*, comportant notamment des limites d'exposition, en vue d'assurer une bonne protection des travailleurs professionnellement exposés et des personnes du public contre les dommages que pourraient entraîner les RNI.

2- *L'élaboration de normes techniques* s'appliquant à la conception, à la construction et au mode d'emploi des appareils ou dispositifs émetteurs de RNI afin de garantir, dans toute la mesure du possible, le respect des normes sanitaires au niveau de la source.

Les pouvoirs publics devraient veiller, en particulier, à ce que les prototypes de chaque appareil ou dispositif obtiennent l'agrément de services spécialisés avant toute production en série ou commercialisation.

3- *La définition de mesures de protection opérationnelle* : Il ne saurait être question de les énumérer toutes ici car elles sont très nombreuses et varient considérablement selon les rayonnements en cause, les types d'émetteurs et les circonstances. Citons simplement quelques mesures de caractère général fréquemment mises en oeuvre :

3/1- *sélection appropriée des lieux d'implantation* (par ex. pour les antennes émettrices des installations radio, radar, de forte puissance, etc...)
3/2- *délimitation de zones à accès réglementé ou de zones interdites*
3/3- *aménagement des locaux* : mise en place d'écrans adaptés, dispositifs de sécurité automatiques...
3/4- *signalisation* des zones et signalisation de l'état de marche des appareils...
3/5- élaboration de *consignes de sécurité* pour l'emploi des appareils aux différents postes de travail, y compris pour la protection contre les risques associés (électriques, chimiques...)
3/6- emploi, dans certaines circonstances *de moyens de protection individuels* (vêtements spéciaux, lunettes, etc...)
3/7- *formation, éducation et entraînement du personnel*
3/8- *surveillance de l'exposition* aux RNI au moyen d'un matériel de mesure normalisé et étalonné selon une procédure également normalisée
3/9- *surveillance médicale* adaptée à la nature et à l'importance des risques auxquels les travailleurs sont exposés.

VII - CONCLUSION

Les normes sanitaires, les normes techniques ainsi que les mesures générales de prévention et de surveillance à mettre en oeuvre pour la protection des travailleurs et de la population contre les effets nocifs des différents types de RNI devraient faire l'objet d'une réglementation. Les recommandations de l'IRPA/INIRC ont été rédigées pour servir de guide aux pouvoirs publics des différents pays auxquels incombe la responsabilité de l'élaboration et de l'application d'une telle réglementation dans le cadre de la législation nationale.

Les incertitudes qui subsistent sur les mécanismes d'interaction de ces rayonnements avec l'organisme humain et l'absence d'une relation dose-effet nettement établie peuvent certes entraîner encore quelques divergences d'opinion sur les limites d'exposition à appliquer pour la protection des travailleurs et du public. Mais il faut espérer que les efforts faits par le Comité international de l'IRPA pour élaborer des recommandations fondées sur les données scientifiques les plus récentes et sur le consensus le plus large permettront d'aboutir à une meilleure harmonisation des réglementations adoptées à l'avenir par les divers pays.

R E F E R E N C E S

UNEP/WHO/IRPA : Environmental Health Criteria 14, Ultraviolet radiation - WHO, Geneva 1979
UNEP/WHO/IRPA : Environmental Health Criteria 16, Radiofrequency and Microwaves - WHO, Geneva 1981
UNEP/WHO/IRPA : Environmental Health Criteria 22, Ultrasound, WHO, Geneva 1982
UNEP/WHO/IRPA : Environmental Health Criteria 23, Lasers and Optical Radiation, WHO, Geneva 1982
IRPA/INIRC : Interim Guidelines on limits of exposure to electromagnetic fields in the frequency range from 100 kHz to 300 GHz, Health Physics, April 1984
IRPA/INIRC : Interim Guidelines on limits of human exposure to airborne ultrasound, Health Physics, April 1984
IRPA/INIRC : Guidelines on limits of exposure to ultraviolet radiation (incoherent optical radiation) - In preparation
IRPA/INIRC : Guidelines on limits of exposure to laser radiation of wavelengths between 180 nm and 1 mm - In preparation

GESETZGEBUNGS- UND RECHTSPROBLEME IM STRAHLENSCHUTZ

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I. Strahlenschutzrecht und Atomenergierecht

Es ist nicht selbstverständlich, daß die Tätigkeiten der Erzeugung ionisierender Strahlen durch technische Apparaturen und der Umgang mit radioaktiven Stoffen zum Gegenstand von rechtlich verbindlichen Vorschriften gemacht werden, die von staatlichen oder internationalen Gesetzgebungsorganen erlassen werden. Nach Entdeckung der X-Strahlen durch Röntgen und der radioaktiven Stoffe durch Becquerel und das Ehepaar Curie und nach ihrer Verwendung vor allem in der Medizin dauerte es mehrere Jahrzehnte, bis der Staat und bis internationale Organisationen diesen neuen durch technische Erfindungen und Entdeckungen erschlossenen Tätigkeits- und Anwendungsbereich rechtlichen Regelungen unterworfen haben mit dem Zweck, die dabei entstehenden Gesundheitsgefahren für die Berufstätigen, für Patienten in der Medizin und für die Bevölkerung nach Möglichkeit einzuschränken. Bis dahin wurde der Schutz vor den Strahlen denjenigen überlassen [1], die vor allem mit ionisierenden Strahlen und radioaktiven Stoffen umgingen: den Ärzten, ihren Standesorganisationen und radiologischen Gesellschaften.

Erst als die im Jahre 1928 auf dem internationalen Kongreß für Radiologie in Stockholm von dem Internationalen Röntgenstrahlen- und Radiumschutzkomitee verabschiedeten Empfehlungen für Röntgenstrahlen- und Radiumschutz vom 27. Juli 1928 bekannt wurden [2], haben verschiedene Staaten in den darauf folgenden Jahren in der Rechtsform eines Gesetzes oder einer Verordnung Spezialvorschriften über Röntgengeräte und radioaktive Stoffe erlassen [3]. Zur Ausbildung eines eigenen Rechtsgebietes, des Strahlenschutzrechts kam es jedoch erst nach dem zweiten Weltkrieg, als die Strahlenanwendung immer größeren Umfang annahm und die Nutzung der Kernenergie mit Hilfe technischer Großanlagen den verstärkten Strahlenschutz erforderlich machten.

Das Strahlenschutzrecht wird durch die Gesamtheit der rechtlichen Spezialnormen gebildet, die die Wirkungen von radioaktiven Stoffen und von ionisierenden und nichtionisierenden Strahlen innerhalb einer Rechtsgemeinschaft rechtlich regeln und den Schutz der Einzelmenschen, der Sachgüter und der Umwelt vor den schädlichen Wirkungen dieser Strahlen zum Gegenstand haben [4]. Das Strahlenschutzrecht wird im allgemeinen dem Atomenergierecht als dessen wesentlicher Teil zugerechnet [5], doch decken sich beide Rechtsgebiete nicht vollständig. Es herrscht Einvernehmen, daß zum Strahlenschutzrecht auch die Rechtsregeln hinsichtlich der Erzeugung und Anwendung von Röntgenstrahlen gehören, obwohl diese mit der Atomenergienutzung unmittelbar nicht im Zusammenhang stehen. Außerdem sollte auch der Bereich der rechtlichen Regelungen zum Schutze vor Schäden durch nichtionisierende Strahlen dem Strahlenschutzrecht zugerechnet werden. Man muß daher von einer Eigenständigkeit des Strahlenschutzrechts neben dem Atomenergierecht ausgehen. Abzugrenzen ist das Strahlenschutzrecht auch von dem Reaktorsicherheitsrecht (oder dem Sicherheitsrecht für kerntechnische Anlagen), das durch Rechtsvorschriften und technische Regeln eine gewisse Eigenständigkeit im Rahmen des Atomenergierechts gewonnen hat, obwohl eine vollkommen schlüssige Abgrenzung beider

Rechtsbereiche bisher noch nicht gefunden wurde. Zum Strahlenschutzrecht im engeren Sinne gehören ferner nicht die Rechtsnormen, die zum Schutze gegen Eingriffe von außen (physical security) erlassen werden, sowie schließlich auch nicht der Bereich der Sicherheitskontrolle (safeguards).

II. Internationale Rechtsgrundlagen [6)]

Im Gegensatz zu anderen Rechtsgebieten (etwa des Zivilrechts, des gerichtlichen Verfahrensrechts, des Haftungsrechts) hat das Strahlenschutzrecht seit Beginn der technischen Entwicklung dieses Anwendungsbereichs den großen Vorteil, daß die innerstaatlichen Rechtsvorschriften in sehr weitgehendem Maße auf einer internationalen Harmonisierung der Regeln aufbauen und sich an ihnen orientieren können. Dabei ist es für den ganzen Bereich des internationalen Rechts und der Rechtsvereinheitlichung eine singuläre Erscheinung, daß dieses Rechtsgebiet von den wissenschaftlichen Erkenntnissen und den Empfehlungen einer privaten, nichtstaatlichen internationalen Organisation beeinflußt, je geradezu beherrscht wird: Der Internationalen Strahlenschutzkommission (ICRP), derselben Kommission, die 1928 bereits die ersten Strahlenschutzempfehlungen herausgegeben hat.

In den letzten Jahrzehnten hat sich nun ein außerordentlich dynamischer Prozeß der internationalen und innerstaatlichen Gesetzgebung auf dem Gebiete des Strahlenschutzes ergeben und eingespielt. Die auf Grund von Forschung, Statistik und Erfahrung, bisweilen auch durch eingetretene Schäden gewonnenen wissenschaftlichen Erkenntnisse werden zu internationalen Empfehlungen geformt, die sodann von den verschiedenen, auf Grund ihrer Satzung für den Strahlenschutz zuständigen internationalen staatlichen Organen übernommen werden. Nach ihrem Satzungsrecht bestimmt sich auch, ob die Strahlenschutzregeln für die Mitgliedstaaten völkerrechtlich verbindlich sind, ob die Staaten also verpflichtet sind, sie einzuhalten, und ihre Staatsbürger zur Einhaltung durch verbindliche Rechtsvorschriften verpflichten müssen. Sofern eine Verbindlichkeit satzungsrechtlich nicht möglich ist oder nicht opportun erscheint, haben die Strahlenschutzregeln nur die rechtliche Qualität von Empfehlungen (Recommendations).

Folgende internationale staatliche Organisationen (IGOs) haben sich mit dem materiellen Strahlenschutz befasst:

- Die Vereinten Nationen durch den Strahlenschutzausschuß der Generalversammlung, das United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). Dieser Ausschuß ist am 3. Dezember 1955 durch die Resolution Nr. 913 der Generalversammlung errichtet worden und besteht seit 1974 auf Grund der Resolution Nr. 3154 vom 14. Dezember 1973 aus 20 Mitgliedern. UNSCEAR hat nicht die Aufgabe, internationale Strahlenschutzübereinkommen oder Empfehlungen auszuarbeiten oder zu erlassen, sondern dieser Ausschuß sammelt Informationen über die möglichen schädlichen Wirkungen ionisierender Strahlen und analysiert die Strahlenwirkungen auf den Menschen und auf die Umwelt. Die Ergebnisse werden in Berichten an die Generalversammlung der Vereinten Nationen zusammengefaßt, deren Auswertung für die internationale und die innerstaatliche Strahlenschutzgesetzgebung von großem Nutzen ist.
- Für den Bereich des Arbeitsschutzes hat die Internationale Arbeitsorganisation (ILO) am 22. Juni 1960 das Übereinkommen Nr. 115 über den Schutz der Arbeitnehmer vor ionisierender Strahlung sowie die

Empfehlung Nr. 114 betreffend den Schutz der Arbeitnehmer vor ionisierender Strahlung verabschiedet. Das Übereinkommen ist zur Zeit für 38 Staaten völkerrechtlich verbindlich.

- Die Weltgesundheits-Organisation (WHO) könnte nach ihrer Satzung für den Bereich des Strahlenschutzes verbindliche Übereinkommen durch ihre Vollversammlung verabschieden; sie hat von dieser Möglichkeit jedoch bisher noch keinen Gebrauch gemacht. Wiederholt hat die WHO jedoch Empfehlungen für den Strahlenschutz, insbesondere in der Medizin, herausgegeben, zuletzt im Jahre 1982 zwei Guides zur Qualitätssicherung auf den Gebieten der radiologischen Diagnostik und der Nuklearmedizin.
- Die Internationale Atomenergie-Organisation (IAEA) gibt für den Bereich des Strahlenschutzes die Safety-Series heraus, in der bis Ende 1983 insgesamt 61 Bände erschienen sind. Diese Veröffentlichungen beruhen auf Artikel III A. 6 der Satzung der IAEA, wonach Sicherheitskriterien dieser Organisation grundsätzlich nur auf ihre eigenen Tätigkeiten Anwendung finden und für die Mitgliedstaaten nur dann verbindlich sind, wenn sie auf Grund von bilateralen oder multilateralen Abkommen mit der IAEA für anwendbar erklärt werden. Davon abgesehen haben die Strahlenschutzregeln der Safety-Series den Rechtscharakter von internationalen Empfehlungen ohne Bindungswirkung.
- Nach Artikel 11 der Satzung der Kernenergieagentur (NEA) der Organisation für Wirtschaftliche Zusammenarbeit und Entwicklung (OECD) vom 17. Dezember 1957 fördert die Agentur die Ausarbeitung und gegenseitige Abstimmung von Rechtsvorschriften betreffend die Kernenergie in den Teilnehmerstaaten, insbesondere im Hinblick auf den öffentlichen Gesundheitsschutz und die Verhütung von Unfällen in der Kernindustrie. Zu diesem Zweck hat der Direktionsausschuß gemeinsame Regeln als Grundlage für die innerstaatlichen Rechts- und Verwaltungsvorschriften ausgearbeitet. Die OECD-Strahlenschutzgrundnormen von 1959, die 1962 und 1968 neu gefaßt wurden, werden nunmehr überlagert durch die "Basic Safety Standards for Radiation Protection", die 1982 als Nr. 9 der IAEA-Safety-Series unter Beteiligung der ILO, der NEA (OECD) und der WHO in einer Neufassung und weitgehend auf der Grundlage der allgemeinen ICRP-Empfehlungen Nr. 26 aus dem Jahre 1977 veröffentlicht wurden.
- Der Vertrag zur Gründung der Europäischen Atomgemeinschaft (Euratom) vom 25. März 1957 enthält selbst keine materiellen Strahlenschutzvorschriften. Nach Artikel 30 werden in der Gemeinschaft Grundnormen für den Gesundheitsschutz der Bevölkerung und der Arbeitskräfte gegen die Gefahren ionisierender Strahlungen festgesetzt. Die Euratom-Strahlenschutzgrundnormen wurden zunächst 1959 erlassen und sodann 1962 und 1966 an die jeweiligen ICRP-Änderungen angepaßt. Eine Neufassung der Grundnormen erschien am 1. Juni 1976; die jetzt geltende Fassung der Euratom-Grundnormen wurde am 15. Juli 1980 erlassen. Nach Art. 33 Abs. 1 des Euratom-Vertrages sind die zehn Mitgliedstaaten der Europäischen Atomgemeinschaft verpflichtet, durch den Erlaß geeigneter Rechts- und Verwaltungsvorschriften die Beachtung der festgesetzten Grundnormen sicherzustellen und die für den Unterricht, die Erziehung und Berufsausbildung erforderlichen Maßnahmen zu treffen. Die Strahlenschutzgrundnormen sind damit hinsichtlich des zu erreichenden Ziels für jeden Mitgliedstaat der Europäischen Gemeinschaften verbindlich. Die Richtlinie überläßt jedoch den innerstaatlichen Stellen die Wahl der Form und der Mittel. Die geltenden Euratom-Grundnormen beruhen auf den Empfehlungen der ICRP-Veröffentlichung Nr. 26. Die erforderlichen gesetzgeberischen Maßnahmen müssen in den Euratom-Mitgliedstaaten spätestens am

3. Juni 1984 getroffen sein.

- Für den Strahlenschutz bei der Beförderung radioaktiver Stoffe sind die von den internationalen Spezialorganisationen für den Transport gefährlicher Güter durch die Eisenbahn, im Straßenverkehr, im Binnenschiffs-, im See- und im Luftverkehr erlassenen internationalen Bestimmungen einschlägig, die auch für die Rechtsvereinheitlichung der für den innerstaatlichen Verkehr geltenden Rechtsvorschriften von großer Bedeutung sind.
- Für den Bereich der Reinhaltung der Flüsse, Binnengewässer und des Meerwassers von schädlichen radioaktiven Stoffen sind die einschlägigen internationalen multilateralen und bilateralen Übereinkommen zu beachten.
- Hinzuweisen ist ferner auf die Beschlüsse der Internationalen Organisation für das gesetzliche Meßwesen (Übereinkommen vom 12. Oktober 1955), die auch bezüglich der Verwendung der Meßeinheiten in der Radiologie und im Strahlenschutz Anwendung finden, nachdem sie durch die staatlichen Gesetzgebungsorgane in innerstaatliches Recht transformiert worden sind.

Neben diesen genannten auf Völkerrecht beruhenden internationalen Organisationen und Übereinkommen haben für die Rechtsvereinheitlichung und Schaffung eines einheitlichen Strahlenschutzrechts auch die Arbeiten mehrerer internationaler nichtstaatlicher Organisationen (NGOs) eine nicht zu überschätzende Wirkung. Zu nennen sind hier außer der ICRP deren Schwesterorganisation im Rahmen des Internationalen Kongresses für Radiologie, die International Commission on Radiation Units and Measurements (ICRU) sowie die internationalen Normungsorganisationen (International Organization for Standardization - ISO und das International Electrotechnical Committee - IEC). Nicht zuletzt ist auf die International Radiation Protection Association (IRPA) hinzuweisen, die durch ihre Kongresse und den Gedankenaustausch ihrer nationalen Fachverbände und Mitglieder auf die einheitliche Gestaltung und optimale Lösung aktueller Strahlenschutzfragen und die Gesetzgebung auf diesem Gebiet im internationalen und nationalen Rahmen einen erheblichen Einfluß ausübt.

III. Hauptprobleme innerstaatlichen Strahlenschutzrechts

1. Völkerrecht und Landesrecht

Völkerrechtliche Rechtssätze, seien sie in völkerrechtlichen Verträgen oder in den Rechtsakten internationaler staatlicher Organisationen enthalten, bedürfen auf Grund des dualistischen Verhältnisses des Völkerrechts zum innerstaatlichen Recht eines Transformationsaktes oder eines Rechtsanwendungsbefehls durch den innerstaatlichen Gesetzgeber. Ob der Staat zu einer solchen Umsetzung völkerrechtlich verpflichtet ist, richtet sich nach dem Inhalt des jeweiligen völkerrechtlichen Abkommens und der mitgliedschaftlichen Beziehungen des Staates in der betreffenden internationalen Organisation. Diese allgemeinen Rechtsprinzipien gelten auch für völkerrechtliche Normen auf dem Gebiete des Strahlenschutzes.

2. Verfassungsrechtliche Fragen

Das innerstaatliche Verfassungsrecht bestimmt, welche Verfassungsorgane zum Erlaß von Strahlenschutzvorschriften zuständig sind. In föderalen Staaten (zum Beispiel USA, Schweiz, Bundesrepublik Deutschland, Jugoslawien) ist aus der Verfassung abzuleiten, ob und inwieweit der Bund oder die Teilstaaten (States, Länder, Kantone) zur

Gesetzgebung und Verwaltung Befugnisse besitzen. In den USA haben die Einzelstaaten (States) gegenüber dem Bund die Befugnis zum Erlaß von eigenen Strahlenschutzgesetzen. Diese Strahlenschutzgesetze der Einzelstaaten sind allerdings weitgehend einem Modellgesetz nachgestaltet und in weiten Regelungsteilen vereinheitlicht. Für die Tätigkeiten der Nuclear Regulatory Commission der Vereinigten Staaten und für die Vertragspartner der NRC gelten jedoch die Bestimmungen des Parts 20 "Standards for Protection against Radiation" des Title 10 des Code of Federal Regulations. In der Bundesrepublik Deutschland und in der Schweiz liegt die Kompetenz für den Erlaß von Strahlenschutzgesetzen beim Bund.

Die Verfassung regelt auch, in welcher Rechtsform die strahlenschutzrechtlichen Vorschriften zu erlassen sind, ob durch Gesetz, Rechtsverordnung, Ministerialverordnung, allgemeine Verwaltungsvorschriften, Richtlinien, Codes of Practice, Empfehlungen, Runderlasse, Regulatory Guides usw. Überblickt man die innerstaatliche Gesetzgebung auf dem Gebiet des Strahlenschutzes [7], so stellt man fest, daß die Mehrzahl der Staaten die grundlegenden Vorschriften in Gesetzen (Atomgesetzen, Strahlenschutzgesetzen) erlassen haben, die jedoch meistens auch Ermächtigungsvorschriften enthalten, die der Regierung oder einem oder mehreren Ministern die Befugnis übertragen, Ausführungsvorschriften zu dem Gesetz in der Rechtsform einer Verordnung (Strahlenschutzverordnung) zu verkünden. Nicht selten werden zentrale Strahlenschutzverordnungen durch weitere Verordnungen ergänzt, die Teilbereiche regeln. Die Gestaltungs- und Regelungsformen sind außerordentlich vielfältig. In Belgien, Finnland, Jugoslawien, Luxemburg, Österreich und Schweden gibt es Strahlenschutzgesetze, die durch Strahlenschutzverordnungen ergänzt werden. Frankreich besitzt nebeneinander drei Strahlenschutzdekrete von 1966, 1967 und 1975, die sich auf die allgemeinen Grundsätze, auf den Beschäftigtenschutz in Kernanlagen und außerhalb von Kernanlagen (beim Umgang mit radioaktiven Stoffen und beim Betrieb von Strahlengeräten) beziehen. Daneben enthält das Gesundheitsgesetzbuch spezielle Strahlenschutzvorschriften. Diese Dekrete werden durch eine ganze Reihe von Ministerialverordnungen (Arrêtés) ergänzt. In Großbritannien finden sich die einschlägigen Schutzvorschriften in den Radioactive Substances Acts von 1948 und 1960, im Factories Act von 1961, der durch die Strahlenschutzverordnung für offene radioaktive Stoffe von 1968 und für geschlossene radioaktive Stoffe von 1969 ergänzt wird, sowie im Radiological Protection Act von 1970 und im Health and Safety at Work Act von 1974. Einzelfragen sind durch eine große Zahl von Einzelverordnungen (Orders, Regulations, Statutory Instruments) geregelt. In der Bundesrepublik Deutschland, in der Deutschen Demokratischen Republik, in Italien, in den Niederlanden, in Norwegen, in der Schweiz und in Spanien werden die wesentlichen Strahlenschutzfragen im Atomgesetz geregelt, die durch Strahlenschutz- und Röntgenverordnungen ergänzt werden. Dänemark besitzt ein Röntgengesetz von 1930, auf Grund dessen eine ganze Reihe von Röntgenverordnungen für die einzelnen Anwendungsarten erlassen sind, und daneben ein Gesetz über die Verwendung radioaktiver Stoffe von 1953 mit einer Durchführungsverordnung von 1975. Über die Gesetzeslage in den Vereinigten Staaten von Amerika (Federal Law und State Law) ist bereits berichtet worden.

3. Gesetzliche Vorschriften und Normadressaten

Bei der Ausweitung, die die Verwendung radioaktiver Stoffe und die Anwendung ionisierender Strahlen in der Medizin und auch in der

Technik, Forschung, Landwirtschaft und in anderen Bereichen gefunden hat, nicht zuletzt auch durch die Nutzung der Kernenergie in kerntechnischen Großanlagen (einschließlich der Abfallager), ist es für die meisten Staaten eine verfassungsrechtlich gebotene Obliegenheit, einen wirksamen Schutz der Bevölkerung, der Berufstätigen und Umwelt vor Strahlenschädigungen durch den Erlaß von Rechtsvorschriften zu gewährleisten. Rechtsnormen regeln das Zusammenleben der Menschen und Institutionen innerhalb einer Rechtsgemeinschaft, sie sind das Mittel zur Gewährleistung von Leben, Gesundheit, Freiheit und anderen Rechtsgütern. Rechtsnormen sind im Vergleich zu unverbindlichen Empfehlungen für die Mitglieder der Rechtsgemeinschaft verbindlich, und ihre Einhaltung ist durch Strafen, Bußgelder und durch Verwaltungs- und Vollstreckungsmaßnahmen erzwingbar und durchsetzbar.

Verbindliche Rechtsnormen müssen auch im Bereich des Strahlenschutzes von den Betroffenen ein konkretes Tun oder Unterlassen verlangen; sie haben Gebote und Verbote zum Inhalt. Aus der einzelnen Rechtsvorschrift muß möglichst klar und eindeutig hervorgehen, an wen sich der Rechtsbefehl richtet, wer der Adressat der Norm ist. Darüber hinaus muß die Rechtsvorschrift von dem Normadressaten auch ausführbar sein, es darf nicht Unmögliches oder Unzumutbares von ihm verlangt werden (ultra posse nemo obligatur). Bei der Auslegung von Rechtsvorschriften ist im allgemeinen der Grundsatz der Verhältnismäßigkeit zu beachten.

Prüft man die Strahlenschutzvorschriften der verschiedenen Staaten, so ist festzustellen, daß sich die gesetzlichen Verpflichtungen in erster Linie an diejenigen richten, die kerntechnische Anlagen oder radiologische Geräte errichten und betreiben oder mit radioaktiven Stoffen umgehen, sie befördern oder ein- oder ausführen. Da der Staat im Atom- und Strahlenschutzrecht weitgehend das gesetzgeberische Instrument des Verbots mit Genehmigungsvorbehalt verwendet, sind die bezeichneten, mit Strahlengefahren verbundenen Tätigkeiten grundsätzlich verboten und nur dann und nur insoweit zulässig, als sie von der zuständigen Behörde ausdrücklich genehmigt worden sind. Tätigkeiten mit einem verhältnismäßig geringen Gefahrenpotential werden häufig nur einer Anzeigepflicht unterworfen oder von einer Genehmigungs- oder Anzeigepflicht vollständig befreit (Freigrenzenregelungen)[8]. In vielen Rechtsordnungen gibt es auch die gesetzliche Möglichkeit, standardisierte und in Serie gefertigte Gerätetypen auf Grund eines bestimmten Gerätemusters zuzulassen (Bauartzulassung) mit der Rechtsfolge, daß die nach einem solchen Bauartmuster gefertigten Strahlengeräte sodann einem vereinfachten Verwaltungsverfahren unterworfen werden können.

Soweit in den Rechtsvorschriften Einzelregelungen nicht getroffen werden (häufig bewußt mit dem Ziel, die Rechtsanwendung flexibel und entsprechend der technischen Fortentwicklung anpassungsfähig zu gestalten) sind Ermessensbereiche und Regelungslücken ausfüllungsbedürftig. Der staatliche Gesetzgeber verweist dabei regelmäßig auf die anerkannten Regeln der Wissenschaft und der Technik nach ihrem jeweiligen aktuellen Stand. Die Gesetzesanwender, die Rechtsunterworfenen, die staatlichen Genehmigungs- und Überwachungsbehörden und schließlich im Konfliktsfalle die Gerichte sind dabei gehalten, den jeweiligen Stand von Wissenschaft und Technik festzustellen. Die Rechtsvorschriften verweisen häufig auf technische Regeln, Standards und Normen sowie auf Richtlinien und Codes Practice. Zu beachten ist jedoch, daß es sich dabei streng genommen nicht um Rechtsvorschriften handelt,

da sie nicht von verfassungsrechtlich zur Schaffung von Rechtsnormen befugten staatlichen Organen ausgearbeitet und veröffentlicht werden. Betreiber, Genehmigungs- und Aufsichtsbehörden sowie Gerichte bedienen sich zur Feststellung des Standes von Wissenschaft und Technik im allgemeinen der Hilfe von Sachverständigen und Sachverständigenorganisationen.

4. Strahlenschutzgrundsätze

Die ICRP hat in der allgemeinen Publikation 26 (1977) in den Paragraphen 12 und 68 die wesentlichen Grundsätze für ein wirksames System des Strahlenschutzes aufgestellt:
- Dosisbegrenzungsprinzip;
- Zulassung nur solcher radiologischer Tätigkeiten, die einen positiven Nettonutzen aufweisen (Rechtfertigungsprinzip);
- Beschränkung der Strahlenexposition auf das vernünftigerweise erreichbare Maß, wobei wirtschaftliche und soziale Faktoren zu berücksichtigen sind (Strahlenminimierungs- oder -optimierungsgebot, ALARA-Prinzip).

Diese von der ICRP im Laufe der letzten Jahrzehnte entwickelten Grundsätze sind - wenn auch in unterschiedlichen Formulierungen und verschiedener Intensität - in das Strahlenschutzrecht der internationalen Staatengemeinschaften und der Staaten übernommen worden [9]. Das gilt vor allem für das Dosisbegrenzungsprinzip und für das Strahlenminimierungsgebot. Die Staaten haben bisher (von wenigen Ausnahmen abgesehen) jedoch noch keine näheren Vorschriften zur Konkretisierung des ALARA-Prinzips über eine Kosten/Risiko-Nutzen-Analyse mit verbindlicher Wirkung erlassen. Bemerkenswert ist in diesem Zusammenhang auch, daß in den USA diese Analyse nicht allgemein im Strahlenschutz obligatorisch gilt, sondern nur für die Festlegung von Mengen für die Ableitung gasförmiger und flüssiger radioaktiver Stoffe an die Umwelt zum Zwecke der weiteren Minimierung der Strahlenexposition der Bevölkerung, die im Umkreis von 80 km um einen Reaktor wohnt. Auf die Schwierigkeiten und die Anfechtbarkeit des 1.000 Dollar-man-rem-Wertes hat die Nuclear Regulatory Commission der USA bei der Aufnahme dieser Bestimmung in die NRC-Regulations selbst hingewiesen. In den letzten Jahren hat auf Grund der Empfehlungen der ICRP in der Publikation 26 zur praktischen Anwendung des ALARA-Prinzips und über die Gewinnung praktikabler Optimierungsmethoden eine ausführliche Diskussion stattgefunden, die offenbar noch nicht abgeschlossen ist.

5. Regelungsbereiche innerstaatlicher Strahlenschutzvorschriften

Prüft man die Strahlenschutzvorschriften der verschiedenen Länder, so ergeben sich zusammengefaßt etwa die folgenden Regelungsbereiche 10):
- Anwendungs- und sachlicher Geltungsbereich;
- Begriffsbestimmungen;
- Strahlenschutzgrundsätze;
- Genehmigungs-, Anzeige- und Überwachungsvorschriften, Freigrenzenregelungen, Bedingungen und Auflagen, behördliche Zuständigkeiten;
- bauliche und technische Anforderungen an Anlagen, Geräte und Vorrichtungen;
- persönliche Anwendungsvoraussetzungen (Fachkunde, Kenntnisse im Strahlenschutz, Ausbildung und Belehrung);
- Beförderung radioaktiver Stoffe;
- Einfuhr, Ausfuhr und Durchfuhr radioaktiver Stoffe;
- Ableitung und Beseitigung radioaktiver Stoffe;

- Bevölkerungs- und Umweltschutz (Dosisgrenzwerte für die Bevölkerung, Umgebungsüberwachung);
- innerbetrieblicher Strahlenschutz (Strahlenschutzorganisation, Schutzvorschriften für beruflich strahlenexponierte Personen, Strahlenschutzbereiche, physikalische Strahlenschutzkontrolle);
- Maßnahmen bei Störfällen und Unfällen;
- ärztliche Überwachung;
- Sonderregelungen für die medizinische Anwendung radioaktiver Stoffe und ionisierender Strahlen;
- Sondervorschriften für Lebensmittelbestrahlung und Radiopharmaka;
- Straf- und Bußgeldvorschriften.

ANMERKUNGEN

(1) Vgl. TAYLOR, L.S., Radiation protection standards, London 1971; MUTH, H., Die Entwicklung der Dosisgrenzwerte im Strahlenschutz, in: Röntgenstrahlen Nr. 47 (1982), S. 39 ff.; POCHIN, E.E., The development of radiation protection, in: Journal of the Society of Radiological Protection 1 (1981), Nr. 4, S. 17 ff.

(2) British Journal of Radiology 1 (1928), S. 358; Fortschritte Röntgenstrahlen 39 (1929), Nr. 2, S. 343.

(3) Zum Beispiel: Dänemark - Lov nr. 147 on Brugen af Roentgenstraaler m.v. (Lovtidende A 1930, S. 812); Frankreich - Décret du 5 décembre 1934 (JORF 1er janvier 1935, S. 29).

(4) Zur Definition des Strahlenschutzrechts vgl. BISCHOF, W., Zur Entwicklung des internationalen und innerstaatlichen Strahlenschutzrechts, in: Energiewirtschaftliche Tagesfragen 1978, S.671.

(5) Vgl. FISCHERHOF, H., Zur Terminologie des Atomenergierechts, in: Neue Juristische Wochenschrift 1962, S. 2095 ff.

(6) Vgl. dazu auch JACCHIA, E., Atom - Sicherheit und Rechtsordnung, Freudenstadt 1965; ERISKAT, H., Strahlenschutz - eine europäische Herausforderung bei der friedlichen Nutzung der Kernenergie, Frankfurt/Main usw. 1979; BISCHOF, W., Internationale Rechtsgrundlagen des Entwurfs der Strahlenschutzverordnung, in: Viertes Deutsches Atomrechts-Symposium, Köln usw. 1976, S. 39 ff.

(7) Nachweise der staatlichen Gesetzgebung im Strahlenschutz finden sich im Nuclear Law Bulletin (NEA/OECD, Paris), im International Digest of Health Legislation (WHO, Geneva) und im Göttinger Atomrechtskatalog (Institut für Völkerrecht der Universität Göttingen, bisher 29 Bände seit 1960). Vgl. ferner OECD/NEA, Regulations governing nuclear installations and radiation protection, Paris 1972, und Regulations governing the transport of radioactive materials, Paris 1980.

(8) Vgl. BISCHOF, W., und N. PELZER, Die Ausübung der staatlichen Kontrolle über die Einhaltung der Freigrenzen für Kernbrennstoffe und sonstige radioaktive Stoffe in den Ländern der Europäischen Gemeinschaften, Luxemburg 1973 (EUR 4906).

(9) Vgl. BISCHOF, W., in: Commission of EC, As Low as Reasonably Achievable, Luxemburg 1984 (im Druck).

(10) Vgl. IAEA PR 69/82 (8. Dezember 1969).

LATE HEALTH EFFECTS AMONG HIROSHIMA AND NAGASAKI ATOMIC BOMB SURVIVORS

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Atomic bombs were dropped on Hiroshima and Nagasaki in August 1945. The number of acute deaths due to the atomic bomb radiation, burns and/or mechanical injuries as of the end of 1945 is estimated to be somewhere between 90,000 and 120,000 out of population of approximately 330,000 in Hiroshima. The corresponding figures in Nagasaki would be between 60,000 and 80,000 out of around 250,000.

The Atomic Bomb Casualty Commission (ABCC) was established in the two cities by the United States government in 1947 to initiate follow-up studies on the late health effects of atomic bomb radiation. ABCC was soon joined in its studies by a branch laboratory of the Japanese National Institute of Health (JNIH).

A nation-wide survey of atomic bomb survivors was first conducted in 1950, enumerating a total of 285,000 survivors throughout the country. A cohort of about 110,000 subjects was thus selected from among these survivors, both radiation exposed and non-exposed controls, who were resident in Hiroshima and Nagasaki at the time of this survey.

Mortality in this cohort has been under study since 1950, and autopsies have also been performed on many of the individuals who have died. Since 1958, detailed biennial medical examinations have been carried out with 75-85% participation rates on a subsample comprising about 1/5 of this cohort. A separate cohort of 2,800 in utero exposed children and their non-exposed controls has been followed for mortality after birth and for morbidity.

In addition, a large scale genetic study was conducted based on the pregnancy registration in both Hiroshima and Nagasaki City in 1948-1954. Furthermore, a cohort consisting of 54,000 children whose parents were either exposed or not exposed was set up in 1958 for mortality follow-up and cytogenetic and biochemical genetic studies.

Under a reorganization in 1975, responsibility for the research performed by ABCC and JNIH was assumed by the Radiation Effects Research Foundation (RERF), an independent binational institution equally funded by the governments of Japan and the United States.

EXPOSURE ASSESSMENT

The atomic bombs dropped on Hiroshima and Nagasaki were of different design and content. A uranium bomb with an explosive yield equivalent of 12.5 kilotons of TNT was used in Hiroshima and a plutonium bomb with that of 22.0 kilotons of TNT in Nagasaki. Whereas the radiation generated was almost entirely gamma rays in Nagasaki, neutrons made a some contribution to the total dose received by survivors in Hiroshima; the amount, however, has recently become controversial and remains unsolved.

The dose actually received by individuals differs according to their shielding conditions. Their gamma and neutron kerma (kinetic energy released in materials) doses, with consideration of the attenuation from shielding, were tentatively calculated in 1965 (Tentative 1965 dose or "T65D"), but uncertainty continues to surround both the quantity and quality of the radiation released by these two nuclear devices, particularly the Hiroshima bomb. Only one weapon of the latter type ever having been detonated, its yield has had to be reconstructed from other weapon tests and reactor experiments.

A recent reassessment suggests that the gamma estimates used in T65D calculations might be too low in Hiroshima and high in Nagasaki, and the neutron estimates too high in both cities. Be this as it may be, given the uncertainties, attention here is restricted mainly to exposure expressed as total kerma (tissue) of T65D since this parameter changes least. The reassessment of atomic bomb dosimetry is now underway by the joint efforts of committees appointed by both Japanese and the United States governments.

LATE HEALTH EFFECTS

The late health effects of atomic bomb radiation which have been studied based on the various research programmes mentioned above are summarized in Table 1. These effects can be divided into the following three types: (1) effects for which a definite relationship to atomic bomb exposure has been established; (2) effects for which there is a borderline or suggestive relationship; and (3) effects for which no relationship has been shown.

Radiation effects are limited to an increase in some sites of malignant tumors as well as an excess of cataracts, chromosomal aberrations, small head size and mental retardation in the in utero exposed, and a delay in growth and development which occurred shortly after bombing.

We failed to observe any radiation-related acceleration of non-specific aging, suppression of immune response and fertility, or genetic effects. However, these negative findings can not be considered as establishing that such effects do not exist. They may mean that sample sizes were not adequate or the indices of radiation effects employed were not sufficiently sensitive.

Because none of the disease entities which have been studied are uniquely attributable to radiation, it has been necessary to employ the epidemiological approach to investigate their possible relationship with radiation. This means that there is always a possibility that some unknown bias could affect the results. For example, our mortality study cohort which was, as mentioned before, set up in 1950 does not include those who died in the interval from exposure to 1950.

It is, therefore, conceibable that survivors with a strong resistance to radiation had been selectively included and thus the effects could be underestimated. There are available, however, incomplete as they may be, lists of survivors made several months to one year after the bombing. These individuals too are under

scrutiny, and the effect of selection is believed to be small.

Table 1. Summary of Findings on Late Effects of Atomic Bomb Radiation

ABCC-RERF, 1950-1981

I. Established Increase in:
1. Malignant tumors (leukemia, thyroid, breast, lung, stomach, multiple myeloma)
2. Lenticular opacities
3. Lymphocyte chromosomal aberrations
4. Small head size and mental retardation in the in Utero exposed
5. Retardation of early growth and development
II. Suggestive Increase in:
1. Malignant tumors (esophagus, colon, salivary glands, urinary tract, malignant lymphoma)
2. Certain changes of humoral and cell-mediated immunity
III. No Demonstrable Increase in:
1. Malignant tumors (chronic lymphatic leukemia, osteosarcoma)
2. Mortality from other than malignant tumors
3. Accelerated aging (including cardiovascular disease)
4. Infertility
5. Birth defects or mortality in F_1 generation

It is observed that the risk of radiation-induced malignant tumors other than leukemia has increased with age. Since those who were young at the time of bombing have just entered the cancer age, careful long term follow-up of the cohort should be continued.

INTERIM SHIELDING DOSE ESTIMATES

Interim shielding dose estimates were tentatively calculated for 75,183 survivors belonging to our mortality study cohort, utilizing free-in-air dose curves by Pace and Scott of Oak Ridge National Laboratory (ORNL) and shielding factors developed by Marcum of R & D Associates. Deaths from leukemia, deaths from all sites of cancer except leukemia, chromosomal aberration and epilation were used as biological indices for assessing the effects of radiation as measured by these "ORNL" doses and "T65D". Results thus obtained are as follows:

1. The Hiroshima-Nagasaki inter city differences in the dose-response are smaller for ORNL than that for T65D, but the risk is still higher for Hiroshima than for Nagasaki.

2. The estimated risk coefficients for gamma rays based on a linear model for both gamma rays and neutrons are larger for ORNL than for T65D for each effect, and the ratios of coefficients (ORNL vs T65D) are in the range of 1.2 to 1.7.

3. As the ORNL neutron dose becomes very small even in Hiroshima, estimates of risk coefficients for ORNL neutrons may not be warranted. However, similar estimates as for gamma rays suggest that the risk coefficients are larger for ORNL than for T65D and ORNL-T65D ratios are in the range of 4 to 6. The estimated relative biological effectiveness (RBE) for neutrons using ORNL dose and the linear model indicate that the RBE values for neutrons are larger for the ORNL dose than that for the T65D dose.

Figure 1 shows the dose-response curves for chromosomal aberrations. The relations are quantitatively similar to those observed for leukemia death rates. For deaths from all sites of cancer except leukemia and for epilation a similar tendency was also seen.

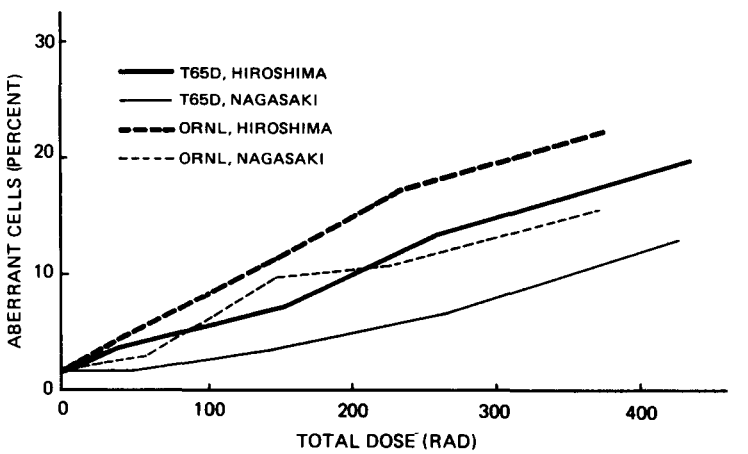

Figure 1 Chromosome aberrations for T65 dose and ORNL dose, Hiroshima and Nagasaki

RECENT DEVELOPMENT IN RADIATION PROTECTION POLICY

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INTRODUCTION

When this paper is presented in Berlin, there may be some more recent development; the manuscript was requested one half year in advance.

Policy rather than regulatory action will be discussed. There have been interesting steps in the direction of improved regulations in a number of countries, but there will not be enough space or time to treat this here.

Policy discussions often occur at the international level, within organizations such as IAEA, ICRP, OECD/NEA and WHO, not to mention the IRPA international and regional Congresses. This does not mean that ideas are not conceived in national organizations and regulatory bodies, but merely that they are hatched at the international level.

In describing recent development, the material can be arranged in many ways; for example, by radiation source or practice, by the receiver of the dose (workers, members of the public, patients), by international organization, or by policy concept (optimization of protection, dose limitation, exemption policy, etc.). No way is entirely logical, and the following presentation is therefore somewhat arbitrary.

One reference document is the paper "International Radiation Protection Recommendations - 5 years experience of ICRP Publication 26" by Lindell, Beninson and Sowby, presented at the IAEA International Conference on Nuclear Power Experience, in Vienna, 1982.

GENERAL OBSERVATIONS

The problems of nuclear power have lately dominated the interest and the resource allocation of radiation protection. It is therefore only natural that developments in this field will dominate also this presentation. The emphasis has shifted slowly from the normal reactor operation to accidents and emergency planning, and then further to long-term high level waste disposal.

The highest individual radiation doses outside of radiation therapy, and the largest collective doses in many countries, however, come from natural sources of radiation. The extreme variation of this exposure and, particularly, the unbelievably high lung doses from radon daughter products in several countries have come as a surprise, even though the problem has been known for many years. This has made the development of a radiation protection policy for natural radiation urgent.

A shift in the interest of some international organizations has also influenced policy development. For example, WHO used to be interested primarily in the medical uses of radiation but has recently

shown a greater interest in the environmental aspects of the nuclear power industry. NEA used to give some non-nuclear energy questions considerable attention, e.g. consumer products and radioactive building materials, but, for a period, was limited to the nuclear energy field by policy decision. ILO used to give significant weight to the protection of radiation workers, but has for many years been less active in this area. Changes in the relative emphasis of efforts by international organizations should not necessarily be seen as something negative; they are sometimes signs of vitality and flexibility, but they always influence the way in which international radiation protection policy is formed.

Non-ionizing radiation, a subject in which IRPA has a vested interest, is deliberately avoided by ICRP as not falling within its present competence and resource dedication. However, in many countries, the NIR problems have been found to be of great practical importance. Contrary to common belief, it is not the microwave sources that create the serious problems but artificial sun lamps and high-powered lasers. Indeed, the sun itself is not only the major source of life and energy but perhaps also of skin cancer. The recognition of these problems has called for new policy approaches, in some cases not unlike those related to the radon problem. How far is it reasonable or necessary to control natural sources of radiation?

More experience has accumulated on the practical application of the recommendations in ICRP Publication 26, the basic radiation protection recommendations. IAEA, ILO, NEA and WHO have completed their new Basic Safety Standards on this basis, thereby consolidating recommendations and concepts which were first received with some hesitation. The NEA Committee on Radiation Protection and Public Health, in particular, has been active in exploring practical application problems and policy gaps.

The heavy emphasis on optimization of protection caused a number of policy problems. The ICRP dose limits obtained a new meaning, which caused a communication problem. The lack of application recommendations at the time when ICRP Publication 26 was published gave the basic recommendations a touch of abstractness which repelled some of those who were immersed in practical issues. The situation improved after thorough discussions within other organizations and with the publication of the complete set of ALI- and DAC-values in ICRP Publication 30 and the explanations given in ICRP Publication 37 on Cost-Benefit Analysis.

The ICRP policy of limiting the annual committed dose rather than the actual annual dose has caused some application problems, particularly with regard to radionuclides with long effective half-lives, such as plutonium isotopes. A body burden of plutonium that would only give a small annual dose may still imply that the committed dose in some year has exceeded the ICRP dose limit. This has seemed objectionable to some national authorities which prefer to limit the actual annual dose. However, the annual dose alone does not provide the necessary information that there is less room for continued intakes if there is a body burden of a long-lived nuclide. On this particular issue, there is no conceptually new approach in ICRP Publication 26. Even with the previous recommendations, a full body burden of plutonium should not have been considered acceptable until the end of an occupational life-time. The problem was, and

still is, the practical difficulties in assessing the annual committed dose rather than the actual annual dose that can be calculated from the observed body burden.

AREAS OF POLICY DEVELOPMENT

The most recent policy developments are, not surprisingly, within areas of current radiation protection interest. Since achievement is often a sigmoid function of time, and policy development describes the slope of the curve, lack of development is characteristic of areas where problems have either been solved or not recognized. Development is to be expected where new problems have been detected or unsolved problems recognized. Accordingly, the main areas of recent policy development are those of natural radiation, waste disposal, emergency planning, and refinement of basic concepts. In addition, there have been numerous policy discussions without development, merely because of misunderstanding of or objection to proposed changes in policy. Such discussions will not be reviewed here, some of the topics were treated in the Lindell-Beninson-Sowby review paper of 1982.

It is interesting to notice that, in parallel with the development in the field of radiation protection, other areas of human protection show a tendency of moving towards similar principles. This goes for occupational as well as for environmental protection. This may have less to do with assumptions of linear, non-threshold dose-response relations than with concern that multiple risk sources may eventually add up to unpleasant situations unless the full picture is considered. The widespread sulphur contamination and acid rain is an example of such situations. Although the new thinking is still far from the main-stream, it is nevertheless of interest as a trend.

Quantities, Units, Concepts

The introduction of the SI units with the special names gray, sievert and becquerel is no longer a recent development, although, unfortunately, some countries, for example the United States, show no intention of imminent change. The prolongation of the transition period may cause considerable confusion. Organizations like IRPA may help avoiding such confusion. Like the SI units, quantities such as effective dose equivalent, collective dose and dose commitment have now lost their property of being "new". They have been used by both ICRP and UNSCEAR since 1977 and are generally accepted internationally.

However, some concepts and quantities are still subject to international discussions. The meaning of the dose limit has been further analyzed.

The ICRP dose limit is an individual-related limit, designed to limit any individual's total radiation dose from all sources (natural radiation and medical exposures excluded). However, the dose limit as such is not always suited to give the boundary condition for optimization of protection against one particular source. What is needed in that case is a dose "upper bound" such that the dose contributions from all sources with optimized protection will not add up to a total that exceeds the dose limit. The upper bound is source-related in the sense that it restricts the optimization procedure

for the particular source under consideration, and perhaps even for a particular protective action relating to that source (e.g. the dimensioning of a protective barrier).

Another concept, of old origin but recent recognition is the authorized limit. If an optimization assessment indicates that a certain protective measure is reasonable, then that measure should be mandatory (unless some other method is even better) if all radiation doses are to be kept "as low as reasonably achievable". This may be enforced by formal requirements, which may be in the form of authorized limits (e.g. release limits, limits for the minimum barrier thickness, etc.).

In the author's opinion, it would not be appropriate to give a dose upper bound as the authorized limit. The upper bound should give the boundary condition for the optimization assessment, while the authorized limit should be based on the optimization result (restricted by the upper bound). Since the upper bound is a dose restriction and the authorized limit is often a limit of something else, giving both would create an over-determined system. The authorized limit should be a limit of something which is easy to control and measure, preferably before the exposure (as is the case with release limits and barrier requirements), not a limit of something which will be found as the resulting fact (as the dose). The resulting fact will be useful in a different sense, as an indicator of whether the authorized limits have been respected or correctly chosen.

There have been discussions within ICRP and ICRU on the quantity dose equivalent, its definition, usefulness and shortcomings. It is not likely that this will lead to any significant policy changes.

The secondary limits recommended by ICRP for practical use are the ALI and a limit of the dose equivalent index. The recommendation on the use of the index has caused some confusion because of not too clear wording of paragraphs 107 and 108 of ICRP Publication 26. It is the author's interpretation that the intention was to endorse the usual practice of letting the reading of personal dose meters be the "dose" that is compared with the dose limit, provided that the dose meters are calibrated to read approximately the dose equivalent index. The intention was not to encourage detailed assessments of the actual dose equivalent in everyday practice. The procedure is justified if the index does not underestimate the effective dose equivalent significantly. The relation between the index and the effective dose equivalent is being studied by an expert group within ICRU.

The ALI-values recommended by ICRP reflect the annual dose limit for workers. The derivation of ALI-values for the public is an intricate matter. The annual dose limit for members of the public is 1/10 of the annual dose limit for workers, but there is the additional recommendation that it would be prudent to keep the effective dose equivalent within 1/50 of the limit for workers, on average, in the case of life-long exposure.

Because of the different ages and physiological parameters involved, and also because of the different routes of intakes and chemical form of the nuclides (in the case of the public, the food-chain is of importance), the ratios of ALIs cannot be expected to be equal

to the ratios of the dose limits. Conservative ratios have been proposed, for example, in the Basic Safety Standards of the international organizations, but may involve an oversimplification. In 1983, ICRP adopted a statement on ALI for members of the public, describing the problems. For some nuclides used as examples, the ALIs for infants aged six months will be smaller than those given to limit stochastic effects in workers, by factors that range from just more than 10 (for cesium-137) to 1000 (for ingested plutonium-239). In the case of the more cautious life-long limitation the further factor of 5 cannot be applied to these values since individuals are not perpetual infants. The ALIs for adults would be related to those for workers by intermediate factors. For these the factor of 5 may be relevant in the case of prolonged exposures. The ICRP statement illustrates the shortcoming of one single factor for all circumstances.

The ICRP choice of quality factors has been debated. There are biological indications of high RBE for high-LET radiation (e.g. neutrons) in relation to low-LET radiation (gamma rays) at low doses. This would also be expected if the dose-response relationship is linear for the high-LET radiation but linear-quadratic for low-LET radiation up to the dose levels at which epidemiological observations have been made. If that is the case, a direct interpolation between zero dose and the high doses for which there are observations would overestimate the risk of the low-LET radiation but would justify a quality factor correspondingly lower than the true RBE. If, instead, the risk factors for low-LET radiation have been based on other assumptions and are believed to be realistic at low doses (but underestimate the risk at high doses), then the ICRP quality factors for high-LET radiation may be too low. This question is being studied but no generally accepted answer is yet available. The Loewe and Mendelsohn (Lawrence Livermore National Laboratory) revision of the "T65D" dose estimates for Hiroshima and Nagasaki have invalidated an important basis (perhaps the only) for human neutron risk estimates. This causes a further complication.

Optimization of Protection

An important policy step is the recognition of complete identity between the concepts "as low as reasonably achievable" and "optimization of radiation protection". It is also generally agreed that the differential cost-benefit analysis described in ICRP Publications 22, 26 and 37 is only one method of finding the optimum solution. Other methods, e.g. multicriteria methods and aggregate methods based on non-linear utility functions are being developed.

ICRP Publication 37 makes a distinction between the "objective" radiation health detriment (the "alpha" term) and other components of detriment (the "beta" term). A recent development is the IAEA efforts to find an internationally agreed minimum value of "alpha", i.e. the monetary cost assigned to a unit collective dose, to apply to collective doses outside the jurisdiction of the country that causes a widespread environmental pollution.

The "beta" term is still under discussion. Some national authorities propose to compensate a low value of "alpha" by giving more weight to collective doses caused by individual exposures near the dose limit. Other authorities question any quantitative relation

between "subjective" detriment and radiation dose, and prefer to derive the optimum result from differential cost-benefit analysis with the "alpha" term alone, supplemented by less analythical methods to decide on extra protection if there are additional components of detriment that the authority wishes to eliminate.

There have been questions about the ethics of allocating a sum of money for protection per unit collective dose avoided, the implication being that this is equivalent to setting a price on human lives. However, the purpose of optimization is to obtain the maximum benefit with the minimum disadvantages. Failure to optimize may lead to unnecessary total harm. It is the duty of those responsible for radiation protection to request the highest valuation of collective doses that is not in conflict with other needs of society. To ask for "indefinite" efforts in one area would mean denying other areas legitimate attention to an extent that would be truly unethical. This problem has recently been discussed within a Study Group appointed by the Pontifical Academy of Sciences, from which a report may be issued.

Differential cost-benefit analysis for optimization of protection is sometimes confused with cost-effectiveness procedures which are conceptually different.

Release Limitation Policy

It is interesting to follow the development of a more cautious policy with regard to releases of radioactive effluents. This is illustrated by the shift of title of the IAEA report on "Principles for Establishing Limits for the Release of Radioactive Materials into the Environment"(IAEA Safety Series No.45, 1978) which was intended to be called "The Capacity of the Environment Safely to Receive Radioactive Materials" when the work was planned in 1974. Although this useful report was written before ICRP Publication 26 was published, it was consistent with the ICRP policy. An Annex was published in 1982 to bring Safety Series No. 45 up to date. The IAEA document thus amended is an important source of policy advice.

Radioactive Waste Disposal

Radioactive waste disposal is a subject which has attracted more and more public attention. It involves a number of important radiation protection policy aspects which need new approaches because of the time scales involved. Any exposure of the public is either due to very rare events or will occur first after so long time that the reliability and relevance of any dose estimates are seriously questioned. This invites probabilistic assessment methods and discounting procedures, including the "de minimis" concept. Such methods and procedures have not obtained general acceptance and this is therefore a field of current discussions and policy development.

The probabilistic approach is based on equal risk limitation. If, for a given practice such as high level waste disposal, an upper bound for the individual effective dose equivalent is given, this implies a limitation of the individual probability of serious harm at a certain level p_{lim} corresponding to the dose upper bound. This is the probability limit of harm if the dose is received with cer-

tainty. It seems reasonable to say that any event that causes a dose high enough to cause harm with certainty must also have its probability limited at p_{lim}. It then follows that events causing a dose, H, higher than the upper bound, having a probability $p(harm|H)$ of causing harm, should be permitted only if its probability $p(H)$ makes $p(H) \cdot p(harm|H) < p_{lim}$.

These questions are discussed both within IAEA and NEA, and in ICRP. They were reviewed at the IAEA International Conference on Radioactive Waste Management in Seattle in May, 1983.

The probabilistic approach may lead to a workable policy with regard to the individual risk limitation, but the basis for optimizing protection on probabilistic grounds is still very weak, for a number of reasons. One is that it seems impossible to arrive at any meaningful estimate of collective doses at the future time when radioactive material once safely deposited may leak into the atmosphere. Cost-effectiveness assessments may make it possible to indicate a minimum requirement on a repository, but there is hardly a reliable basis for differential cost-benefit optimization in the choice of alternative solutions or dimensioning of parameters. Another reson is that the collective dose from scenarios with very low probabilities cannot usefully be presented as a mathematical expectation value, since its standard deviation would be so large that it would not provide a suitable basis for decisions. The usual differential cost-benefit analysis, therefore, cannot be expected to be useful and the optimization must be based on other approaches.

Emergency Planning

The question of emergency planning for reactor accidents has been extensively treated in many countries after the TMI accident, and the approaches have become more realistic with the increased understanding of likely accident scenarios. An expert group report from a meeting arranged by WHO in Brussels in Brussels in 1981 is a good example of the current school of thinking. Although inhalation risks may previously have been underestimated, it seems that severe health problems would only be caused if there is a deposition of radioactive material on the ground. The technical and organizational problems involved in localizing such deposition are difficult, but would have to be solved in order to give sufficient basis for remedial actions. If evacuation is justified, reentry may not be advisable for a very long time. Good reentry criteria have not yet been established and it is interesting to compare various proposals with society's actual reaction in the case of buildings that have been found to have very high radon concentrations.

One major weakness with all emergency planning is that there is yet no realistic assessment of the source term. Accident scenarios have been based on the assumption that certain fractions of the core inventory of radioactive substances are released into the environment, but the physical qualifications for this are not well understood.

Natural Radiation and Radon

According to ICRP Publication 26, the ICRP dose limits do not apply to doses from "normal" natural radiation. Expert groups within

NEA and ICRP have tried to formulate guidance on how to deal with natural radiation sources, e.g. radioactive building materials. However, over the last few years the severe problems caused by radon in dwellings in many countries has been recognized as one of the most urgent, perhaps the most urgent, radiation protection task. Equilibrium equivalent concentrations (EEC) of radon of more than 10 000 Bq/m^3 have been found in some homes, exposing lung tissues to near 10 Sv (1000 rem) per year.

In October, 1983, ICRP adopted new recommendations on protection against natural radiation. The distinction between "normal" and "elevated" levels of natural radiation was not found helpful. Instead, the Commission suggests that the controllability of the exposure is an important factor.

In the case of existing situations, e.g. an existing building with a high radon concentration, the only action is remedial action. For this some action level may be recommended, but since the justification of a remedial action would depend on not only the level of radiation risk but also on the type of action, no generally applicable action level can be given. In the case of radon in dwellings, however, if the action is fairly simple, the Commission suggests that an action level in the region of 200 Bq/m^3 (EEC) might be considered.

For future exposure situations, e.g. planned new buildings, the basic principles of justification of practice and optimization of protection should apply. For artificial sources of radiation, the boundary condition for the optimization procedure is determined by the ICRP dose limit or some fraction thereof in the case that there are several sources. For natural sources of radiation, the dose limits do not apply, but because of the high radon levels currently found, there is an obvious need of some upper bound.

The new ICRP recommendations suggest that, in principle, some indication of reference levels of risk would be helpful in order to establish an appropriate upper bound for exposures from radon indoors. For example, this could be the overall risk from other risk sources at home (falling down stairs, electricity, fires, etc.). However, these risks are highly age dependent and a reference risk is not easily derived. The Commission expresses the belief that "a reasonable upper bound for the equilibrium equivalent radon concentration is of the order of 100 Bq/m^3 and that, in many countries, a value of this magnitude would prevent radon from becoming a dominant source of risk in dwellings". The optimization of protection under the boundary condition of this upper bound would influence building standards for construction, manufacture, ventilation, etc.

CONCLUDING REMARK

This review, obviously and by practical necessity, has been subjective; other authors might have made a different selection of examples with different emphasis. In particular, the development is seen from the viewpoint of a Swede and ICRP member, which is an explanation if not an excuse for the bias undoubtedly present although not intended. The reader is advised to make the necessary allowances.

CONTAMINATION OF RESIDENTIAL AREAS BY A FORMER RADIUM PROCESSING PLANT

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INTRODUCTION

Residential areas affected by increased radon concentrations and gamma radiation levels have previously been reported for lands contaminated with uranium mill tailings,[1,2] and land enriched in radium 226 from wastes from phosphate processing.[3]

In Sydney, Australia a plant operated from 1912 to 1915 to produce radium bromide from uranium ore concentrates which originated in South Australia and were transported to the plant, a distance of 2,000 kilometres, by sea.

A total of approximately 500 tonnes of low grade ore concentrate (1.4% uranium) were treated to produce an estimated 1.8 grams of radium bromide.[4]

The site has since been developed as prime waterfront residential areas, and a radiation survey of the site was carried out after a request from a resident who had learned of the plant's existence from records with the local historical society.

DESCRIPTION OF AREA

The processing plant was built on land reclaimed from Sydney harbour and extended up a sheer cliff face to road level. A series of terraces were constructed to enable level surfaces to be created for the construction of the plant. The total area of contamination was 2000 m^2 which includes 1300 m^2 over six residential blocks and 700 m^2 of harbour foreshore. The estimated volume of contaminated soil is 1000 m^3 with an average depth of 0.5 metres. A plan of the contaminated areas is shown in Fig. 1.

MEASUREMENT TECHNIQUES

(i) Dose rates were measured at waist level using Berthold LB-1200 and Eberline PRM-7 monitors calibrated against a radium 226 standard.

(ii) Radon was measured by the two filter method [5] and by a Johnson 955B radioactive gas monitor.

(iii) Working levels were determined by the Rolle method.[6]

(iv) Radium 226 in soil samples was determined either by direct gamma counting or by radiochemical analyses.

CONTAMINATED AREAS

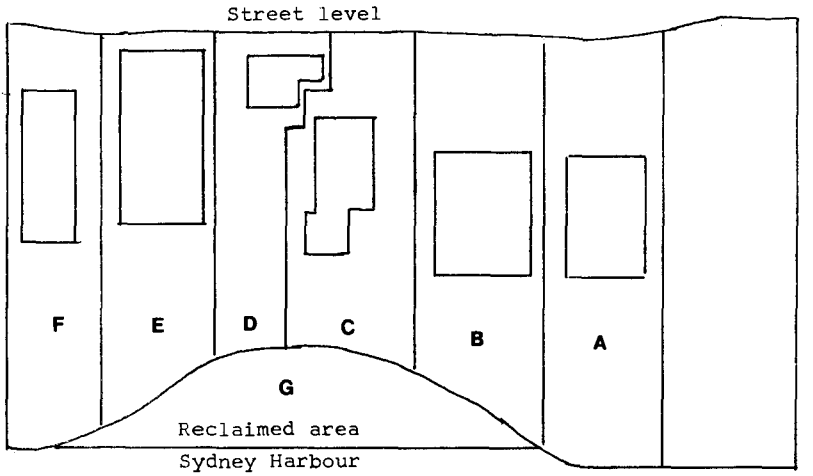

Fig. 1

Area A

A small area of contamination near the cliff face had a dose rate 1 μSv/hr and a radium content of approximately 6 kBq/g. The area on which the house was constructed was not contaminated.

Area B

From old photographs of the plant, this is identified as the site of the plant manager's residence, since demolished. Areas around the new house were free of contamination. Radon was not detectable inside the house.

A hot spot with a dose rate of 30 μSv/hr was found in the grounds and consisted of broken bricks, crucibles etc. Gross alpha and beta activity of samples of this rubble were an average of 60 kBq/g and 190 kBq/g respectively. Scrapings from an iron bar had a gross alpha activity of 920 kBq/g and gross beta activity of 3.2 MBq/g.

Apart from this "hot spot", general contamination of about 4 kBq/g of radium was found in an area near the swimming pool constructed adjacent to the house. The dose rate was 0.5 μSv/hr.

Area C

A soil sample taken from under the kitchen of the house had a radium concentration of 244 kBq/g and it is likely that the house has been erected over the site of the plant's chemical laboratory. Soil under other areas of the house had an average radium content of 18.5 kBq/g

A number of spot measurements of radon and working levels were made which showed wide fluctuations. Radon varied from 260 - 3000 kBq/m^3 with an average of 1200 kBq/m^3 and working levels ranged from 0.02 - 0.32 with an average of 0.12.

Fluctuations in radon levels were also evident with the use of a continuous radioactive gas monitor. The average radon concentration was 190 kBq/m^3 with a maximum concentration of 2600 kBq/m^3. Radon built up to a maximum between 9.00 am and 6.00 pm each day.

The average indoor gamma dose rate was 0.3 µSv/hr and the average outdoor dose rate was 0.5 µSv/hr.

The radium concentration of surface samples ranged from 1.1 to 24 kBq/g with an average of 37 kBq/g. Core sampling showed maximum activity at 0.7 metres.

Area D

The grounds were extensively contaminated. However, the house was built on the upper (road level) and was not affected by increased radon or gamma levels (working level 0.003, dose rate 0.1 µSv/hr).

The average dose rate in the grounds was 0.5 µSv/hr with localised "hot spots".

Samples of soil from one "hot spot" with a dose rate of 10 µSv/hr had an average gross alpha activity of 27 kBq/g and gross beta activity of 92 kBq/g. Another "hot spot" had a radium concentration of 592 MBq/g.

The radium concentration of surface samples excluding the "hot spots" ranged from 1.1 - 37 kBq/g with an average of 13 kBq/g.

Activity increased with depth to a maximum at 0.7 metres.

Area E

One "hot spot" found in the grounds with a dose rate of 10 µSv/hr had an activity of 1.5 MBq/g of radium.

The rest of the block was relatively free of contamination due to extensive movement of soil for the construction of the house. The house was not affected by increased radon or gamma radiation.

Area F

One "hot spot" was found which had a radium concentration of 52 MBq/g and was a cardboard type material. Apart from this, radium concentrations of surface samples ranged from 1.1 - 74 kBq/g with an average of 15.5 kBq/g.

Area G

This was land reclaimed from the harbour. Contamination was unevenly distributed. One "hot spot" had a radium concentration of 137 kBq/g. The average radium concentration over the rest of this area was 8 kBq/g. The average gamma dose rate was 0.5 µSv/hr.

REMEDIAL ACTION

The Radiological Advisory Council, of New South Wales has adopted the recommendation of the U.S. Surgeon General's guidelines for construction of houses over land contaminated with uranium mill tailings[7]. It has recommended removal of contaminated soil.

DISCUSSION

This contamination problem is unlike other cases of contaminated residential areas in that -

(i) the plant was not associated with any nearby deposit of uranium ore

and

(ii) Radium was being extracted and not dumped. That is, radium would ingrow from Uranium 238 and Thorium 230 and radioactivity would increase with time.

In addition the plant was operating long before any legislation regarding radioactive materials was introduced in New South Wales and subsequently there were no records of the plant with the Radiation Health Services. Subsequently the contamination would have remained undiscovered if the local resident had not learned of the plant's existence.

REFERENCES

1) United States Congress. Joint Committee on Atomic Energy. Use of Uranium Mill Tailings for Construction Purposes. Washington, 1971.

2) Spurgeon, D. Nature 260:278 (1976).

3) United States Environmental Protection Agency: Indoor Radiation Exposure Due to Radium-226 in Florida Phosphate Lands. Feb 1979 EPA 520/4-78-013.

4) South Australian Review of Mining Operations; 1908-1912, No. 17.

5) Thomas J.W., & Le Clare P.C., Health Physics 18; 113-122 (1970).

6) Rolle R., Health Physics 32; 233-238 (1972).

7) U.S. Atomic Energy Commission (1972) Grand Junction Remedial Action Criteria - Federal Register, Vol. 37 (235) - Wednesday December 6th - pp. 25918-25919.

THORIUM - 232 CONTAMINATION IN A HOLIDAY RESORT

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INTRODUCTION

Beaches with deposits of mineral bearing sands on the coast of New South Wales, Australia have been mined and milled for a number of years to extract zircon, rutile, monazite and ilmenite. Monazite, containing thorium 232, was of minor importance and emphasis was on the extraction of rutile and zircon. Monazite was usually stockpiled until it was economically viable to process to a 90% concentrate suitable for export.

In Byron Bay (population 3500), a holiday resort 900 km north of Sydney, a sand processing plant operated from 1934 to 1975. This mill was situated in the main street and adjacent to the commercial centre of Byron Bay. Byron Bay has a high annual rainfall and in order to make land suitable for construction purposes, it has been necessary to use filling material to reclaim swampy areas. Monazite and ilmenite have been widely used throughout the township for this purpose. Consequently a number of houses and other buildings have been constructed on land containing thorium, resulting in an increased gamma radiation exposure to the occupants.

RADIOLOGICAL INVESTIGATION

A. Measurement Techniques

(i) External dose rates were measured at waist level with Eberline PRM-7 monitors. These were calibrated against cobalt 60 and radium 226 standard sources, with an accuracy of $\pm$ 10%.

(ii) Thorium and Uranium concentrations in soil samples were determined by Ge - Li spectroscopy.

B. Survey

A preliminary survey identified the school, a supermarket and a number of houses having increased gamma radiation levels when compared to background. Contamination was found to be widespread throughout the town, however, there was no general contamination; that is, in some instances a block was contaminated but those on either side were unaffected. A house to house survey was then carried out to fully establish the extent of contamination, to define contaminated areas, to measure radiation levels and to formulate procedures for any required remedial actions.

C. Remedial Action Criteria

Guidelines for action levels in land contaminated with uranium mill tailings have been proposed by the U.S. Surgeon General[1] for cleanup of uranium mill tailings in Grand Junction and Salt Lake City. These guidelines have been adopted by the Atomic Energy Commission of Canada[2] for areas of significant contamination for Port Hope and Elliott Lake in Ontario and Uranium City in Saskatchewan.

The relevant criteria adopted by the Radiological Advisory Council of New

South Wales for land contaminated with beach sand residues is:-

Indoor radiation levels -
(i) Remedial action is necessary if the indoor radiation level exceeds 1 uSv/hour (100 urem/hr) at 1m above floor level.
(ii) It is desirable to reduce the indoor radiation levels to 0.6 uSv/hour (60 urem/hr).

Out of door radiation levels -
(i) Remedial action is necessary if the out of door radiation level exceeds 1 uSv/hour (100 urem/hr) at 1m above bare ground.

RESULTS

A. Gamma Radiation

(a) Residential Areas:- A summary of the range of indoor gamma levels in residences built over land contaminated with thorium and the maximum levels found outdoors are presented in Table 1.

TABLE 1

Residence	Indoor Levels uSv/hr	Outdoor Levels uSv/hr	Residence	Indoor Levels uSv/hr	Outdoor Levels uSv/hr
A	0.6 - 1.2	1.5	M	0.4 - 1.8	1.0
B	1.0 - 2.8	1.5	N	2.0 - 2.8	2.0
C	1.0 - 2.2	2.0	O	2.2 - 2.8	3.8
D	0.6 - 1.4	1.8	P	0.8 - 1.0	1.0
E	1.0 - 4.8	1.4	Q	0.7 - 3.2	27.0
F	1.6 - 1.8	1.8	R	1.1 - 1.5	1.8
G	0.4 - 1.2	1.4	S	0.5 - 1.5	1.5
H	2.0	3.5	T	0.3 - 1.0	1.8
I	2.6 - 3.0	9.0	U	0.75- 1.0	1.1
J	1.7 - 2.2	5.0	V	0.4 - 1.2	1.3
K	1.6 - 2.8	4.0	W	0.2 - 0.9	3.5
L	0.6 - 2.0	1.0			

All residences are of wooden floor construction with the exception of Q and T which have cement floors. Soil under those residences with a wooden floor are easily accessible and removal of the soil and replacing with clean fill is the remedial action to be carried out.

Residence Q had the highest gamma dose level of 27 uSv/hr in the garden. The thorium concentration of soil from the area is approximately 0.8% which suggests that a 30% monazite concentrate was used as fill. As it is probable that the outdoor contamination is contributing to the indoor gamma exposure, the remedial action to be carried out is firstly to remove all contaminated soil from the gardens and to then re-assess the indoor gamma levels.

(b) Other Areas:- (i) School
Playing field - 0.9 - 3.6 uSv/hour
Transportable classrooms -
(a) 0.7 - 1.5 uSv/hour (b) 0.2 - 0.5 uSv/hour
(c) 0.6 - 0.8 uSv/hour (d) 0.3 - 0.7 uSv/hour
(e) 0.3 - 0.6 uSv/hour

Soil from the playing field was removed and replaced with clean fill, re-

ducing gamma dose levels to 0.3 uSv/hr. Transportable classroom (a) was re-sited to an area where the gamma dose level was 0.4 uSv/hr and contaminated soil under the original site was replaced with clean fill. The gamma dose levels were reduced to 0.2 uSv/hr.

(ii) Supermarket (original site of the mill)
(a) Loading bay area - 1.4 - 9 uSv/hour
(b) Carpark - 0.3 - 0.9 uSv/hour
(c) Supermarket building - background

The supermarket was under construction and soil removal from the loading bay area reduced the levels to 0.4 uSv/hr.

(iii) Hospital
(a) Areas with wooden floor - 0.4 - 2 uSv/hr
(b) Areas with cement floor - background

Contaminated soil has been removed from under the hospital and replaced with clean fill to reduce gamma dose levels to 0.2 uSv/hr.

(iv) Girl Guides' Hall
1.2 - 5 uSv/hour

(v) Tailings Dump (proposed residential development)
1.5 - 7 uSv/hour

(vi) Baby Health Centre
Indoor - 1.4 - 2.0 uSv/hr
Outdoor - 2.2 uSv/hr

(vii) Church
Indoor - 0.4 - 1.4 uSv/hr
Outdoor - 1.7 uSv/hr

(viii) Bank
Indoor - 0.5 uSv/hr
Outdoor - 3.0 uSv/hr

(ix) Masonic Hall
Indoor - 0.6 uSv/hr
Outdoor - 1.0 uSv/hr

(x) Old People's Home (cement floor)
Indoor - 0.3 uSv/hr
Outdoor - 2.0 uSv/hr

(ix) Recreation Park - 1.0 uSv/hr

In addition to the areas discussed above, there are numerous areas such as gardens, driveways, roads, etc. where contaminated sand has been used as fill to create localised areas of contamination. These areas were assessed on an individual basis to determine the necessary remedial action.

B. Thorium and Uranium Concentrations

Table 2 summarises concentrations of thorium and uranium in selected soil samples together with the calculated dose rate [3] at 1 metre for soil with these concentrations and measured dose rates.

TABLE 2

Sample	Thorium ppm	Uranium ppm	Calculated Dose Rate (uSv/hr)	Measured Dose Rate (uSv/hr)
Residence Q	7600	500	23.0	27.0
Residence R	120	110	0.9	1.2
Residence F	370	74	1.4	1.8
Residence E	860	79	1.9	1.4
Residence O	580	140	2.3	30.0
School (classroom (a))	970	130	3.3	2.4
Girl Guides' Hall	450	130	7.2	5.0
Tailings Dump	560	49	1.8	1.8

Taking into consideration that these are spot samples and that variations in concentrations of thorium and uranium would occur over area and depth, the calculated dose rates are in reasonable agreement to the measured dose rates.

DISCUSSION

It is apparent from the age of the majority of affected houses (30 to 40 years) that most contamination had occurred in the infancy of the mill. One exception is residence Q, about 5 years old, where contaminated fill was apparently obtained from a dump of 30% monazite concentrate after the mill had ceased operation. Remedial action in the form of excavation of radioactive soil and replacement with clean fill is currently being undertaken to clean up the town, with the local council, the sand mining company involved and the state government sharing costs.

This incident has highlighted the possible problems associated with this industry. Follow-up investigations in other areas of New South Wales and other States of Australia have uncovered similar occurrences although on a smaller scale. A code of practice[4] dealing with this industry has been introduced in Western Australia and a national code is currently being formulated.

REFERENCES

(1) U.S. Atomic Energy Commission (1972) Grand Junction Remedial Action Criteria - Federal Register, Vol. 37 (235) - Wednesday, December 6th - pp. 25918 - 25919.

(2) The Federal Provincial Task Force on Radioactivity (1977) Criteria for Radioactive Clean-up in Canada - Information Bulletin 77-2, 1977.

(3) United Nations Scientific Committee on the Effects of Atomic Radiation (1977) - Report to the General Assembly "Sources and Effects of Ionizing Radiation". U.N. No. E.77.IX.1.

(4) Code of Practice on Radiation Protection in the Mining and Processing of Mineral Sands (1982). The Chamber of Mines of Western Australia (incorporated)

CALCULATION ON COSMIC RAY EXPOSURE RATE IN CONCRETE BUILDING

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INTRODUCTION

Information on exposure indoors is essential if we are to estimate the contribution to population dose of living environment since people spend most hours in buildings. The indoor exposure arises from terrestrial sources including building materials, directly and indirectly through the air, and from cosmic rays. Most concern has been currently concentrated on the former because large variations are expected in it. But the latter is by no means ignorable. The cosmic ray exposure depends mainly on building structures because the altitude dependence as well as "Modulation" seems to be very small in most cases (Tall buildings like the Empire State Building are only exceptions). In view of this, the variation of the cosmic ray exposure inside various types of buildings is studied here.

BASIC SCHEME OF CALCULATION

The calculation was initiated in 1978 with the aid of suggestions by Dr. Keran O'Brien and Dr. Harold L. Beck of EML, U.S.A. It was not the exposure rate but the absorbed energy by uniform building materials that was obtained by the model of continuous slowing down of cosmic rays. Water was an assumed building material in most of the early calculations as its physical parameters were well known in addition to that both stopping power and range do not greatly depend on material. However, concrete was also adopted in a few calculations. All the calculations were done in the two-dimensional framework. Muons were assumed to be the only incident particles, which seems to be a fair approximation at the ground level, if roughly. The energy-angular spectrum of the incident muons was given by the result of Luin-Code of Dr.K.O'Brien(Ref.1). The stoppimg power and the range were calculated separately following the method of Barkas and Berger (Ref.2). Energy of the incident muons ranged from 100 MeV to 44 GeV. Eyges theory(Ref.3) which was developed for accelerator's shield was applied to the scattering calculations. Cascade processes inside model buildings were not taken into account, which yet remains to be considered.

Fig.1 is an example for a "filled-in" model of water where the whole body of the building including ceiling, floor and the inbetween space is made of only water. Comparisons were made between scattering and non-scattering cases. It appeared that the scattering effect was appreciable only in the vicinity of side walls. The difference seemed to be only several per cent. Therefore authors determined to use the non-scattering model afterwards.

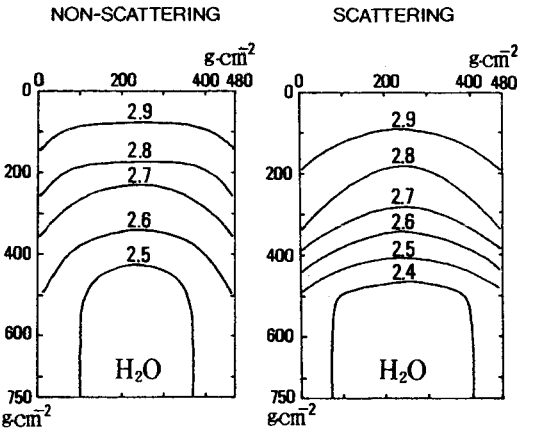

Fig.1 Contour maps of absorbed energy by water-blocks. Numerical figures of the level are in arbitrary unit. Height and width are given by culumn density where density of water is assumed to be 1.0 g/cm^3.

IMPROVED MODEL AND RESULTS

Although Fig.1 looked reasonable, the validity of the "filled-in" model was to be checked. It was not until 1982 that a new computer code was developed for stratified models, by which more realistic estimates became possible. In the new calculations, the building is no longer uniform, though still rectangular. Vacant space between each floor and ceiling is taken into consideration. The building material is a normal concrete whose density is 2.35 g/cm^3 composed of 10 elements. Assumptions made in the old calculations still hold on incident particles as well as continuous slowing down. But the energy range now covers from 10 MeV to 300 GeV. These values were determined by the trial-and-error method to adjust the results of this code to that of Luin-Code for free space.

Results are given in terms of exposure rate(μR/h), and are exemplified in Fig.2(a, b, c). Three cases of different widths are exhibited, all of which have 25 stories and have no wall. The ceiling height is commonly 2.8 m, and the thickness of floorboard 0.2 m. The contour lines are drawn based on calculated exposure rates of approximately 1400(a) and 700(b and c) points. It is seen that the lines given for each 0.24 μR/h constitute archlike figures. It is recognized that the narrower the width the higher the exposure rate on the average. It is our relief to see that the distribution pattern is fundamentally the same as that of Fig.1. Therefore it might be that the "filled-in" model is valid if only the gross pattern is wanted regardless of the absolute value.

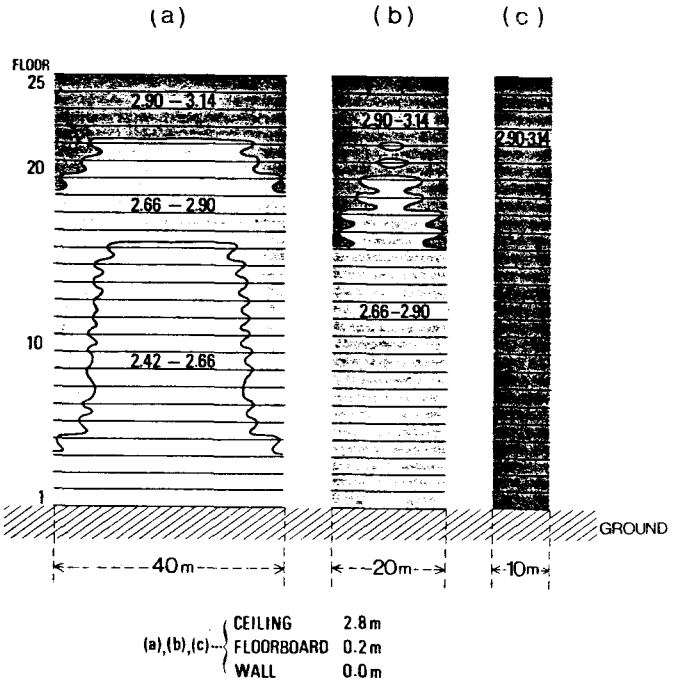

Fig.2(a,b,c) Exposure rates in stratified model buildings. Numerical figures of the level are in the unit of μR/h.

Fig.3(a, b, c) represents various modifications of the model building shown in Fig.2(a). Fig.3(a) reflects the effect of a side wall which is attached to the Fig.2(a)-building. Fig.3(b) corresponds to a thin floorboard. The thickness is exactly a half of the Fig.2(a)-case, but the remainder parameters are not changed. In Fig.3(c), only the ceiling height is lessen by about 14 per cent. Comparisons of the figures reveal that only slight reduction of the exposure rate results if either the side wall is equipped or the ceiling is changed lower. On the contrary, the effect of the thin floorboard is distinguished, where the mean level is raised significantly.

CONCLUSION

Historically, we can recognize a notable change of the building standards between the pre- and the post-World War II periods. The ceiling has become lower, and the floorboard has become thinner in the general trend that everything is getting tinner and tinner. As far as based on the non-scattering as well as non-cascade model, we can suspect that the thickness of

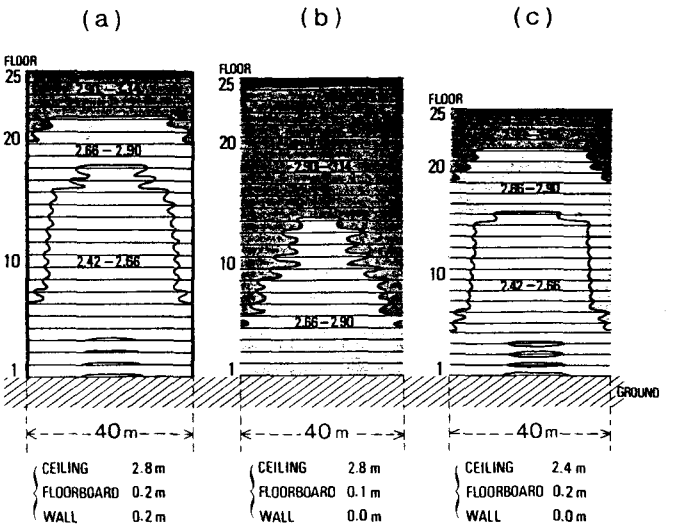

Fig.3(a,b,c) Exposure rates in three types of stratified model buildings. Numerical figures of the level are in the unit of μR/h.

floorboard is the most effective factor in determining the mean exposure rate from cosmic rays in indoor conditions. The ceiling height and the side wall are not likely to bring serious effects if extraordinary tall and slender buildings are excluded.

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REFERENCES

(1) K.O'Brien: Luin, a code for the calculation of cosmic ray propagation in the atmosphere(update of HASL-275). EML-338,Technical Information Center,U.S.DOE, 1978

(2) W.H.Barkas and M.J.Berger: Tables of energy losses and ranges of heavy charged particles,"Studies in penetration of charged particles in matter". National Academy of Science-National Research Council, Washington,DC,p108-172,1964

(3) L.Eyges: Multiple scattering with energy loss. Physical Review, Vol.74, p1534-1535, 1948

INVESTIGATIONS ON THE INCREASE OF NATURAL RADIATION EXPOSURE IN BUILDINGS DUE TO THE INHALATION OF Rn-222- AND Rn-220-DAUGHTERS.

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Abstract

Results of measurements of the Rn-222 and Rn-222- and Rn-220-daughter product concentrations in dwellings and in the open air are presented. Comparison was made between the experimentally obtained activity concentrations and the concentrations derived from model calculations using measuring data of Rn-222- and Rn-220-exhalation rates from building materials as input data. The results agreed well with the experimentally data obtained from random sampling. Dose calculations gave a range of the mean effective dose equivalent by residence in dwellings of 0.2-0.8 $mSv \cdot a^{-1}$ for Rn-222-daughters and about 0.1 $mSv \cdot a^{-1}$ by Rn-220-daughters.

Introduction

The highest contribution to the natural radiation exposure is delivered by inhalation of the short lived Rn-222- and Rn-220-daughters(1).These natural radionuclides arrive to the breathing air by exhalation of Rn-222 and Rn-220 from the soil and from the building materials. Especially in room air an enrichment of Rn-222, Rn-220 and their short lived daughters can be observed. In order to obtain a more comprehensive understanding of the mechanisms, which lead to the observed enrichment of radon and radon daughters in dwellings, we performed measurements of the activity concentrations of Rn-222, its short lived daughters and the Pb-212/Bi-212-concentrations in dwellings and in the open air. The concentrations of Rn-222 were measured by electrostatic deposition of Po-218, whereas the concentrations of the short lived daughters were determined by air sampling and alpha-spectroscopy (2). Furthermore, we performed laboratory investigations on the exhalation rates of Rn-222 and Rn-220 (3). The results were used as input data for model calculations of the Rn-222-concentrations in dwellings and were compared with the results of the experimental study from random sampling in dwellings.

Results

Figure 1 shows the results of the measurements of the radon and daughter product concentrations from random sampling in dwellings and in the open air. The values for the activity concentrations of short lived Rn-222- and Rn-220-daughters are given as equilibrium equivalent concentrations Ceq. The graphic representation shows, that the median value for the Rn-222-concentrations indoors is about a factor of 4 higher, than the corresponding value for the open air. In the case of the short lived Rn-222-daughters, this ratio amounts to about a factor of 3.

Using our measuring data, we investigated, whether the potential alpha energy concentrations of the Rn-222- and Rn-220-daughters are correlated. Figure 2 shows the relation obtained for Rn-222- and Rn-220-daughters in dwellings.

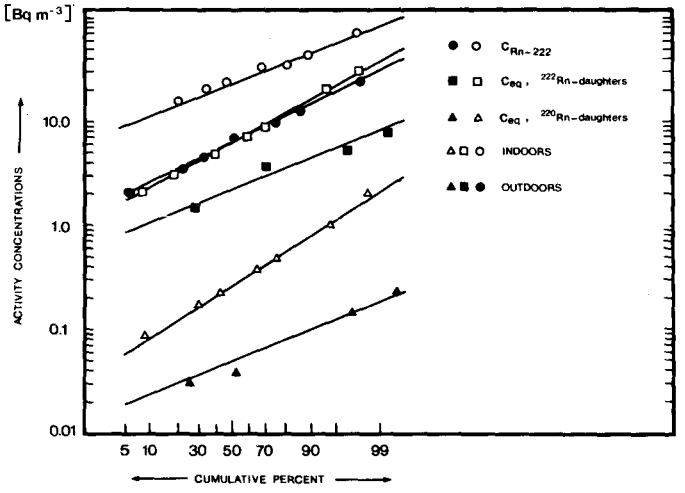

Figure 1: Cumulative frequency distribution of the Rn-222 and daughter product concentrations in dwellings and in the open air. ($Ceq\ [Bq \cdot m^{-3}] = 3.7 \cdot 10^3\ Cpot,\alpha\ [WL]$ for Rn-222-daughters and $Ceq\ [Bq \cdot m^{-3}] = 280\ Cpot,\alpha$ [WL] for Rn-220-daughters.)

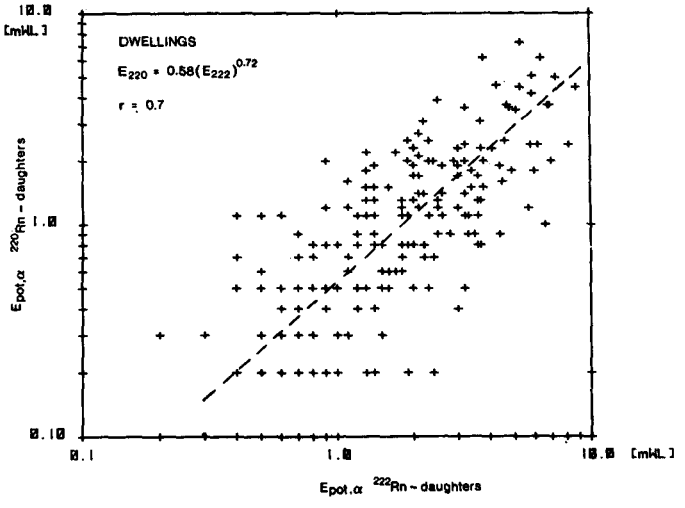

Figure 2: Correlation between Rn-220- and Rn-222-daughter product concentrations in dwellings.

The regression coefficient of r = 0.7 and the graphic representation indicates, that the Rn-220-daughter product concentrations in dwellings are correlated with the corresponding Rn-222-daughters. Moreover, our data show, that the contribution of the Rn-220-daughters to the total potential alpha energy concentration is not to be neglected compared with the Rn-222-daughters. This is a finding, similar to that of STRANDEN (4).

Another important quantity is the ratio between the equilibrium equivalent concentration Ceq of the daughters and the concentration of its parent nuclide C(Rn-222). This ratio is called the equilibrium factor F. In dwellings the median value of F was determined at $F = 0.3 \pm 0.1$ and in the open air F was $F = 0.4 \pm 0.1$. This values agree well with results from theoretical considerations (5).

In order to obtain a better understanding of the results found in dwellings by random sampling, we performed measurements of the exhalation rates of Rn-222 and Rn-220 from building materials. The results are shown in table 1.

Table 1: Rn-222- and Rn-220-exhalation rates from building materials.

Material	Φ_{Rn-222} 10^{-3} {Bq·m^{-2}·s^{-1}}			Φ_{Rn-220} 10^{-3} {Bq·m^{-2}·s^{-1}}		
	min	mean	max	min	mean	max
Pumice	0.13	0.74	1.75	27	112	565
Gypsum	0.05	0.79	5.10*	10	123	277
Lime stone	0.35	1.33	3.00	24	62	120
Heavy concrete	0.05	0.35	0.55	10	43	107
Aerated concrete	0.12	0.27	0.51	15	24	44
Slag stone	-	0.18	-	-	33	-
Brick	-	0.05	-	-	10	-
Quarry st. (Phorphyry)	-	0.57	-	-	82	-
Sandstone	-	0.26	-	-	51	-
Marble	-	0.05	-	-	10	-

If less than five different samples were measured, only the arithmetic mean is given.
* This value was measured for chemical gypsum.

As can be seen from table 1, the measured activity exhalation rates of Rn-220 are higher by about a factor of 100 than the values for Rn-222, as is expected by diffusion theory (6). Using the median values of all exhalation measurements, which were determined at $\Phi(Rn-222) = 0.5 \cdot 10^{-3}$ Bq·m^{-2}·s^{-1} and $\Phi(Rn-220) = 50 \cdot 10^{-3}$ Bq·m^{-2}·s^{-1} as input data for model calculations of the resulting indoor concentrations, one obtains for an interval of possible ventilation rates of 0.1-1.0 h^{-1} the following intervals of possible Rn-222- and Rn-220-indoor concentrations.

Rn-222: $C(Rn-222) \approx 10 - 42$ Bq·m^{-3}

Rn-220: $C(Rn-220) \approx 8$ Bq·m^{-3}

The result for Rn-222 agrees very well with the results of our own measurements in dwellings (c.f. Figure 1) and with results of investigations of a long term study, presently performed in the Federal Republic of Germany. Unfortunately there are only a little experimental data on Rn-220-concentrations in dwellings. But the few data available, seem to confirm this calculations.

Applying the dose conversion factors recommended by UNSCEAR (1), one now can estimate the probable range of the annual effective dose

equivalents due to inhalation of short lived Rn-222- and Rn-220-daughters for members of the public. From the calculated indoor concentrations an equilibrium equivalent Rn-222-concentration of Ceq = F·C(Rn-222) = 3.0-12.6 $Bq \cdot m^{-3}$ and for the Rn-220-daughters of Ceq = 0.4 $Bq \cdot m^{-3}$ results (F(Rn-220) = 0.05 (1)). The dose conversion factors D are given in Sievert per Joule inhaled potential alpha energy. D(Rn-222) = 2·$Sv \cdot J^{-1}$ and D(Rn-220) = 0.7 $Sv \cdot J^{-1}$ for the effective dose equivalent (1). Under consideration of a breathing volume of 15 $m^3 \cdot d^{-1}$ indoors, a range of the effective dose equivalent of H(Rn-222) = 0.2-0.8 $mSv \cdot a^{-1}$ and about H(Rn-220) = 0.1 $mSv \cdot a^{-1}$ by inhalation of short lived Rn-222- and Rn-220-daughters in dwellings can be expected. This values, which were derived from the results of the exhalation measurements agree well with the results obtained from the direct measurements in dwellings. Furthermore, the results show, that the contribution of the Rn-220-daughters should not be neglected when estimates of the effective dose equivalent from natural airborne radioactivity in dwellings are performed. Another important result, which can be derived from this investigations is, that the exhalation from the walls and, therefore, the activity concentration of Ra-226 and Ra-224 in the building materials seems to be the main source for indoor radon in the western part of the Federal Republic of Germany, thus causing the enhancement of the natural radiation exposure. The higher values of the activity concentrations found in cellar rooms (about a factor of two on average) could be attributed to the higher exhalation rates of the cellar walls, mainly caused by the moisture insulation outsides(6). Therefore, influences from the soil and sources like tap water or water from showers seem to be only in regions with special geological structures of significance.

References

1.) UNSCEAR'82: "Ionizing radiations; sources and biological effects", United Nations Scientific Committee on the Effects of Atomic Radiations, Report 1982, New York, United Nations Publication, Sales No. E82,IX,8 06300 p, 141-210, (1982).

2.) Keller,G., Folkerts,K.H., Muth,H.: "Activity concentrations of Rn-222, Rn-220 and their decay products in German dwellings, dose calculations and estimate of risk", Radiat. Environ. Biophys. 20, 263 274, (1982).

3.) Keller,G., Folkerts,K.H., Muth,H.: "Method for the determination of Rn-222 (Radon)- and Rn-220 (Thoron)-exhalation rates using alpha spectroscopy", Rad. Prot. Dos., Vol.3, No.1/2, 83-89, (1982).

4.) Stranden,E.: "Thoron and Radon Daughters in Different Atmospheres", Health Physics 38, 777-785, (1980).

5.) Jacobi,W.: "Activity and Potential Alpha Energy of Rn-222, Rn-220 Daughters in Different Air Atmospheres", Health Physics 22, 441-450, (1972).

6.) Folkerts,K.H.: "Theoretische und experimentelle Untersuchungen über Diffusion und Exhalation der natürlich radioaktiven Edelgase Rn-222 (Radon) und Rn-220 (Thoron) aus Baustoffen und deren Bedeutung für die Strahlenexposition in Wohnräumen", Ph.D.Thesis, Universität des Saarlandes, Saarbrücken/Homburg, (1983).

RADON CONCENTRATION LEVELS IN HOMES IN THE FEDERAL REPUBLIC OF GERMANY

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ABSTRACT

A large-scale radon survey is presently carried out in the Federal Republic of Germany using time-integrating radon dosemeters. Based on measurements in 4512 homes the distribution of the radon concentration levels fairly well exhibits a log-normal form with a median value of 42 Bq/m^3.

Significant parameters influencing the radon level indoors are the type of housing and the geological environment. Highest radon concentration levels are found in detached houses with direct ground contact.

INTRODUCTION

Natural radon in air is expected to be one of the most important contributors to the radiation exposure of the general population. Major sources of radon are the ground and - in structures - the building material. In some cases natural gas for cooking and heating and the use of radon-rich tap water may release substantial amounts of radon into indoor space. In normal, naturally ventilated houses with stone walls the radon concentration is expected to be about a factor of 2 to 5 higher than in free air near ground level.

Depending on location, geology, weather conditions, ventilation of the house and living habits of the people, the concentration of radon may change substantially with time and place (Wi83).

In order to obtain representative annual exposure data for the general population time-integrating measurements at many locations are necessary. In this paper the nation-wide radon survey program is outlined and some of the most important results are presented.

METHOD AND ORGANIZATION

For the purpose of long-term integrated measurements of radon, a purely passive radon dosemeter is employed in this study (Ur81). The instrument consists of a small diffusion chamber with a nuclear track detector (Polycarbonate). After exposure the detector foil is removed from the cup and electrochemically etched. Track counting is done visually. Calibration at standard radon atmosphere was repeated several times by different laboratories. Reproducibility, fading characteristics, mechanical stability and low price make the system advantageous for long-term large-scale measurements.

To organize the distribution and recollection of the radon dosemeters, nine laboratories in different parts of Germany are involved in the survey. In order to achieve an adequate distribution of the instruments, cooperation with the local administration of towns and counties turned out to be very effective. Different housing densities in various regions of Germany have been taken into account.

For the general survey each selected home was provided with two dosemeters, one of each was placed in the living room and one in the bedroom. After three months of exposure both dosemeters are returned to the laboratory together with a questionnaire.

Electrochemical etching and the evaluation of the radon concentration was performed by Hauptabteilung Sicherheit, Kernforschungszentrum Karlsruhe, data collection and the electronic data processing was done by the Institut für Strahlenhygiene, Neuherberg. According to our plan, more than 5 000 homes will be surveyed by the end of 1983.

RESULTS

Complete results have been obtained from 4 512 homes. The distribution of the radon levels are approximately log-normal with 90 % of the dwellings having radon concentrations of less than 80 Bq/m^3 (Fig. 1). The median value is close to 42 Bq/m^3, the geometric standard deviation 1.7.

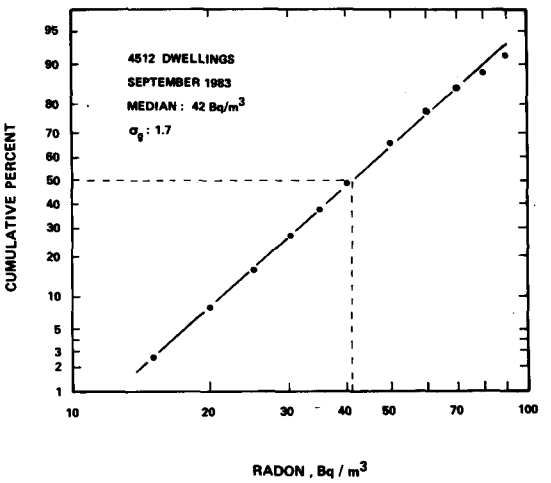

Fig. 1: Distribution of the radon concentration levels indoors

According to Fig. 2 the calculated median radon concentration is significantly higher for detached houses (45 Bq/m^3) compared with multy-family houses (36 Bq/m^3) and high-rise buildings (33 Bq/m^3). The two main probable explanations for this observation are that detached houses have lower ventilation rates and are easier exposed to radon from the ground.

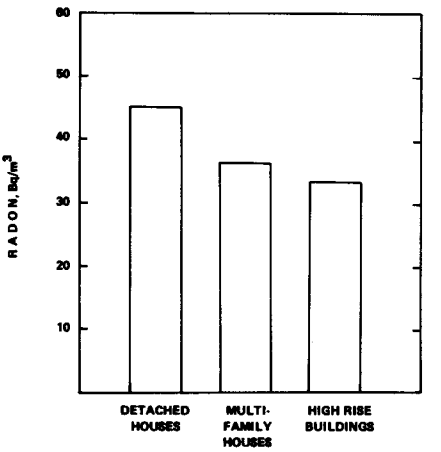

Fig. 2 : Median radon concentration levels in different types of houses

This fact is also reflected in Fig. 3. Highest radon concentrations are found in homes with direct ground contact. Apartments on upper floors show a decreasing radon level with increasing distance from the ground.

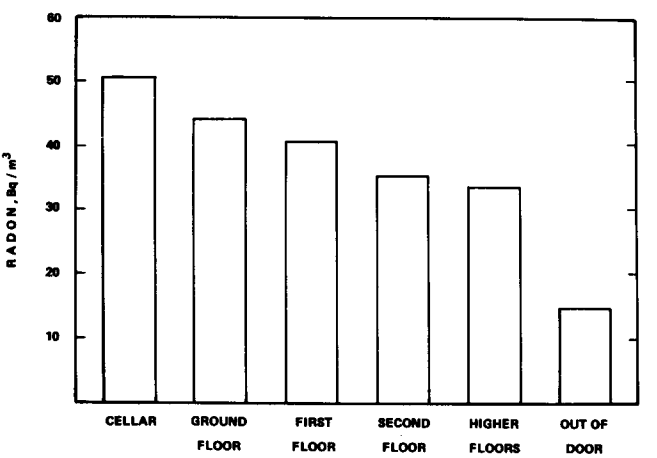

Fig. 3 : Median radon concentration levels indoors depending on different distances from the ground

The importance of different types of foundations of a building is demonstrated in Fig. 4. The most frequent type of house built in Germany has a complete basement. Such a house has a median radon concentration of 40 Bq/m^3, while houses without or partial basements show elevated levels, 45 Bq/m^3 and 51 Bq/m^3, respectively.

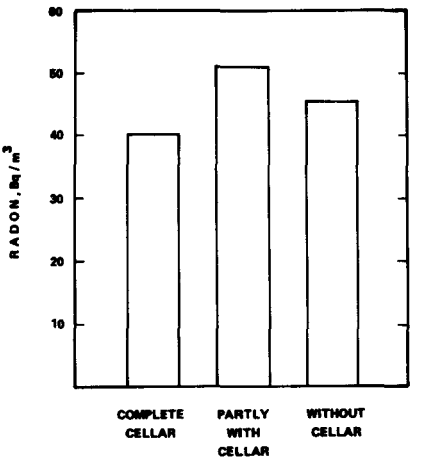

Fig. 4: Median radon concentrations in houses with different types of basements

The different types of houses might also be responsible for the fact that relatively low levels of radon are observed in big cities, such as Hamburg, Munich, Frankfurt or Berlin, in contrast to higher values present in rural areas. In addition, high radon levels are expected in houses located in certain granite areas. Those "regional" studies are still in progress.

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REFERENCES

(Ur 81) Urban, M., Piesch, E. (1981): Low level environmental radon dosimetry with a passive track etch detector device. Rad. Prot.Dosimetry 1, 97-109.

(Wi 83) Wicke, A. (1983): Exposure of the population to radon daughters - problems associated with the assessment of the annual dose. Paper presented at the INTERNATIONAL SEMINAR ON INDOOR EXPOSURE TO NATURAL RADIATION AND RELATED RISK ASSESSMENT, Capri, Italy, October 3-5, 1983.

EVALUATION OF INHALATION RISK FROM MAN-MADE AND NATURAL SOURCES OF ATMOSPHERIC RADON

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Introduction

Nuclear power generation causes concern to the public due to the ionising radiation hazard associated with it. In fact almost all types of power generation at present, except hydroelectric power generation, cause radioactive releases into the atmosphere. Man is exposed also to natural radiation ever since his origin on earth. Of all the natural radioactivities from all the important sources to which the public is exposed, radon activity is the highest. Radon and its daughters enter the human system through inhalation which is the most important pathway compared to external radiation and ingestion. This paper deals with the different sources of atmospheric radon and the inhalation risk due to breathing air containing radon and its daughter products and compares the different sources to find out the major contributor to the internal dose to which the public is exposed. It also deals to some extent with the possible effects of breathing air containing radon and its daughter products of lower concentrations.

Sources of Radon

Table 1 gives the significant sources of radon in the atmosphere and the equilibrium activity released. The sources considered for the purpose of this paper are (a) natural, (b) coal burning, (c) natural gas burning, (d) nuclear power generation and (e) nuclear weapon tests. Average values of the equilibrium activity released have been calculated assuming that the present rate of release into the atmosphere is maintained. Technological advancement might reduce or increase the activity levels in future.

Table-1. Equilibrium Activity of Radon Released into the Atmosphere from Different Sources at the Present Level

Source	Activity in Ci
Natural	3.329×10^7
Natural gas burning	9.726×10^2
Nuclear power generation	7.795×10^2
Coal burning	3.515×10^2
Nuclear weapon testing	0.396×10^2

The important natural sources of radon considered include emanations from the soil (1) and oceans (2) and radon released when ground water and sub-surface water carrying dissolved radon become warmed at the surface (2). The release of radon from volcanic erruptions is estimated to be of minor importance from a global perspective (3).

The world production rate of natural gas is at 1.278×10^{15} l/y (4). The average radon concentration in natural gas is taken to be 50 p Ci/l (5). Radon in natural gas is released into the atmosphere when the gas is used as a fuel for power production and in chemical industries (6) and in houses (5).

Assuming a load factor of 0.6, the world nuclear energy production works out to be 86.6 GW(e) in 1981 (7). Taking into consideration the radon levels released

due to operations in the uranium mines and mills and from the tailings area (8), the equilibrium activity has been found to be 7.795×10^2 Ci.

Radon released due to power generation from coal burning includes both the radon concentration in coal and the radon emission due to coal mining which is 1000 times higher than radon trapped in coal (9).

Between 1972 and 1981, the natural uranium supply for nuclear weapon production is estimated to be 864 tons per annum (9). This corresponds to a release of radon whose equilibrium activity is 0.396×10^2 Ci. Estimation of the activity released takes into consideration the uranium mining and milling operations and emanations from tailings area (8).

Risk Evaluation

Inhalation of radon alone does not result in any appreciable dose to the lung. Normally radon can be expected to be in equilibrium with its short-lived daughters, Ra-A, Ra-B, Ra-C and Ra-C' which contribute to the radiation dose delivered to the T-B region of the lung. Assessment of dose to this region of the lung from deposition of radon daughters present in the atmosphere depends on their attachment to aerosol particles and the nature of these particles. The different parameters involved in the dose assessment for a uranium mine atmosphere are described in detail elsewhere (10). The mean particle size and the hygroscopicity of the particle which are the most important criteria deciding the activity to dose conversion factor in the assessment of dose are the same for the free atmosphere also. The unattached fraction in the free atmosphere works out to be 5% since the average aerosol concentration is assumed to be 1.5×10^4 per cm^3(11). On this basis, the activity to dose conversion factor for breathing of atmospheric air containing radon and its decay products is estimated to be 41 μrad per pCi hr per litre. This factor is used to calculate the dose rate due to radon released from different sources described earlier and to compare them.

Risk Comparison

Assuming the half-depth for radon concentration in the atmospheric air to be 1 km (12), the surface volume of the atmosphere for obtaining the surface concentration of atmospheric radon from different sources is found to be 5.1×10^{17} m^3. Then the average radon concentrations breathed by the population over the globe due to releases from natural emanation, natural gas burning, nuclear power generation, coal burning and nuclear weapon production work out to be 65.3, 190.7×10^{-5}, 152.8×10^{-5}, 69.0×10^{-5} and 77.7×10^{-6} in pCi/m^3 respectively. The corresponding dose rates due to inhalation of radon daughter products are 2.6, 7.8×10^{-5}, 6.3×10^{-5}, 2.8×10^{-5} and 3.2×10^{-6} in μr/hr. At the present level of releases of radon from man-made sources, the inhalation dose rate to the T-B region of the lung, in all, amounts to about 0.007% of the corresponding dose rate from natural sources.

Conclusions

The present study of the estimation of dose rates to the T-B region of the lung from radon and its daughters released to the atmosphere from different sources shows that the contribution from man-made sources is negligibly small compared to the inhalation dose received by man from natural sources to which he is subjected for a very long time.

Next to the release from natural gas burning, the release from nuclear power generation is the highest due to the use of natural and enriched uranium as fuel in the reactors. The fact that the total installed capacity in MW(e) from breeder reactors can grow to many hundreds of thousands (13), shows that the dose equivalent from radon release can be made negligibly small in the case of nuclear power generation.

We feel the paper will not be complete without a remark on the effects of low-level radiations on man. The theories and experimental results of the beneficial and harmful effects of ionising radiations are quite controversial and are not clear at present. Any argument leading to either of the effects is not correct without taking the synergestic effects of ionising radiation and chemical agents. There appears to be sufficient reason to think that ionising radiation alone may have beneficial effects but in the presence of certain chemical agents, the effect may be harmful. This aspect needs a rigorous study.

References

(1) UNSCEAR, Ionising Radiation: Sources and Biological Effects, Annex D 141 (1982)
(2) J.H. Harley, Noble Gases Symp., Las Vegas, Sept. 24 - 28 (1973)
(3) G. Lambert, A. Buisson, J. Sanak and B. Ardovin, J.G.R., 84(c), 6980 (1979)
(4) World Statistics in Brief, United Nations, New York (1976)
(5) R.H. Johnson, Jr., D.E. Bernhardt, N.S. Nelson and H.W. Calley, Jr., EPA-520/1-73-004 (1973)
(6) M.C. Subba Ramu, T.S. Muraleedharan and K.G. Vohra, in Natural Radiation Environment (Ed. K.G. Vohra et al.) Wiley Eastern Ltd. New Delhi (1982)
(7) Power Reactors in Member States, IAEA, Vienna (1981)
(8) UNSCEAR, Exposures Resulting from Nuclear Power Production, Annex F, A/AC.82/R-400, 80 (1981)
(9) Z. Jaworowski, IAEA Bull., 24(2), 35 (1982)
(10) K.G. Vohra and M.C. Subba Ramu, in Current Concepts in Lung Dosimetry (Ed. D.R. Fisher), U.S. Dept. Energy, 139 (1983)
(11) C.E. Junge, Air Chemistry and Radioactivity, Acad. Press, New York, 111 (1963)
(12) W. Jacobi, Biophysik, 1, 175 (1963)
(13) R. Ramanna,Nuclear India, 16(6), 7 (1978)

METHODOLOGY AND RESULTS OF A NATURAL RADIOACTIVITY ASSESSMENT IN A REGION OF CENTRAL ITALY

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INTRODUCTION

The natural radioactivity survey being conducted in Umbria (800 000 inhabitants) is designed not only to evaluate mean doses associated with indoor exposure but above all to identify those areas and building features with the highest radiation exposure risk factor. Consequently,the sampling of houses must be particularly exact and the experimental measuring techniques particularly reliable at low exposure levels, too.

The study began by identifying five main geological areas within the region. A town which would be representative from an architectural point of view was then selected from within each area, the building materials used therein were analyzed (also by gamma spectrometry) and building material origins were identified. A certain number of houses built with these materials were then selected.Indoor and outdoor radiation measurements were taken with an ionization chamber and suitable data cards were compiled including information on the houses, inhabitants and respective home-utilization patterns. A gamma radiation exposure measurement campaign was conducted quarterly over an entire year using thermoluminescence detectors whereas a similar campaign to monitor radon with passive diffusion detectors was also begun and conducted according to the same schedule.

METHODS FOR MEASURING EXTERNAL EXPOSURE AND RESULTS

In order to evaluate the exposure levels expected to be found in the houses, spectrometry measurements of building material specific activity were taken with a GeLi detector, after it had been suitably calibrated with radiation sources with energies ranging from 88 to 1836 keV in various-density matrices (from 1000 kg/m^3 to 1333 kg/m^3). Analysis of the 35 samples enabled the range of building material specific activity levels to be identified. Table 1 summarizes measurement results. It should be noted that the specific activity values reported are generally higher for radionuclides of the Th-232 families than for those of the U-238 families. A rough computer program was designed for planning the passive gamma exposure measurements. This program, starting from the spectrometry results,calculates expected exposure. Calculation hypotheses were fairly simple, i.e., that a source be formed by two walls of the materials identified, a wall in hollow brick,the ceiling and the floor in hollow brick and cement components, the missing wall being equated to the openings due to windows and doors.Furthermore, each surface was approximated to an infinite half space , in which radionuclides are distributed uniformly and in secular equilibrium. Air attenuation was assumed to be negligible and the build-up factor in air less than 5% for the energy range studied.

Table 1 - Specific Activity Range for Building Materials

Material	Specific Activity ($Bq \cdot kg^{-1}$)		
	K-40	Bi-214	Pb-212
TUFF	(1468±48)÷(1952±50)	(136±8)÷(243±8)	(468±8)÷(541±9)
CEMENT PZ	(607±24)÷ (667±26)	(72±4)÷ (81±4)	(164±4)÷(172±4)
CEMENT PTL	(253±17)÷ (270±18)	(17±2)	(27±2)÷ (29±2)
GRAVEL	(100±13)÷ (134±12)	(11±2)÷ (14±2)	(14±2)÷ (16±2)
SAND	(393±19)÷ (491±18)	(8±2)÷ (11±2)	(13±2)÷ (20±2)
BRICKS	(550±21)÷ (883±31)	(29±3)÷ (81±4)	(49±3)÷(148±4)
POZZOLANA	1888±46	238±8	481±8

The characteristic of LiF, CaF_2:Dy and MgB_4O_7 thermoluminescence detectors were then studied. The minimum exposure detectable was seen to vary from 0.1 mR to 5.0 mR depending on the type of detector used. Linearity, for X and γ energy from 27 keV to 1250 keV, was checked in the exposure range between a few mR and 3 R for LiF and CaF_2:Dy. The coefficient of correlation was seen to be good. Reproducibility, within the exposure range studied, was seen to be ± 4% for commercial TL dosemeters (TLD-100, TLD-200 and TLD-700) but considerably less for MgB_4O_7 dosemeters. Fading was studied in TLD-100s and TLD-200s suitably shielded for different periods of time and at mean ambient temperatures of 20 °C and 30 °C and mean relative humidities of 65% and 50%, respectively. Suitable statistic check tests showed no significant fading in LiF detectors over the entire period considered whereas CaF_2:Dy detectors began to show significant variations (20%) from initial values after 30 days. Lastly, detector energy dependence was studied with different types of packaging.

To avoid the necessity of coupling a filter system to detectors, with the consequential anisotropy and related calibration difficulties, it was finally decided to select the least energy-dependent detector, i.e., the LiF (TLD-100) detector packaged in a black paper envelope 67 mg/cm^2 thick. Differences between ambient temperatures inside and outside the package were evaluated between 26 °C and 51 °C with a digital thermometer having copper and constantan probes. The maximum difference was found to be (1.6 ± 0.1) °C. Beta contribution to exposure, considering source spectrum and package characteristics, was seen to be approximately 2% of the total.

Dosemeters thus selected were paired and placed both in one bedroom and in the most lived-in room of each home selected. Table 2 reports the mean exposure rates measured over the year in some sampling homes grouped according to building materials, without subtracting cosmic ray contribution (that has been measured at the centre of Lake Trasimeno).

Table 2 - Indoor Exposure Rates in Three Different Towns

Towns	Building Material	Exposure Rates ($\mu R \cdot h^{-1}$)
Foligno	Stone	10.7 ± 1.1
	Hollow bricks and Cement	20.9 ± 2.1
	Red Tuff with Black Scoriae	30.1 ± 3.1
Orvieto	Local Tuff and Pozzolana	61.3 ± 6.1
	Local Tuff and Cement	55.3 ± 5.5
	Local Tuff	60.0 ± 6.0
Todi	Stone	14.3 ± 1.4
	Stone and Red Tuff with Black Scoriae	18.1 ± 1.8

METHODS FOR MEASURING INTERNAL EXPOSURE AND RESULTS

In order to evaluate internal exposure for inhabitants, radon and radon daughter concentrations must be measured in the houses. This may be done with integrated and/or prompt measurements taken with an active or passive device. Integrated measurements are essential to assess the mean concentrations because of the well-known radon dependence on ventilation and atmospheric conditions. Conversely, prompt measurements give an idea of radon and radon daughter concentration variations.

As integrating and passive devices, Solid State Nuclear Track Detectors were chosen for their practicality, economy, and specific qualities: sensitivity to alpha particles without being infuenced by gamma radiation or electrons, and for their nonfading and stability of recorded information even after reading [1].

Initially, considerable laboratory-based experimental work was done with CR-39 and LR-115(Kodak-Pathé). For the indoor survey, the first type was preferred because it has no energy threshold. After some tests, CR-39 about 600 μm thick from American Acrylics was chosen. Etching conditions were optimized paying particular attention to etching times, weight losses and thickness reductions being studied with both NaOH (at 70 °C) and KOH (at 60 °C) at various times. The resultant data were compared with the theoretical curves of alpha penetration in CR-39 with varying energy.

Particular CR-39 diffusion dosemeters were designed and built in the laboratory after an analysis of specific literature and some preliminary experiments performed fastening small cups containing CR-39 to some walls of the laboratory rooms. At this stage, tracks are still being counted with an optical microscope, however, automation of this step has also been studied [2].

To test the detection method, some diffusion dosemeters were distributed in some of the same survey homes in the rooms with TLDs. Preliminary calibration of these diffusion dosemeters was performed with a radon chamber at the ENEA Centre of Casaccia. Table 3 shows the mean radon concentrations for some types of building materials

Table 3 - Mean Radon Concentration in Homes Built with Different Materials

Building Material	Mean radon concentration ($Bq \cdot m^{-3}$)
Stone	46 ± 12
Tuff	73 ± 19
Stone + Tuff	70 ± 29
Tuff + Cement	90 ± 30

for the spring-summer survey (90 days). Background has been subtracted. Final calibration at the National Radiological Protection Board Laboratories is planned for the near future.

Lastly, the Harshaw Radon Daughter Analyzer was tested as a prompt and active device in some selected houses, under different ventilation conditions, and within this framework satisfactory agreement was found with passive detector results. Systematic measurements with this equipment would provide estimates of radon and radon daughter concentrations in terms of Working Levels thereby enabling internal exposure variations connected with season, ventilation, atmospheric and living conditions to be evaluated.

CONCLUSIONS

It is most certainly too early to draw any definite conclusions from the measurements taken so far. However, a correlation between building materials used and indoor external exposure levels does seem feasible at this stage.

Annual doses of radiation absorbed by the inhabitants living in tuff rock homes in southern Umbria are in the region of 4.2 mGy, cosmic radiation included, assuming a home occupancy factor of 0.8. Adding to this value the mean individual absorbed dose rate of approximately 0.7 mGy/y due to outdoor gamma exposure from cosmic rays and terrestrial radiation in the same area, provides[3] an annual effective dose equivalent from gamma radiation of approximately 3.4 mSv.

Conversely, preliminary results concerning internal exposure based on only one three-month campaign and on the measurements of radon concentration alone, do not enable forecasts of expectable internal radiation doses to be made.

REFERENCES

1) i.e.,see Proceedings of the 11th International Conference on Solid State Nuclear Track Detectors. Bristol (U.K.), September 1981.

2) See G.Campos Venuti et al. in Proceedings of the 12th International Conference on Solid State Nuclear Track Detectors,Acapulco(Mexico). September 1983.

3) Ionizing Radiations: Sources and Biological Effects. United Nations Scientific Committee, 1982 Report.

ASSESSMENT OF NATURAL RADIATION EXPOSURE RATE IN KOREA +

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INTRODUCTION

The measurement of natural environmental radiation exposure rate in Korea was initiated in 1961, and has been routinely carried out since then over twenty years in the present Korea Advanced Energy Research Institute (KAERI) as a part of environmental radiation monitoring program for north-eastern part of Seoul area where the institute is located. Similar surveys were also made around Kori nuclear power plant on south-eastern coast of Korea in two separate years, 1978 and 1982, respectively, by the same group. Major part of the measurement was made using portable G-M counting system.

Very recently another type of measurement of natural radiation exposure rate was carried out by a group of the Department of Physics, Chungnam National University(CNU) for Daejeon area, which is a city in mid-western part of south Korea, by means of gamma-ray scintillation spectrometry. The areas mentioned are shown in Fig. 1.

In this study a series of assessment is made to figure out the absorbed dose rate in air, corresponding organ dose, and collective dose due mainly to the external exposure of terrestrial component of natural environmental radiation based on the analysis of KAERI and CNU data obtained for those three areas.

THE MEASURED EXPOSURE RATE

1. Measured by KAERI group

For the assessment of exposure rate measured in Seoul area, the data obtained off site of KAERI for recent five years from 1977 to 1981 were statistically analyzed, for which the influence of fission products originated from atmospheric nuclear explosions conducted early 1960s are negligible. The measurement in this area has been carried out at 5 sampling sites distributed in the region of 1 to 12 km in radius from the KAERI reactor building.(1,2) The measurement was done on monthly basis using unshielded end-window type portable G-M counting system calibrated periodically with a reference source of ^{226}Ra of known activity encapsulated in 0.5 mm platinum capsule for converting count rate into exposure rate. Thus determined latest conversion factor is 0.512 uR/h.cpm.

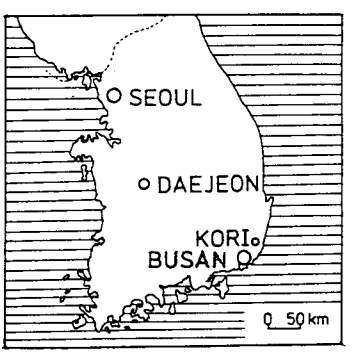

Fig. 1. Areas where Natural Radiation Exposure Rates were Measured

+ Part of the work sponsored by the Korea Science Foundation

The exposure rate at each point was measured at a height of 1 m from the earth with the detector window facing the ground surface.[1-4] The results were summarized in Fig. 2 as a variation of monthly average of the given period of time, with yearly average of 16.1 ± 0.8 uR/h. The pattern of the monthly variation of natural radiation exposure rate shown in the figure is quite similar with that observed by Minato[5] in Nagoya, Japan, for four years from 1977 to 1980, which shows strong dependence of terrestrial radiation exposure on the dryness of surface soil.

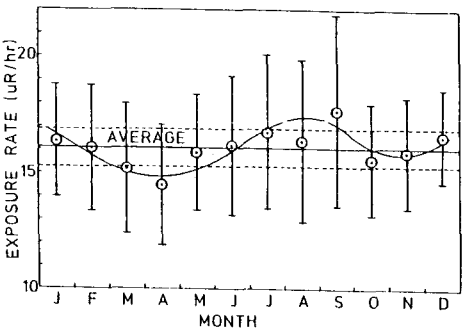

Fig. 2. Monthly Variation of Natural Radiation Exposure Rate in Seoul Area (1977 - 1981) measured by G-M counting system

In the same way the natural radiation exposure rate in Kori area were measured in the region covering 1 to 30 km in radius from Kori unit 1 nuclear power reactor. In 1978 the measurement was made at 12 different points distributed in the region[2], while it was done at 23 points in 1982.[6] The measurement was made quarterly at each point, and the results[2,6] obtained are summarized in Fig. 3, with yearly average of 16,3 ± 0.8 uR/h.

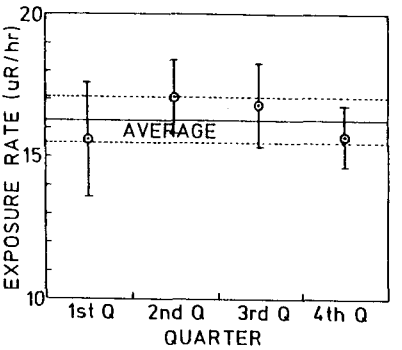

Fig.3. Quarterly Variation of Natural Radiation Exposure Rate in Kori Area (1978 and 1982) measured by G-M counting system

2. Measured by CNU group

A series of in-situ gamma-ray spectrometric investigation on the natural radiation exposure has been carried out in Daejeon area using 3"ϕ x 3" cylindrical NaI(Tl) scintillation detector in association with a 400 channel pulse height analyzer. It was carried out at 8 different points distributed in an area of about 200 km^2. During the coarse of the measurement the detector was kept at a height of 1 m from the ground with the detector axis vertical to the earth surface.

The measured gamma-ray spectra were directly converted into exposure rate by using Moriuchi's spectrum-exposure rate conversion operator, G(E),[7,8] which was derived from the assumption of linear proportionality between the output pulse height of the detector and the energy of secondary electrons generated by incident photons in the detector, and the existence of an integral equation

$$D(E_o) = \int_o^\infty f(E, E_o)\ G(E)\ dE \qquad (1)$$

where $f(E, E_o)$ is pulse height distribution of the secondary electron of energy E due to the incident photon of energy E_o of unit flux density, and $D(E_o)$ is the exposure rate in the identical gamma-ray field. The conversion operator, G(E), is a function of incident photon energy and the size of

scintillation crystal, and the numerical vulues for commonly used crystals are readily available.

The standard error involved in the exposure rate evaluated by means of G(E) operation is a function of total counts under the pulse height spectrum. It is estimated to be less than 0.4 % for 3"ϕ x 3" NaI(Tl) detector at 10 uR/h level[9].

In order to figure out the terrestrial component of exposure rate($\dot{X}$), it is necessary to correct the exposure rate calculated from the spectra($\dot{X}_s$) for cosmic ray contribution, the contribution of ^{40}K gamma-rays in PM tube glass, and directional dependence of the detector on the incident photons. This can be made by

$$\dot{X} = 0.94(\dot{X}_s - 0.25) \quad \text{uR/h} \qquad (2)$$

according to an evaluation for presently available 3"ϕ x 3" NaI(Tl) detector.

Table 1. Terrestrial Radiation Exposure Rate in Daejeon Area+

Location No.	$\dot{X}$ (uR/h)
1	10.30
2	10.34
3	9.00
4	10.25
5	6.38
6	7.25
7	8.42
8	9.90
Average	8.98 $\pm$ 1.52

+ Daytime, March 1983

The results thus obtained are summarized in Table 1. Reproducibility of this type of measurement was within 0.1 to 1 %.

ASSESSMENT OF THE EXPOSURE RATE AND DOSE

1. Assessment of the Exposure Rate

In order to assure the exposure rate measured by means of gamma-ray spectrometry it was compared with those measured by the Reuter/Stocks' pressurized ionization chamber, which was calibrated with ^{226}Ra and ^{60}Co standard sources, by carrying out a series of simultaneous measurement made at the same place. The terrestrial radiation exposure rates measured by the spectrometry and ion chamber in this series of experiment were 10.61 $\pm$ 0.44 uR/h and 9.95 $\pm$ 0.19 uR/h, respectively, which are in satisfactory agreement within statistical error considering that the response of this particular ion chamber to the photons of below 50 keV is zero, while that of the NaI(Tl) detector down to photons of 30 keV is still existing. The numerical value of 9.95 uR/h resulted from the elimination of cosmic ray contribution to the ionization chamber which is known to be 3.59 uR/h at normal atmospheric pressure.

For the assessment of exposure rate measured by portable G-M counting system, similar comparative measurement was again made with the gamma-ray spectrometry by carrying out ten times simultaneous measurement at the same place. According to the result of the assessment the expesure rate measured by the G-M system is higher as much as 8.4 $\pm$ 0.7 uR/h than those measured by the gamma-ray spectrometry. This discrepancy is understandable in comparison with the value of 6.9 uR/h determined by Moriuchi for a G-M counter as a result of cosmic ray contribution and self contamination due to such materials as ^{40}K and other radioisotopes in the detector tube[9] when we consider the standard deviation in exposure rate measured individually by a G-M survey meter reaching about 30 % at background level[10]. Taking all these factors

into account, the yearly average of exposure rates due to the terrestrial component of environmental radiation in Seoul and Kori areas are estimated to be 7.7 ± 0.4 uR/h and 7.9 ± 0.3 uR/h, respectively.

2. Assessment of Dose

On the basis of thus estimated numerical values of the terrestrial radiation exposure rate ($\dot{X}$) the absorbed dose rate in air ($\dot{D}_a$) in "receptor free" condition is easily calculated by applying the conversion factor, 0.869 rad/R.

From $\dot{D}_a$ organ dose rate ($\dot{D}_o$) is calculated by

$$\dot{D}_o = sg\dot{D}_a \qquad (3)$$

where s is the ratio of mass energy absorption coefficient for tissue and air, and g is geometrical factor. $sg = 0.82$ is used to calculate gonad dose rate for external outdoor exposure ($\dot{D}_g$), and 0.7 is used for average absorbed dose rate in the body ($\dot{D}_b$)[11]. $\dot{D}_b$ is also calculated directly from $\dot{X}$ applying conversion factor, 0.6 rad/R[12].

Annual gonad dose (D_g) and annual body dose (D_b) are calculated by time integration of $\dot{D}_g$ and $\dot{D}_b$, respectively. Successive calculation of the annual collective dose (S_y) in the areas of the exposure rate measurement can be carried out using presently available population data. The results of the assessment are summarized in Table 2.

Table 2. Exposure Rate and Dose from Terrestrial Radiation in Korea

Area	$\dot{X}$(uR/h)	$\dot{D}_a$(urad/h)	$\dot{D}_b$(urad/h)	D_g(mrad)	D_b(mrad)	S_y(man-rad)
Seoul	7.7	6.7	4.6	33.1	40.3	3.50×10^5
Kori	7.9	6.9	4.7	33.8	41.2	1.67×10^5
Daejeon	8.98	7.9	5.4	38.9	47.3	3.67×10^4

CONCLUSION

With the results obtained through this study it is concluded that as far as the areas mentioned are concerned, Korea is one of the "normal" areas in the world in radiological point of view suggested by UNSCEAR. According to the 1977 report of the Committee[11] annual gonad dose from external terrestrial radiation in "normal" area is, on average, 32 mrad, and 95% of the world population would receive annual gonad doses in the range of 21 to 43 mrad. Korea belongs to this category.

REFERENCES

1. J.S.Jun et al. : KAERI/110/RR-35/78 (1978)
2. H.D.Lee et al. : KAERI/222/RR-91/39 (1979)
3. C. Rho : J. of the Academy, ROK, vol.20, pp.19-80 (1981)
4. H.D.Lee et al. : KAERI/RR-316/81 (1982)
5. S. Minato : J. Japan Atom. Energ. Soc. 25, 637 (1983)
6. H.D.Lee et al. : in KAERI/RR-386/82(J.H.Lee ed.) (I) (1983)
7. S.Moriuchi and I.Miyanaga : Health Phys. 12, 541 (1966)
8. S. Moriuchi : JAERI-1209 (1971)
9. S. Moriuchi : JAERI-M 7066 (1977)
10. H. Morishima et al. : Ann. Rept. Nucl. Res. Inst. Kinki Univ.13, 1 (1976)
11. UNSCEAR : Sources and Effects of Ionizing Radiation (1977)
12. K. O'Brien and R. Sanna : Health Phys. 30, 71 (1976)

TECHNOLOGICALLY ENHANCED RADIATION EXPOSURE OF POPULATION DUE TO RADIUM 226 IN WASTE WATER

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In the nuclear fuel cycle radium is of major concern for radiation exposure of population by mining and milling of uranium ores. In the field of technologically enhanced radiation exposure, coal burning power plants have been identified as sources for airborne emissions of radium and other natural radionuclides.

Data for radium emissions in waste water from technological sources have been reported only occasionally.
In the Federal Republic of Germany, a comprehensive research project was carried out in order to evaluate relevant radium 226 emissions from industrial and mining activities. In most cases samples were taken continuously and radium 226 was measured by emanation techniques. The yearly emissions extrapolated from the samples ranged from less than 1 mCi to almost 1 Ci.

Radioecological investigations were carried out along the river Lippe which receives most of the pit water from coal mines, in order to evaluate the resulting radiation exposure of population. Besides fish consumption the soil - grass - cow - milk pathway and external irradiation on areas that are occasionally flooded were identified as relevant pathways. In these areas radium concentrations of up to 10 pCi/g were measured in soil. The transfer of radium to grass and milk is lower than known from studies in areas of enhanced natural activity. Milk concentrations are in the range of 0.1 to 1 pCi/l. External radiation reaches in some spots 15 μR/h and more, about 3 times the normal background radiation.

The resulting doses for members of the public are less then the doses that are calculated by radioecological models used in environmental impact assessments for nuclear facilities.

ENHANCED RADIOACTIVITY DUE TO NATURAL OIL AND GAS PRODUCTION

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INTRODUCTION

For more than 60 years it has been known that thermal brines can be enriched in natural radioactive substances. Such brines are raised to the surface together with natural oil and gas as an unwanted by-product. Already in 1927, A. Tcherepennikov (1) measured a radium concentration of $7.4 \cdot 10^{-10}$ % (250 Bq/l) in a brine sample taken from a well of the oil field Uchta in North East Russia. Since that time many papers on the geochemistry of radionuclides in oil and gas fields have been published mainly by Soviet scientists (2). First results for brines from German oil fields (near Hanover) were reported by H.J.Born (3) and ranged from less than 1 to 13 Bq/l. These brines consist mainly of sodium chloride, sometimes in saturated solution and with varying concentrations of alkaline earths. In gas fields excess salt from oversaturated brines is often precipitated after expansion, and scale is formed at the inner walls of tubings, pumps, separation and storage tanks. In many cases precipitates also result from the mixing of brines with surface waters (4). The equipment must then be exchanged from time to time for cleaning purposes.

Moreover, natural gas contains radon in various concentrations (5). Thus workers and consumers are likely to be exposed to enhanced natural radiation. Within the scope of a research contract (Federal Ministry of the Interior, St.Sch 872), a survey program was established for

- dose rate measurements at various production sites,
- assessment of the radionuclide content of brines and scales,
- assessment of the Rn-222 content in natural gas and
- aerosol measurements during cleaning work.

In this paper first results are presented and discussed.

INSTRUMENTS AND METHODS

Two scintillation dosimeters - H 7201 and PTB 7906 - have been used for dose rate measurements below and above 3 µSv/h.

Rn-222 was measured after transfer of the gas samples into 500 ml scintillation chambers (background count rate 0.5 min^{-1}, efficiency 68 %, detection limit 4 mBq/l at a counting time of 1000 min and a standard deviation of 10 %). Ra-226 in brines was measured according to the emanation method. Its decay product Rn-222 was emanated either directly from a brine sample or from the EDTA solution of the $BaSO_4$ precipitation into a 100 ml scintillation chamber (background counting rate 0.05 min^{-1}, Ra-226 efficiency 65 %, detection limit under the same conditions as above 1 mBq). The scintillation counter (BCIF China, type FD 125) was connected with an amplifier, discriminator, scaler and printer and calibrated with a PTB Ra-226 standard solution. A detailed description is given elsewhere (6). Scale samples were analysed for Ra-226, Ra-228, Th-228 and some other nuclides in a well-type Ge(Li) spectrometer.

Brines with higher activity concentrations were also measured with a Ge(Li) spectrometer using 400 ml Marinelli bakers. The results for Ra-226 measured according to the two methods agreed fairly well.

RESULTS

External Radiation

77 % of 83 surveyed well head sites did not show an increase in radiation above the natural background. Dose rates between 0.1 and 1 µSv/h were observed at 17 % of the sites investigated. At five sites, however, the dose rates were above 1 and up to 50 µSv/h. The highest dose rates were measured at the external surface of storage tanks for brines and were mainly caused by gamma radiation from Ra-226 contained in celestobaryte scales inside the tanks. In some cases slightly enhanced radiation observed on gas/oil separators was caused by Ra-222 and its daughter products contained in condensates. Under normal conditions the dose rates at working places were in all cases below the lower limit of a controlled area (7.5 µSv/h).

Radium Content of Brines

All brine samples analysed by gamma spectroscopy contained mainly Ra-226 and Ra-228. For 84 samples analysed for Ra-226 a variation range of more than 4 orders of magnitude was found. For 10 % of the samples the activity concentration was less than 4 mBq/l but another 10 % exceeded 20 Bq/l. The median value was 1.5 Bq/l. Ra-226 concentrations of 80 Bq/l were found in brine samples from several gas fields close to the Dutch border.

Specific Activity of Scales

Ra-226 and Ra-228 are also the dominating radionuclides in scales and other precipitates. In addition, Th-228 (depending on the age of the precipitate) and sometimes significant amounts of Pb-210 (in samples from East Hanover gas fields) were found. Specific activities up to 1 kBq/g were found for samples of $BaSO_4$ scale. The specific activity of $CaCO_3$ scales is usually much lower but such scales may contain Ac-227 as the dominating radionuclide. The activity ratio Ra-228/Ra-226 was found to be higher than 1 in samples from oil fields and lower than 1 in most of the samples from gas fields (variation range 0.056 to 4.4). Some results of gamma spectrometric analyses are presented in Table 1.

Table 1:

Specific activities in Bq/g of some scales and other precipitates

Nuclide	Sample No.					
	114/10426	126/10983	33/11070	33/10972	190/BP1	190/BP2
Ra-226	59	350	160	7.4	1000	0.85
Pb-210	-	-	30	70	22	1.4
Ac-227	n.d.	n.d.	n.d.	n.d.	n.d.	2.5
Ra-228	240	7.4	120	5.9	<10	n.d.
Th-228	48	n.d.	26	1.6	<10	n.d.
$\frac{Ra-228}{Ra-226}$	4.1	0.02	0.75	0.8	< 0.01	-

n.d. = not detectable
- = not analysed

Sample description:

No.	Origin	physical form	chemical form
114/10426	oil field	scale	$Ba/SrSO_4$, PbS
126/10983	gas field	scale	$Ba/SrSO_4$
33/11070	gas field	scale	Pb, $Ba/SrSO_4$
33/10972	gas field	deposition	SiO_2, PbS, Hg
190/BP1	gas field	scale	-
190/BP2	gas field	scale	$CaCO_3$

Rn-222 Content of Natural Gas

140 gas samples mainly from North German gas and oil fields were measured until summer 1983. The radon concentrations are log-normally distributed and range from less than 0.1 Bq/l (15 % of the samples) up to 4 Bq/l with a median value of 0.30 Bq/l. 15 % of the samples exceed 1 Bq/l. Values in the upper range were found in gas samples from Permian deposits near the Dutch border and from lower new red sand-stone deposits in the Ems-Weser area. In general a relationship between radon and other gas constituents could not be ascertained. 30 samples from the Emsland gas field, however, showed a weak positive correlation ($r^2 = 0.45$) between Rn-222 and n-butane.

In most cases the gas samples were taken near the well heads. The average Rn-222 concentration in gas samples taken during half a year from the distribution line in Brunswick was only 0.11 Bq/l (variation range from 0.06 to 0.15 Bq/l).

DISCUSSION AND CONCLUSIONS

It is shown that radioactive scales in tubings, pumps and storage tanks can be detected with common radiation protection dosimeters. Under normal conditions of operation the annual dose for workers due to external radiation from such sources will not, however, exceed 5 mSv.

According to Article 4, of the Euratom Directive (7), notification and prior approval are not required for radioactive materials with a specific activity of less than 100 Bq/g. For solid natural radioactive materials this exemption limit is even increased to 500 Bq/g. About 20 % of the investigated scale and precipitate samples exceeded

this value. It was common practice to remove the scale from the inner walls of the gas tubes by sand-blasting until results of aerosol measurements were submitted (PTB Report 6.62-862/1, not for publication): The derived activity concentrations (DAC) according to ICRP 30 (8) may be exceeded by one order of magnitude, even if the specific activity of the scale to be removed is less than the exemption limit of 500 Bq/g.

In the Federal Republic of Germany, the exemption limits mentioned above are not applicable to waste disposal. According to a very restrictive legal requirement, waste containing highly toxic radionuclides must be considered as radioactive if the specific activity exceeds 0.37 Bq/g, and normal disposal of such waste requires prior approval. As a way out of this legal problem, scales and other precipitates from oil and gas production may be considered as by-product materials which could be used, for instance, together with concrete to refill old wells.

Up to now the large quantities of brines which are raised to the surface together with oil and gas have not raised any problems. These brines are re-used for the extraction of oil from its deposits.

Sewage sludge from oil processing plants is used in agriculture as a soil improver. This is permitted without any limitation of the specific activity as far as radionuclides of natural origin are concerned. For other nuclides the limit is as low as 0.37 Bq/g.

The radiation exposure of the population due to natural gas consumption is negligible, taking into account a median Rn-222 concentration of 0.3 Bq/l at the well (which is just 10 times higher than the radon concentration in dwellings due to the exhalation from building materials) and an even lower concentration due to its short half-life when it is consumed. It should, however, be noted that Rn-222 concentrations of 700 Bq/kg have been measured in certain liquid gas condensates which require further investigations.

REFERENCES

(1) Vernadsky, W., Sur les eaux naturelles riches en radium. C.R. 190 (1930) 1172-1175
(2) Gmelin Handbuch der Anorganischen Chemie RADIUM, Ergänzungsband 1, Berlin 1977
(3) Born, H.-J., Geochemische Zusammenhänge zwischen Helium-, Blei- und Radiumvorkommen in deutschen Salzlagerstätten. Kali, verwandte Salze und Erdöle 30 (1936) H.5, 41-45
(4) Patteisky, K., Die thermalen Solen des Ruhrgebietes und ihre juvenilen Quellgase; Teil I. Glückauf 90 (1954) 1334-1348
(5) UNSCEAR 1982, Ionizing Radiation: Sources and Biological Effects; Annex D, Table 12. United Nations, New York, 1982.
(6) Wojcik, M., Procedure for the measurement of radon concentration in natural gas. PTB-Ra 13 (1983)
(7) 80/836/Euratom: Council Directive of 15 July, 1980 Amending the Directives Laying Down the Basic Safety Standards for the Health Protection of the General Public and Workers Against the Dangers of Ionizing Radiation; Official Journal of the European Communities 23, No. L 246 (1980)
(8) ICRP Publication 30, Parts 1-3: Limits for Intakes of Radionuclides by workers; Annals of the ICRP 2, No. 3/4 (1979), 4, No. 3/4 (1980) and 6, No. 2/3 (1981)

RADIOLOGICAL IMPACT OF A NATURAL GAS POWER PLANT

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Although natural gas (NG) is regarded as cleanest among all fossil fuels, it is not necessarily so clean in radioactivities. Some NG, especially those of U.S.A. and Canada, contain a large amount of radon. Two of radon daughters, ^{210}Pb and ^{210}Po, are very important in the pathway through seafoods as already reported in the last IRPA Congress for a coal-fired power plant, and they are important for an NG power plant also. Another possibility is chromosome aberration of the public near an NG power plant. Since RBE of α vs. β in chromosome aberration is very high, α-activities from a thermal power plant could have a considerable effect. Preliminary estimates have shown that the effect of an NG power plant using some American or Canadian gas could be high enough to cause observable chromosome aberration in the public near the power plant. Since the radioactive contamination due to an NG power plant has been completely overlooked, due care must be given to this possibility.

1. RADON AND ITS DAUGHTERS IN NATURAL GAS

Natural gas (NG) is usually believed as cleanest among all kinds of fossil fuels. For chemical pollutants this is true, but for radioactivities it is not necessarily true. Some of NG, especially those of U.S.A. and Canada, contain a large amount of radon. According to UNSCEAR1982 (1) in ordinary NG the radon concentration is of the order of 1-10 pCi/l, whereas in some American or Canadian NG it is as high as 100-1000 pCi/l.

Although ^{222}Rn itself is short-lived (3.8d), its daughter, ^{210}Pb, is long-lived (22y) and hence may remain in the gas to be emitted from an NG power plant. As already reported in the last IRPA Congress (2), ^{210}Pb and its daughter ^{210}Po (138d) are important in its relation to the pathway through seafoods. Although the dose due to them is probably smaller than the previous estimate (2) when newer data of ^{210}Pb of seafoods near Japan (3) are used (as discussed later), they are still important radionuclides.

Therefore, to assess the radiological impact of an NG power plant not only the concentration of ^{222}Rn but also those of ^{210}Pb and ^{210}Po must be known. They are measured for NG of the North Sea by Wilkins (4). The results are, on the average, about 1 pCi/l for ^{222}Rn, 0.4-0.6 fCi/l for ^{210}Pb, and the ratio of ^{210}Pb to ^{222}Rn is 4-6 x 10^{-4}, which is just about equal to the ratio of their half lives, namely 3.8d/22.3y=4.7 x 10^{-4}.

This shows that, while in an NG well ^{222}Rn and ^{210}Pb are in equilibrium, ^{210}Pb is completely removed when the gas is extracted and only ^{222}Rn remains, which decays to ^{210}Pb. The ratio of ^{210}Po/^{210}Pb in this measurement is 0.1-0.5, but this is probably because the gas is extracted from the North Sea NG well and is transported to the mainland of the United Kingdom, which takes a short time, and hence ^{210}Po has not yet reached an equilibrium with ^{210}Pb. After a sufficient time ^{210}Po reaches equilibrium. Therefore, in the following calculation the ratio of ^{210}Po/^{210}Pb is taken to be 1.

2. RELEASE RATES FROM A POWER PLANT AND COMPARISON WITH A COAL-FIRED POWER PLANT

A 1000 MW NG plwer plant with a capacity factor of 80% consumes NG of about 1.8 x 10^9m^3/y. Assuming ^{210}Pb and ^{210}Po concentrations of 0.4-0.6 fCi/l, the

amount of ^{210}Pb and ^{210}Po in this volume of NG is 0.7-1.1 mCi/y. Usually the dust removal is not made for an NG power plant because it is believed to be clean, all of these ^{210}Pb and ^{210}Po are expected to be released, and hence their release rates are 1 mCi/y. Since the ^{210}Pb and ^{210}Po concentration of 0.4-0.6 fCi/l corresponds to about 1 pCi/l of ^{222}Rn, an easy conversion formula of the ^{222}Rn concentration in NG and the release rates of ^{210}Pb and ^{210}Po from a 1000 MW NG power plant with 80% capacity factor is

$$1 \text{ pCi/l of } ^{222}Rn \longrightarrow 1 \text{ mCi/y of } ^{210}Pb \text{ and } ^{210}Po \qquad (1)$$

As mentioned above, the ^{222}Rn concentration in ordinary NG is 1-10 pCi/l, which corresponds to the ^{210}Pb and ^{210}Po release rates of 1-10 mCi/y, but in some American and Canadian NG the ^{222}Rn concentration is 100-1000 pCi/l, in which case the release rates are 100-1000 mCi/y.

These values should be compared with the ^{210}Pb and ^{210}Po release rates from a 1000 MW coal-fired power plant, which is usually of the order of several tens of mCi/y. Therefore, one can see that, while the release rates of an NG power plant using ordinary NG are much lower than those of a coal-fired power plant, when American or Canadian NG is used, they are higher.

It should also be kept in mind that the conversion formula of (1) is obtained assuming that ^{210}Pb and ^{210}Po in an NG well is removed almost completely. Since they are in equilibrium with ^{222}Rn, if they completely remain, the above estimate could be increased by $(4.7 \times 10^{-4})^{-1} \approx 2100$ times. Thus even if only 0.1% of ^{210}Pb and ^{210}Po in the well remain, they would increase the total release rates by about a factor 3. Hence the above estimate based on the assumption of complete removal of ^{210}Pb and ^{210}Po in the NG well is rather conservative.

These considerations show that the radiological impact of an NG power plant using American or Canadian NG could be considerable, although it has been completely overlooked so far.

3. DOSE DUE TO LEAFY VEGETABLES AND SEAFOODS

In the following the ^{222}Rn concentration in NG is assumed to be 100 pCi/l, which corresponds to the release rates of ^{210}Pb and ^{210}Po of 100 mCi/y.

The estimate of the dose through the pathway of leafy vegetables and seafoods was already given in the case of a coal-fired power plant in ref. 2. These values were somewhat revised in the later calculation (5). In both the papers the intake of ^{210}Pb is assumed to be 12 pCi/d. That of ^{210}Po is assumed 40 pCi/d in ref. 2, and 12-40 pCi/d in ref. 5. These values are based on the data by Takata et al. (6) and Okabayashi et al. (7). It was later found that 40 pCi/d was an overestimate and 12 pCi/d is appropriate for ^{210}Po also.

However, recently new measurement of ^{210}Pb were made (3), according to which the ^{210}Pb intake appears to be lower than the above value by about one order of magnitude. Since there are still uncertainties, in the following calculations the 12 pCi/d is assumed both for ^{210}Pb and ^{210}Po, keeping in mind that when the new data are used, the calculated values could be reduced by one order.

Then the same calculation as in refs. 2 or 5 is repeated. The results for the individual dose are given in Table 1. Although much smaller than natural background, the calculated dose is comparable to that of a nuclear or a coal-fired power plant.

Table 1. Maximum individual dose due to contamination of leafy vegetables and seafoods by a 1000 MW natural gas power plant (mrem/y)

Organ	Leafy vegetables	Seafoods	Total
skelton	1.63	5.70	7.33
gonads	0.29	0.97	1.26
breasts	0.09	0.32	0.41
lungs	0.26	0.86	1.12
thyroid gland	0.26	0.86	1.12
liver	0.74	2.60	3.34
kidney	0.63	2.20	2.83
lymph nodes	0.29	0.97	1.26
pancrea	0.14	0.55	0.69
spleen	0.14	0.51	0.69
whole body	0.49	1.69	2.18

As stated in the above discussion, when the new data on ^{210}Pb (3) are used, the dose is reduced by one order of magnitude, but even in this case if the ^{222}Rn concentration in NG is higher than 100 pCi/l, or if ^{210}Pb or ^{210}Po in the NG well is not completely removed, the dose could again be the same as Table 1.

4. CHROMOSOME ABERRATION

In the above calculation the quality factor (QF) of an α particle is assumed to be 20. However recent data suggest that the relative biological effectiveness (RBE) of α vs. β or γ can be much higher. Especially in the case of chromosome aberration RBE of as high as almost 300 has been reported for α vs. γ. For dicentrics the dose response relations are given by (8)

$$Y = 1.76 \times 10^{-4} X + 2.97 \times 10^{-6} X^2 \quad \text{for } ^{60}Co\ \gamma \qquad (2)$$

$$Y = 4.90 \times 10^{-2} X \quad \text{for } ^{241}Am \qquad (3)$$

where Y is dicentrics per cell and X is the dose in rad.

Since the natural occurrence of dicentrics in 30's or 40's is 0.05% (9), the doubling dose is about 3 rad for γ, but is as low as 10 mrad for α, thus giving RBE $\approx$ 300.

A crude estimate of the effect on chromosome aberration of the population near an NG power plant can be made as follows. The release rates of ^{210}Pb and ^{210}Po are assumed to be 100 mCi/y each, as before. For the dose estimate Camplin's calculation (10) is used with the following corrections.

First his assumption of the release rates is 2.0×10^9 Bq/y = 54 mCi/y, and so to convert it to the above value his results must be multiplied by 2. Moreover in his calculation QF = 20 is assumed, and so to convert it to RBE $\approx$ 300 his results must further be multiplied by 15. On the other hand Camplin's results on the dose due to ^{210}Pb and ^{210}Po are strongly criticized by Corbett (11), who claims that they are overestimated by at least a factor 10. Therefore the reduction of factor 20 is tentatively assumed. Since his results of the maximum individual dose is 23 mrem/y, the value corrected for chromosome aberration in terms of β or γ is:

$$23 \text{ mrem/y} \times 2 \times 15/20 = 34.5 \text{ mrem/y} \qquad (4)$$

Therefore, in the plant life of 30 years the accumulated dose is about 1 rem, and from the doubling dose of 3 rem mentioned above the expected increase of dicentrics is 30% or less (note that during the 30 years the person becomes older and the natural occurrence becomes higher), and so it is unlikely to be observable.

However, as mentioned before, if an NG that contains ^{222}Rn of several hundred pCi/l is used, or the ^{210}Pb and ^{210}Po removal in the well is not complete, the release rates of ^{210}Pb and ^{210}Po are higher, chromosome aberration may become observable.

5. CONCLUSION

An NG power plant using American or Canadian NG could be a significant radioactive polluting source. Especially the effect on chromosome aberration in the population near the power plant can be considerable, and in the extreme case the effect could reach the observable level. Since the radioactive contamination by an NG power plant has been completely overlooked, due care must be taken in the future.

REFERENCES

1. Report of the Scientific Committee of the United Nations on the Effect of Atomic Radiation, 1982
2. K. Okamoto, Proceedings of the 5th International Congress of the International Radiation Protection Association, Israel, 1980, Vol. III, 157
3. M. Shimizu, private communication
4. B. T. Wilkins, ref. 2, 351
5. K. Okamoto, Bull. Austr. Rad. Prot. Soc., March 1982, p. 32
6. N. Takata, H. Watanabe and R. Ichikawa, J. Rad. Res., 9(1968) 29
7. H. Okabayashi, M. Suzuki-Yamamoto, S. Hongo and S. Watanabe, J. Rad. Res., 16(1975) 142
8. R. J. Du Frain, L. G. Littlefield, E. E. Joiner and E. L. Frome, Health Phys., 37(1979) 279
9. T. Ishihara, J. Atomic En. Soc. Japan (in Japanese), 20(1978) 783
10. W. C. Camplin, NRPB-R107, National Radiological Protection Board, 1980
11. J. O. Corbett, RD/B5061N81, Central Electricity Generating Board, 1981

PHYSICO-CHEMICAL FORMS OF ^{14}C IN REACTOR EFFLUENTS*

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INTRODUCTION

^{14}C is produced in nuclear reactors mainly by the neutron activation reactions with the isotopes of nitrogen, oxygen and carbon. The form of ^{14}C produced may either be CO_2 or its reduced states such as CO, CH_4, C_2H_6 etc., depending on the reduced or oxidised conditions existing in the systems. The amount produced vary with the type and power level of reactors(1). This paper deals with the production and releases of ^{14}C from three reactors in India, namely the research reactors Apsara and Cirus and the power reactor, TAPS. The distribution of ^{14}C in different physico-chemical forms is presented for the various reactor system samples.

PRODUCTION RATE OF ^{14}C

The theoretical production rates of ^{14}C are given in Table-1 for the above mentioned three reactors at the indicated power levels. Major fraction of estimated ^{14}C in Cirus is through air in the air annulus around the fuel channels. ^{14}C produced in the fuel is the result of ternary fission in metal fuels (Apsara, Cirus) and activation reaction of ^{17}O and ^{14}N atoms in oxide fuels (TAPS). The measured nitrogen contents in oxide fuel is 21 $\pm$ 13 ppm. ^{14}C production in metal fuels does not include any contribution from possible nitrogen content.

Table-1: Theoretical production rate of ^{14}C in different reactors

Reactor	Type	Power level	^{14}C production rate, Ci/yr			
			Moderator	Coolant	Fuel	Others
Apsara	Swimming pool	175 KW(th)	3.0(-3)		2.9(-5)	-
Cirus	HWR	40 MW(th)	3.1	0.05	7.1(-3)	11.3(air)
TAPS	BWR	200 MW(e)	2.9		9.8	

^{14}C MEASUREMENTS IN AQUEOUS SAMPLES

A method has been developed to determined ^{14}C activity in water samples. The water samples (1 ml D_2O or 1 litre pool or reactor water) are collected in bottles containing known amounts of NaOH and Na_2CO_3. Freshly prepared MnO_2 (100 mg/l) and AgCl (1 g/l) are added to remove fission/activation nuclides and iodine isotopes respectively. The filtrate is precipitated as $BaCO_3$. CO_2 is liberated from this precipitate using lactic acid solution and absorbed in 20% ethanolamine in methanol. An aliquot of

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this solution is counted for ^{14}C in a Packard LSS model 3255 using toluene based scintillator. The efficiency of CO_2 absorption and counting is standardised with known amounts of labelled $BaCO_3$. In order to determine the organic component in 1-2 ml of D_2O moderator samples, persulphate in sulphuric acid is used for oxidation[2] and the total ^{14}C is estimated as per the procedure described above. Labelled benzoic acid upto 6 mg are wet oxidised using this procedure and net absorption-counting efficiency is determined. The results of these standardisations are given in Table-2.

Table-2: Recovery for ^{14}C determinations in water samples

^{14}C dpm Spike	Obtained	Recovery %	Chemical form
3.73(3)	3.40(3) ± 2.32(2)	92.0 ± 6.2*	Inorganic
1.80(5)	1.45(5) ± 0.11(5)	80.0 ± 6.3	Total

* Represents S.D. for 10 samples

Samples of Cirus moderator, Apsara pool water and TAPS reactor water were collected every week for about six months and the mean concentrations are given in Table-3 for both inorganic and total ^{14}C.

Table-3: ^{14}C concentrations in systems water samples

Systems	Location	^{14}C concentration, pCi/ml	
		Inorganic	Total
Moderator-Cirus	Inlet of IX	234.8 ± 86.6	253.3 ± 71.0*
Moderator-Cirus	Outlet of IX	74.7 ± 37.8	85.0 ± 50.3
Pool water-Apsara	Pool	0.3 ± 0.08	Not done
Reactor water-TAPS	Inlet of IX	5.9 ± 1.5	"
Reactor water-TAPS	Outlet of IX	0.3 ± 0.2	"

* Represents S.D. for 24 samples, IX - Ion-exchange bed

^{14}C MEASUREMENTS IN AIR

The determination of ^{14}C concentration and its chemical forms in air samples is carried out using an apparatus similar to that described by Schwibach et al[3]. The air sample is filled up in balloons, 150 litre capacity. The air is sucked at a rate of 0.5 lpm through an assembly consisting of (i) two NaOH traps (1N, 200 ml) before the furnace to absorb CO_2 bound ^{14}C activity, (ii) a furnace maintained at 800°C with CuO catalyst to convert CO and CH_4 bound ^{14}C to $^{14}CO_2$ and (iii) two NaOH traps after the furnace to absorb the oxidised form of ^{14}CO and $^{14}CH_4$. The NaOH solutions, both before and after the furnace are independently converted into $BaCO_3$ after the addition of known amounts of carbonate carrier. The precipitate is further subjected to decontamination from other nuclides by releasing CO_2 from it and reabsorbing in 20% ethanolamine prior to counting in LSS. Measurements of higher hydrocarbons

indicated very small percent (~1-2%)of the total. ^{14}C concentration in different air samples measured are presented in Table-4.

Table-4: ^{14}C in air samples - different chemical forms

Location	Mean concentration in air, pCi/l		
	CO_2 form	CO & CH_4 form	No. of samples
Stack air - Cirus	8.91 ± 2.38	5.62 ± 2.83	10
Peripheral duct - Cirus	9.11 ± 0.40	8.41 ± 1.65	3
Heat exchanger room-Cirus	0.14 ± 0.02	0.10 ± 0.02	3
Reactor Hall air - Cirus	0.13 ± 0.02	ND	2
Helium cover gas - Cirus	2.82(5)	1.40(4)	3
Off gas system - TAPS	3.15(3) ± 1.00(3)	2.78(2) ± 1.57(2)	5
Ventillation - TAPS (70 m elev.)	7.98		
Reactor Hall air - Apsara	0.10 ± 0.02	-	3

DISCUSSION

The mean residence time(mean concentration/production rate per unit volume) for ^{14}C in Apsara pool water works out to be about 7.7 days. The short residence time seems to be the result of atmospheric exchange. This is supported by low concentrations observed in mixed ion-exchanger bed, 10-40 pCi/gm, just before regeneration.

The experimental release rate of ^{14}C from Cirus stack works out to 7 Ci/yr. This is nearly 50% of the theoretical production rate.

The experimental values for both $^{14}CO_2$ form or its reduced forms showed large variations (30-50%) on different days in Cirus stack effluent. The reduced form of ^{14}C (mainly CO and CH_4) formed on an average 35-50% of the total release through stack in Cirus whereas in the off-gas system of TAPS, this is only 8-12% of the total release.

A schematic diagram of the movement of ^{14}C in Cirus reactor is given in Figure-1. The ^{14}C produced in air system is released through stack. The moderator loses its ^{14}C content by two modes - (i) adsorption on the mixed ion-exchanger bed and (ii) escape to Helium cover gas system which has a recombiner unit and a sorber bed that is regenerated on a weekly basis. ^{14}C in the moderator is mainly inorganic and its concentration is reduced to one-third as it passes through the ion-exchanger bed. Eventhough ion-exchange and helium cover gas are replaced during operations (as and when required), ^{14}C content in moderator compartment follows the first order kinetics with respect to removal processes with average rate coefficients. The governing equation is

$$dA_m/dt = R - (\lambda_p + k_1 + k_2 + k_3)\ A_m$$

The mean residence time of ^{14}C in D_2O moderator is calculated to be about 0.5 days. The maximum build-up of $^{14}CO_2$ in Helium cover gas just prior to sorber regeneration (1.5 $\mu Ci/l$) could be accounted for in terms of its sorption circuit and input from moderator circuit. The nitrogen impurity in the Helium cover gas is not allowed to build-up beyond 0.06% and this would not contribute ^{14}C build-up to more than 0.1% of that from moderator circuit.

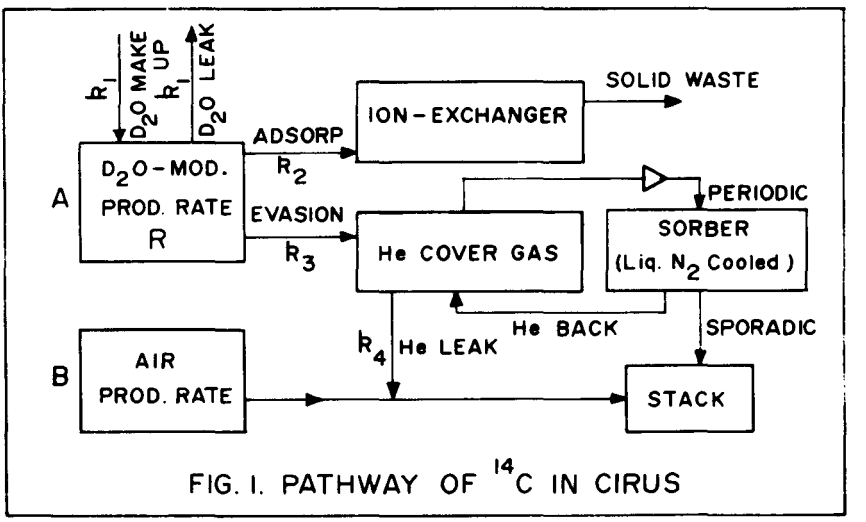

FIG. I. PATHWAY OF ^{14}C IN CIRUS

The theoretical production rate of ^{14}C for TAPS works out to 13 Ci/yr and only about one-fourth of this would be released from the reactor. The remainder would be released during the reprocessing of the fuel. The residence time of ^{14}C in TAPS reactor water is very small, in the order of hours. The off-gas system accounts for nearly 60-70% and general ventillation for 15-20% of the release rate from the reactor.

REFERENCES

1. UNSCEAR - Sources and Effects of Ionising Radiation, Report to General Assembly, with annexes, 1982.

2. Baker, E., Feinberg, H. and R. Hill., Anal. Chem. 26; 1504, 1954.

3. Schwibach, J., Riedel, H. and J. Bretschneider., Investigations into the emission of ^{14}C compounds from nuclear facilities, Report of study contract 1144-77-10L/V, Commission of Europeon Communities, 1978.

NATURAL RADIATION FIELD IN CZECHOSLOVAKIA

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The total exposure rate of natural radiation field (i.e. coincidental radiations from terrestrial and cosmic components) in Czechoslovakia was measured and the average gonadal exposure rate was established. The detail analysis of measured values due to spectral differences, as well as due to different local and meteorological conditions, are given. In the end, the extreme areas along the whole country are pointed out and corresponding collective doses derived.

CONCENTRATION OF NATURAL RADIOACTIVITY IN A TiO_2 PRODUCTION PROCESS

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INTRODUCTION

Ionizing radiation is to some extent present in whole environment. Certain human activities can result in a modification (enhancement or reduction) of natural exposure. A well known example of a localized enhancement of exposure to natural radioactivity is related to the release of naturally-occurring radioactive isotopes, normally present in coal, during its burning for power production. On the other hand, a typical case of a reduction of such an exposure is the use of water-purification processes which decrease the contents of natural radioactive isotopes in water (UN82).

In this paper we present a case of a technologically enhanced concentration of natural radioactive isotopes which occurrs during the production of titanium dioxide from the ore ilmenite. Isotopes of natural series Th-232 and U-238 whose contents in ore is relatively low are being concentrated in the chemical processes at several places in the factory.

MEASUREMENTS AND RESULTS

In 1982 a team of Ecological Laboratory with a Mobile unit from the J. Stefan Institute in Ljubljana detected dose rates well above the typical background values at some places in a chemical factory producing titanium dioxide. Several samples were collected and the preliminary analysis with a high resolution gamma spectrometry showed the presence of concentrated isotopes of natural series Th-232 and U-238.

This discovery requested an action with several important aims:

- to discover the source of these isotopes and their pathway through the process,
- to measure and/or access exposures of workers to the enhanced natural radiation,
- to find and implement necessary protective measures to keep exposures of the workers below the limits and to prevent any unnecessary exposure,
- to undertake measures to prevent the contamination of the environment.

For that reason a thorough radiological investigation of the whole technological process was done. It consisted of detailed dose rate measurements, determination of air concentration of isotopes, measurement of contents of radioactive isotopes in raw and waste materials, end product and in several other samples from the production process.

Dose rates ranged from natural background values to up to 50 μGy/h (measured contactly at the surfaces of some vessels, pipes and wet filters). It was roughly estimated that at mostly 10 % of the whole factory area the dose rate exceeded twice the natural

background, the highest values appearing only in the very vicinity of the contaminated surfaces.

Air concentrations of Rn-220 and Rn-222 were measured at 12 different locations. Most of the values were below the detection limit of the field instrument used (20 Bq/m^3 for Rn-222 and 50 Bq/m^3 for Rn-220). A measurable concentration of Rn-222 was found only in one of the closed vessels (320 Bq/m^3) and that of Rn-220 in the same vessel (6000 Bq/m^3) and in the vicinity of the mill (800 Bq/m^3).

Air concentration of gamma-emmitting radionuclides on dust particles was determined at the location with the highest dose rate. Dust from 14 m^3 of air was collected on air filter and its gamma activity was measured afterwards. The daughters of Rn-220 were detected in a concentration of 0.26 Bq/m^3.

Specific activities of the collected samples were determined by means of the gamma spectrometry. The results are shown in Table 1.

Table 1: Specific activities of different samples

Sample	Th-232 series		U-238 series
	Ra-228(1)	Th-228(2)	Ra-226(3)
ilmenite	180 Bq/kg	180 Bq/kg	150 Bq/kg
TiO_2	31 Bq/kg	9 Bq/kg	32 Bq/kg
filter 1	180 kBq/m^2	88 kBq/m^2	94 kBq/m^2
filter 2	300 kBq/m^2	64 kBq/m^2	170 kBq/m^2
rubber coating	345 kBq/m^2	200 kBq/m^2	160 kBq/m^2

(1) obtained from Ac-228 gamma lines
(2) obtained from Rn-220 daughters gamma decays
(3) obtained from Ra-226 gamma line

Radioactive isotopes of Th-232 and U-238 natural series enter into the chemical process with the ore ilmenite ($FeTiO_3$) of Australian origin. Ilmenite turns out to be the only raw material of significant specific activity. Assuming secular equilibrium the concentrations of Th-232 and U-238 in ilmenite are 180 Bq/kg and 150 Bq/kg respectively.

Specific activities of other samples indicate that chemical processes concentrate isotopes at certain surfaces in the factory, especially on rubber coating of some vessels, pipes and on wet filters. Radioactivity is also detectable in the final product titanium dioxide.

In contrast to the case of ilmenite where assumption of the secular equilibrium seems justified (C080), chemical processing may lead to a selective concentration of different isotopes, resulting in a nonequilibrium state of isotopic ratios. Since gamma spectrometry can directly reveal only the presence of gamma-emitting radionuclides it cannot provide the insight into the migration of all isotopes. Furthermore when the activity is measured with a certain delay after the disequilibration process the initial concentration of only those gamma emitters can be determined whose half-life is comparable to the length of the delay. In our case this delay was

estimated to range from several months to some years what makes possible to determine the activities of Ra-228 (half-life 5.75 y) and Th-228 (half-life 1.9 y) from Th-232 series and that of Ra-226 (half-life 1600 y) from U-238 series. Additionally the contents of Ra-228 may be increased after the disequilibration process by the decay of its predecessor Th-232 and similarly that of Th-228 by the decay of Ra-228.

It is evident from Table 1 that chemical processes concentrate Ra-226 from U-238 series. Because of the same chemical behaviour of Ra-228 from Th-232 series a similar conclusion can be drawn about its concentration though it may be also produced by the decay of its predecessor Th-232. On the other hand it is not possible to attribute uniquely the presence of Th-228 in samples to a concentration process because it may origin from the decay of Ra-228.

An attempt was also made to estimate roughly the activity balance of the whole process. By extrapolating the measured specific activities to the full scale it was possible to explain the redistribution of 20 % of the incoming activity. According to this estimate the radioactive isotopes partly gather at certain places in the factory (10 %) and partly leave the factory in the final product (less than 10 %). The pathway to the environment in waste products was estimated to be unsignificant. The discrepancy of the whole activity balance might be due to possible unrepresentativeness of the collected samples and/or to underestimation of the area of contaminated surfaces.

Final estimate of exposure of workers to the enhanced natural radiation, based on extremely conservative approach, showed that it is very unlikely that any kind of exposure could reach adopted dose limits for individual members of the public.

SUMMARY AND CONCLUSIONS

A new case of modification of natural radiation was discovered in addition to a great number of well known examples. It was found that processes in a chemical factory producing TiO_2 concentrate radioactive isotopes from natural decay series entering the process with the ore ilmenite of relatively low specific activity. Individual chemical behaviour of members of the series seems to lead to a different degree of concentration of various isotopes.

LITERATURE

(C080) R.Colle, The Physics and Interaction Properties of Radon and Its Progeny, NBS Special Publication 581, Washington 1980

(UN82) UNSCEAR, 1982 report

THE CONCENTRATION OF RADON IN A TOWN WHERE RADIUM-ACTIVATED PAINTS WERE USED

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INTRODUCTION

In a particular town in the Swiss Jura, important for its watch industry, probably 20 workshops existed between 1920 and 1963, where dials and faces were painted with radium-activated paints. Some workshops produced even their own paints.

After removal of the radium and complete decontamination of such workshops, which prestens its own problems, one sometimes still finds a high radon concentration in the cellar of such a workshop. Similar high values are sometimes found also in adjacent houses. In that particular town, however, a larger area was concerned.

MEASUREMENTS

In cooperation with the Bundesamt für Gesundheitswesen and the local authorities we decided to place radon dosimeters for several months in the suspected area and in some other quarters, where we did not expect radon. We placed 3 dosimeters in 95 houses, one in the cellar, one in the living room and one in a bed room. 59 dosimeters were placed in shafts associated with the tubes of the drinking water distribution (shafts). 29 dosimeters were put in the shafts giving access to the sewage (sewerage). All dosimeters were of the type SF from Terradex.

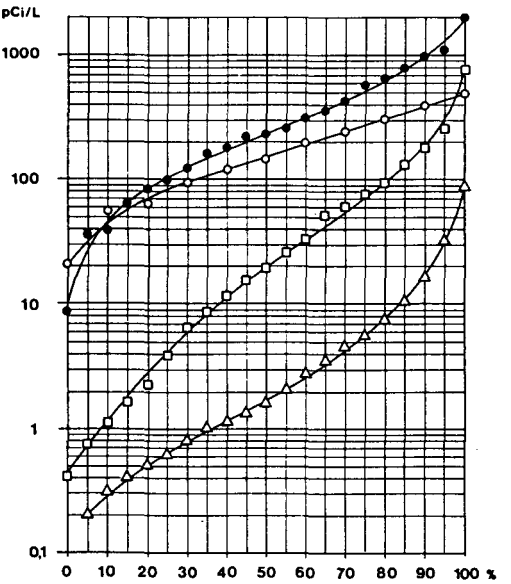
RESULTS

Table 1 shows the mean and extreme values of these measurements. Fig. 1 presents the cumulative distribution of the measurements. For the latter representation we combined the values from living rooms and bed rooms into one value, adding them after weighting with a factor of 3/4 and 1/4 respectively. These weighting factors result from the assumptions of the UNSCEAR-Report 1982 about the breathing volumes in these rooms: Living room 5.5 h x 1200 L/h and 5,5 h x 750 L/h, bed room 8 h x 450 L/h. The assumptions for the calculation

		radon (pCi/L)	
place	n	mean	range
shafts	59	360	8,7 - 2000
sewerage	29	190	20 - 500
cellars	94	60	0,4 - 730
living rooms	95	7,5	0 - 100
bed rooms	95	4,0	0 - 24

Table 1: Mean and range of the radon concentration

Fig. 1:
Distribution of the Rn-concentration in
- ● shafts
- o sewerage
- ◘ cellars
- ▲ living- and bed rooms (weighted 0,75 and 0,25 and added). This value tims 0,1 yields the annual dose in rem.



of the annual effective equivalent dose are a daughter/radon equilibrium of 0.5 (1 pCi/L = 0.005 WL = 1.04 E-10 J/L) and a dose factor of 200 rem/J (as currently used by UNSCEAR, OECD and NEA for indoor radon). For radon at this particular equilibrium we abtain 2.08 E-08 rem/pCi. Multiplication by an annual breathing volume (indoors) of 5.23E06 L yields 0.109 rem/(pCi/L). 0.1 rem/(pCi/L) may be an adequate precision for the calculation of the annual dose from a weighted radon concentration.

From table 1 we combine a weighted mean of 6.6 pCi/L, yielding a mean dose of 0.66 rem per year. From Fig. 1 we see, that a minority of houses with relatively large doses (up to 8.4 rem) contribute to this high mean. If we assume that a person spends some time in the cellar (e.g. half an hour per day, breathing rate 1200 L/h) one would have to add the radon concentration in the cellar multiplied by a weighting factor of 0.04. This would add a dose of 0.24 rem in the mean and 2.9 rem in the extreme case.

Fig. 2 shows the local distribution of the measurements in this town. We have indicated the values measured in the cellars, as the radon appears to leak into the dwellings through the cellars.

Fig. 3 shows the local distribution of the measurements from the shafts associated with the tubes of the drinking water. There is no evidence that the drinking water itself has something to do with the radon in this town. The radon in the shafts is rather a very good indicator for the radon in the soil. The measurements of the shafts, as far as available (Fig. 3), correspond reasonably well to the measurements in the cellars (Fig. 2).

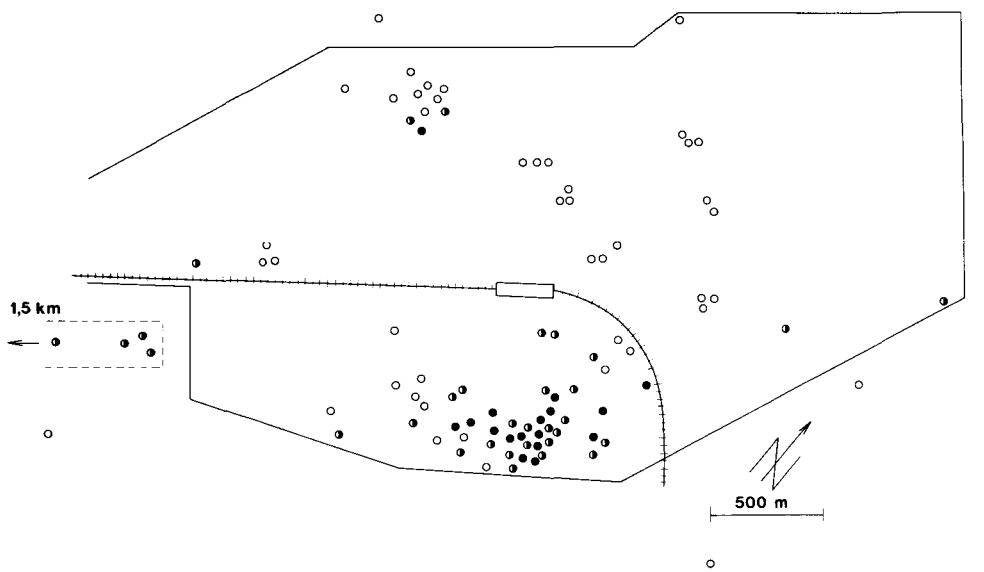


Fig. 2: Rn-Concentration in cellars
○ 0-20 pCi/L ◐ 20-100 pCi/L ● 100-780 pCi/L

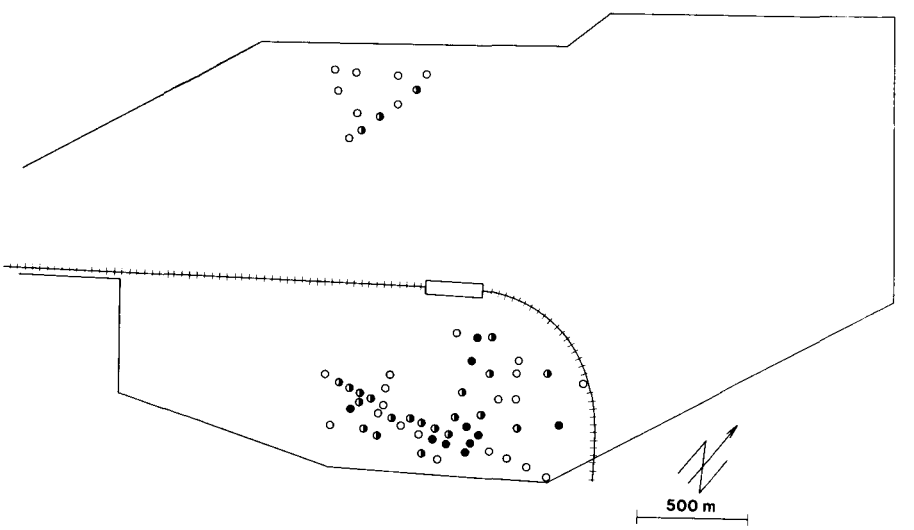


Fig. 3: Rn-Concentration in shafts
○ 0-200 pCi/L ◐ 200-600 pCi/L ● 600-2000 pCi/L

DISCUSSION

The town is located on limestone, which contains very little uranium. Therefore we would not expect much radon. This was confirmed earlier by a study carried out by the Eidgenössisches Institut für Reaktorforschung. The mean from 25 houses elsewhere in the Jura was 2.5 pCi/L in the cellars and 1.4 pCi/L in the living rooms.

On the other hand, the radon seems to emanate from the soil. The highest values are found in the shafts (Fig. 1). These shafts with a volume of about 8 m^3 are hardly ventilated. The measurements in the shafts seem to be the best indicator for the radon in the soil. The radon in the cellars, also strongly correlated to the concentration in neighbouring shafts, shows lower values and a larger scatter due to a larger and veriable ventilation.

The measurements in the sewerage were carried out in order to find out whether deposits of radium in the sewerage could be the source of radon. However this hypothesis must be discarded for two reasons. First the sewerage shows lower concentrations of radon than the shafts (and the soil). We cannot expect the radon to move in the opposite direction of this gradient of concentration. Second the sewage was conducted into local pits until 20 years ago. Radium previously poured into the sink must still stay in these pits. In one case such a pit contaminated with radium was recently found. A workshop where radium was used was connected to this pit as well as other houses where high levels of radon were found. It is planned to ventilate this pit and to see whether the radon in these houses decreases.

There is much suspicion that in some cases of very high radon concentration unknown deposits or pits with radium waste are likely to be the cause. On the other hand, some geological anomalies may also be important. The limestone is very much fissured and contains molasse in certain areas. If this particules geological situation would be another cause for some elevated values of radon, it would explain why some rather high concentrations of radon were also found in houses chosen as a reference, because they were in quarters of recent construction or far away from earlier activities involving radium (as far as we know). This point awaits further clarifaction.

The owners of those houses, where an annual dose above 3 rem is calculated, are invited to study counter measures on a volontary basis in a first round. Depending on the experience gained from these cases, measures may be proposed later also for less urgent cases.

NATURAL RADIATION EXPOSURE FROM ^{226}RA IN GERMANY[+)]

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Concentration of ^{226}Ra has been measured in more than 1ooo samples: human bone, still born infants, fetuses, blood, mixed diet, drinking water and mineral water. It could be demonstrated that there is an age-dependence of ^{226}Ra-concentration in human bone. Up to the age of 2o, two maxima of ^{226}Ra-concentration occur in bone. These maxima coincide with periods of increased velocity of skeleton growth. Normally, food is the main source of the uptake of radium by man. A transfer factor from diet to bone of about o.o98 has been noted in Germany. Compared with data (1) obtained from other countries - mainly the U.S. - the German value is higher by a factor of about 1.5. This difference may be due to the lower calcium intake in Germany, leading to increased resorption of radium.

Materials and Methods

All biological samples were dried and ashed at 65o°C. The radium content of the ash was chemically isolated from other inorganic substances. The measurement was done at a low-level Alpha- or Gamma-Spectrometer. ^{226}Ra-concentration in samples of drinking water and mineral water was determinated by collection of ^{226}Ra on selective ion-exchanger (2) and subsequent measurement in a low-level Alpha-Spectrometer.

The food examined was mixed hospital diet from Kiel University Hospital. Mineral waters from different sources were bought at random from local retailers.

Results

The average values of the measured concentration of ^{226}Ra (3) are designed in Table 1, which shows also the concentration-range (mBq ^{226}Ra/g fresh weight) in the different types of samples (Cortical bone, taken from adults; still births; neonatal deaths and placentae; blood plasma; standard diet; drinking water and mineral water).

The age-dependence of ^{226}Ra in bone is shown in Figure 1. Each point represents an average of 18 to 29 samples measured. All in all, 955 human cortical bone samples were measured.

[+)] With Support of the Bundesminister des Innern of the Federal Republic of Germany

Sample type	^{226}Ra mBq/g	range	sample size (N)
cortical bone (adults)	o.26	o.o6 - o.52	865
still births	o.o7	o.o4 - o.11	45o
neonatal deaths 7d - 1a	o.24	o.o5 - o.38	9o
placentae	o.o3	o.o3 - o.o4	2263
blood-plasma	o.o2	o.o1 - o.o3	7 (liter)
standard-diet	o.o1	o.oo6 - o.16	235 (daily rations) equivalent: 33o kg
drinking-water	o.o1	o.ooo4- o.1o	98
mineral waters	o.o7	o.ooo4- o.6o	157

Tab. 1: Arithmetical average and range of measured radium concentration per g fresh weight

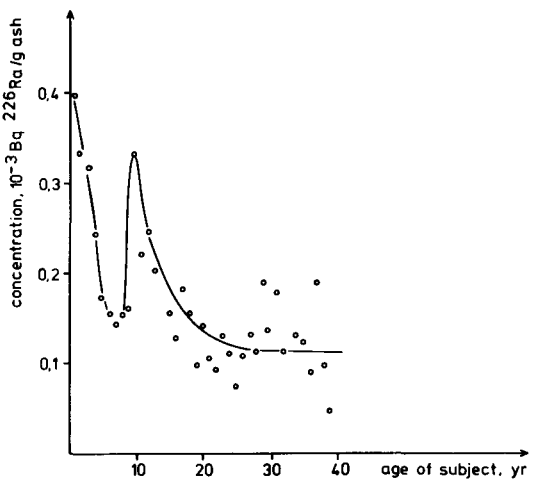

Fig. 1: Radium concentration in bone ash as a function of age

Discussion

There are two maxima to be noted in the function of age-dependence of the ^{226}Ra-concentration in bone. The first of these maxima is obtained at the age of one year, whereas the second one is reached between the age of 1o and the age of 16. These maxima coincide with the first and the second phases of rapid growth of the human skeleton. During these phases, the discrimination of Ra against Ca appears to be considerably reduced in comparison with that dis-

crimination during the normal phases of stationary bone-growth. This is due to the rapid build-up of hydroxylapatite crystals in bone, with the result that the crystals take in as much Ra as Ca. During the phases of stationary bone-growth, diffusion and recrystalisation are the predominant processes which cause Ra to be substituted by Ca, i.e. the elimination of Ra to be accelerated.

Age-dependence of ^{226}Ra in the skeleton has already been observed with chickens.

A continual intake of ^{226}Ra with daily diet can be supposed to occur in both cases. The results can only be explained if the correlation, that exists between Ra and Ca with regard to their equivalent chemical and biochemical behaviour, is taken into consideration.

The transfer factors, which are defined as the ratios of ^{226}Ra-concentration in the target substances to those in the starting substances, were also determinated for some combinations of interest.

The transfer factors measured for these transitions are listed in Table 2.

	^{226}Ra
TF $\frac{\text{blood plasma}}{\text{food}}$	15.5
TF $\frac{\text{placentae}}{\text{food}}$	7.1
TF $\frac{\text{still-births}}{\text{food}}$	o.o31
TF $\frac{\text{bone}}{\text{food}}$	o.o98
TF $\frac{\text{still-birth}}{\text{placentae}}$	o.oo44

Tab. 2: Transfer factor (TF) for ^{226}Ra per gram

If considering the radiation exposure caused by intake of ^{226}Ra in the human body, potable waters represent only a minor part of the total intake. If presupposing a average concentration of 4 mBq ^{226}Ra/l in German drinking water, and, realistically, the consumption of o.5 liters per day of drinking water, the total intake resulting from water is of 2 mBq ^{226}Ra/d. Compared to the 15 mBq ^{226}Ra/d resulting from the intake of normal diet, potable water appears to be of minor importance. Some persons, however, consume up to 2 litres of mineral water a day, which, with a maximum concentration of o.6 Bq ^{226}Ra/l, results in a daily intake of 1.2 Bq ^{226}Ra, i.e. an annual intake of about 438 Bq ^{226}Ra. Presupposing a transfer factor water/bone of o.o98, the amount of ^{226}Ra collected during on year is of 43 Bq ^{226}Ra. Calculated on the basis of the "Strahlenschutzverordnung" (4) (Official German radiation protection regulations), this corresponds to an annual dose of $D_{\sigma}{}^{226}Ra,bone = 3.4$ mSv. This value is ten times superior to the limit of o.3 mSv/a laid down in the "Strahlenschutzverordnung".

Literature

(1) FISENNE, I.M., KELLER, H.W. and HARLEY, N.H.
Worldwide measurement of ^{226}Ra in human bone: estimate of skeletal α-dose. Health Physics 2, 1981, 163-172

(2) GLÖBEL, B. and BERLICH, J.
Eine einfache und schnelle Methode zur Bestimmung von ^{226}Ra in wässrigen Proben. 5. Fachgespräch "Überwachung der Umweltradioaktivität" Karlsruhe 1983, in Druck

(3) MUTH, H. and GLÖBEL, B.
Age dependent concentration of ^{226}Ra in human bone and some transfer factors from diet to human tissues. Health Physics, Vol. 44, Suppl. No. 1, 1983, 113-122

(4) Bundesgesetzblatt, Z 1997A, Bonn 2o.1o.1976
Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung)

INDOOR RADON LEVELS IN SWITZERLAND
- Results of a National Survey -

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PROGRAM

Uneasiness about the influence of energy saving methods such as weatherproofing on radon levels in dwellings was the reason for the Federal Energy office to ask EIR to investigate the radon situation. In a first phase a thorough literature search was done, contacts to institutions with radon experiences were established, passive track etch dosimeters in filter cups of the Karlsruhe type were procured and the electrochemical etching method was made operational. A pilot survey in 123 houses in selected regions in winter 1981/82 aimed at an assessment of the magnitudes and distribution of radon concentrations in Swiss dwellings. Part of the houses were chosen in regions with higher natural radioactivity in ground and local building materials. The control houses were in the low background Jura and Plateau regions around Würenlingen which are rather typical for the more densely populated parts of the country. Three detectors in each house (basement, living room, bedroom) were exposed for six months. The detector foils were counted with an image analyzer. Calibration exposures at Karlsruhe, SSI Stockholm and NRPB Chilton, UK (EC intercalibration) gave results within ± 30% or better. Some high level cases were further investigated in summer 82. In a parallel program building materials and source water were analyzed for radium content.

In a second phase in 1983 the survey concentrated on the effects of weatherproofing on indoor radon levels in 105 houses, of which one group had already been monitored for conventional toxics in an IEA-AIC air quality program. Matched pairs of houses consisting of similar neighbouring houses, one weatherproofed, the other conventional, were selected with the help of architects in order to minimize the large local and regional variations. The radon detectors were fitted with TLD dosimeters for a parallel survey of the gamma levels.

Based on the results and experiences from the first two phases further surveys are prepared for 1984 and later. Using the dosimetry services of EIR the Federal Commission for Surveillance of Radioactivity (KUER) continues the survey of the radon exposure of the population with passive detectors, concentrating now on the main population centers, multistory appartment buildings and working places. EIR in cooperation with building and air quality specialists starts a program of coordinated monitoring of the time variations of radon, ventilation, climate and ventilation habits of the inhabitants. The results shall help to prepare reasonable ventilation, building and energy conservation recommendations and to gain more information about important parameters and possible methods of reducing elevated radon levels.

RESULTS

Phase I: The average radon concentrations in the monitored rooms were between 0.4 and 105 pCi/liter (15 - 3900 Bq/m^3) with most frequent levels between 0.5 and 2 pCi/l (20 - 70 Bq/m^3). The distribution of the values is similar to those found in other countries. The regional differences correspond to the geological expectations, and the decreases towards upper floors show that with a few exceptions the main radon source is located in the ground under the buildings. Special investigation of two houses with elevated levels in low background regions showed not only large differences to neighbouring houses, thus confirming Swedish studies on large local variations of radon exhalation from ground, but also located the radon source in basement rooms with bare earth floors. In a few alpine houses granite as building material made significant contributions /1/ /2/.

RADON CONCENTRATION (pCi/liter) WINTER 1981/82

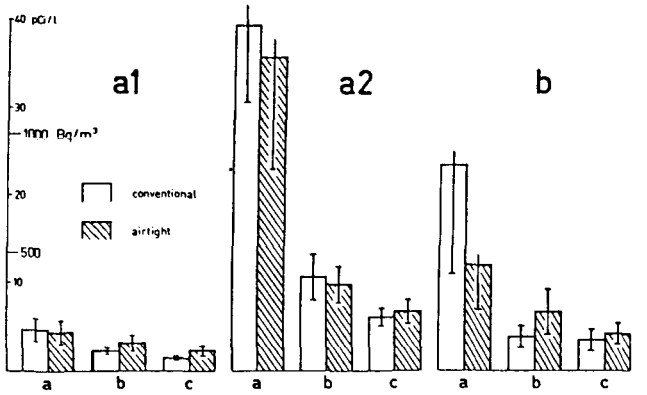
FLOOR	BASEMENT			GROUND FLOOR			UPPER FLOORS		
REGION	number of houses	concentration average	concentration extreme	number of houses	concentration average	concentration extreme	number of houses	concentration average	concentration extreme
JURA	25	2,5	18	33	1,4		28	1,3	
PLATEAU	26	2,6	51	43	1,8		32	1,4	
RHINE AG	4	3,3		4	2,4		4	1,4	
NAPF	5	4,3	30	7	1,4		8	1,3	
ALPS	13	7,8	66 / 68	12	3,8	105	27	2,1	
SOUTH ALPS	10	6,7	31 / 95	12	3,8	81	22	3,8	

Phase II: Radon levels between 0.6 and 149 pCi/l (22-5500 Bq/m^3) were found. In the matched pair samples significant differences between weatherproofed and conventional houses were found with somewhat lower values in the basements but higher values in the inhabitated rooms, indicating different ventilation patterns (increase for ground floor living rooms 2.9±1.6 pCi/l or 1.8 times the conventional level) /3/. TLD measurements of gamma exposures showed a correlation with radon levels but not with airtightness. The alpine high level radon regions have also high annual gamma exposures of 150-240 mrem/a in living rooms compared to 40-95 mrem/a in the lower Plateau regions. One house in the middle of a row had been made very energy efficient, resulting in an extremely low fuel consumption around 250 l/a but also in tripled radon levels up to 23 pCi/l and problems with humidity in the basement.

DISCUSSION AND INTERPRETATION

The number of houses surveyed was limited but not small in relation to the size of country and population. In the selection there was partly a deliberate bias towards probable high levels, but the controls, about half of the houses, were quite representative for living conditions of large parts of the Swiss population. Thus the results were judged to be suitable for a preliminary estimate of the

RADON INDOOR CONCENTRATIONS IN AIRTIGHT AND CONVENTIONAL DWELLINGS
a1) Plateau regions, a2) alpine regions, b) matched pairs
a: basement, b: ground floor, c: first floor. shaded: airtight homes



average radon exposure. Taking an average radon concentration of 1.5 pCi/l (55 Bq/m^3) and applying the parameter and dose estimates recommended by UNSCEAR 1982, an average effective dose equivalent of 180 mrem/a (1.8 mSv/a) is deduced. A significant part of the population is expected to have radon exposures above the ICRP dose limit of 500 mrem/a, in some cases even approaching the limit for professionally exposed persons. This practically doubles the total average annual radiation exposure of the Swiss population from all sources. Qualitatively the influence of energy conservation has been demonstrated, the quantitative contribution to the population exposure cannot yet be estimated. Thus radon at least in part qualifies as "technologically enhanced" exposure, and sooner or later the question of control or not will have to be discussed and radon exposures will have to be put in proportion to exposures from effluents of nuclear installations and other radiation sources. Building and ventilation codes will no doubt be influenced by radon.

AVERAGE ANNUAL EXPOSURE OF THE SWISS POPULATION
effective dose equivalent (mrem/a)

Source	$\bar{H}_{eff}$	Source	$\bar{H}_{eff}$
Natural radiation		Medicine	150
- terrestrial	65	Fallout	4
- cosmic	32	Industry, Nucl.Power	< 1
- internal	30	Consumer Products	1
Radon + daughters	180	Professional Exposure	0.3

INFORMATION AND REACTION OF THE PUBLIC

The volunteers of phase I were found through an inquiry among employees of EIR, weather observers of the Meteorological Institute and members of scientific commissions. They received an information sheet on radon with a questionnaire and a letter asking for collaboration, on which over 30 % answered positively. The selected participants answered questions on the houses at the beginning of the survey and on the positioning of the detectors at the end. Press comments due to an unknown source caused additional inquiries from interested or concerned house owners, authorities and scientists. All participants reacted very matter-of-fact. Outside comments such as "Why do they first recommend energy conservation and only afterwards tell us it may be dangerous?" and inquiries about possible connections of radon with sickness or with "earth rays" were to be expected.

The results of phase I were issued in a planned and coordinated way at the same time to all participants, authorities and mass media. The participants received a 3-page summary and a coded list of the results of all houses (they knew only the code of their own house but got the other values for comparison). The media (newspapers, news agencies, radio, TV) received the summary and a shorter press release. One month later a complete technical report /1/ was ready for distribution to authorities, scientists and others who had asked for it. The media response was surprisingly good with some larger features in major papers and more or less complete reproduction of the press release by most of the others, including radio an TV. The participants' reaction was calm. The public was generally surprised at the magnitudes of levels and estimated exposures, but a frequent reaction and comment was that it was "natural" and therefore not hazardous. As we had avoided estimates of cancer risks or comparisons to nuclear energy, no panic reactions or political issues followed, but the radon begins more and more to play a role in scientific and public discussions of radiation risks, up to now in a reasonable and objective fashion. An unexpected second "radon wave" in the media was triggered in summer 83 by a science writer with a feature article on our radon study which appeared in several regional papers. This resulted in a wave of radio and TV interviews and comments which in turn led to newsagency and newspaper reports most of which showed no sign of memory of the fact that the same sources had reported on the same results half a year earlier...

In phase II the local architects encountered some problems in convincing their clients to participate, in one region due to concurrent problems from exploratory drilling for waste storage, but in various degrees also due to second thoughts or uneasiness about possible claims in case of elevated levels. The owner of one high level house that was further monitored stopped this out of fear that the house would drop in commercial value if this should become known in the community.

REFERENCES

/1/ H.Brunner et.al. "Radon in Wohnräumen in der Schweiz", internal report EIR TM-81-82-11 Dec. 1982

/2/ H.Brunner et.al. "Le radon dans les habitations", compte-rendu Journée Scientifique de la Société Suisse de Radiobiologie et Radiophysique, Lausanne Oct. 1982 p. 131

/3/ W.Burkart et.al.:"Matched Pair Analysis of the Influence of Energy Conservation on Indoor Radon Concentrations in Swiss Dwellings" EC Intern.Seminar on Indoor Exposures to Natural Radiation Capri, Oct. 1983

CALCULATION OF INDOOR EXTERNAL EXPOSURE DUE TO RADON AND ITS PROGENY

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INTRODUCTION

Radiation protection officials concerned with the potential health problems due to the presence of radon and its progeny inside buildings have devoted essentially all of their attention to exposures resulting from inhalation of these radionuclides. Little attention has been directed to the associated external exposures. The external exposure from the sources which are redistributed due to the emanation of radon within the walls and floors of a building, as well as the ground, and its subsequent escape into the air within the building, are considered. Estimated here, using a computer code, are the external exposure rates for each source distribution, taking into account the emanation of radon within the walls, its escape into the room air, and the plateout of radon progeny on the wall surfaces. Finally, the corresponding effective dose equivalents from these external sources are compared to those due to internal exposures from inhalation of radon and its progeny by the people within the room.

CALCULATIONS

The Monte Carlo code, REBEL-3, which was developed by L. Koblinger (Ko80), is used as an original program for our dose estimations. We obtained this code from the Radiation Shielding Information Center at the Oak Ridge National Laboratory. This code provides a methodology to calculate the specific indoor exposure rate and flux from the source uniformly distributed in the walls. However, the source distribution and source intensity in the original code have been modified in order to take account of the above-mentioned changes in the source distributions. Since Rn-222 emanation is all that is considered here, only the uranium series is dealt with as a source. Four kinds of source distributions are considered in the calculations. Source 1 relates to the non-liberated radon, its progeny and its precursor within the walls. They are assumed to have a uniform source distribution. Source 2 is assumed to have a non-uniform distribution within the walls, corresponding to the liberated radon and its progeny. Source 3 is assumed to be a uniform source, corresponding to the distribution of radon and its progeny within the air inside the building. Source 4 corresponds to the radon progeny as plated out on the wall surfaces from the indoor air.

ASSUMPTIONS

For representativeness in the calculations presented here, many parameters are chosen from the "standard" room in the work of L. Koblinger (Ko78). The following assumptions were made. That the room had a cross sectional area of 5mx4m; a height of 2.8m; that the walls and floors were made of SiO_2, with a density of 2.32g/cm^3 and a thickness of 20cm; and that the room had no windows or doors. The

receptor was assumed to be located one meter above the floor in the center of the room.

Source 1 deals with the non-liberated radon and its progeny as well as its precursor within the walls. It is assumed that 90% of the radon was non-liberated. The liberated radon will move through the wall and form a non-uniform distribution of radon and its progeny within the walls. For source 2, the porosity and effective diffusion coefficient for the concrete walls were assumed to be 10% and 2.0E-5cm^2/s (Cu76), respectively. The distributions and concentrations of radon and its progeny within the walls were calculated using the equation for radon diffusion within walls of Moeller (Mo76). For source 3, the exposure rates were normalized for a unit concentration of Rn-222 (1 pCi/l) within the room air. The radionuclides assumed to be suspended in the air were Rn-222, RaA, RaB and RaC. Their relative activity ratios were estimated to be 1.00, 0.901, 0.537 and 0.359 based on an assumed air exchange rate of 1 per hour, and that the plateout rate for RaA was ten times greater than those for RaB and RaC. Based on these assumptions, the concentrations of RaB and RaC are then 0.537, and 0.359 pCi/l and the equilibrium factor is 0.51. The concentrations of RaD and Ra-226 in the room air were assumed to be zero. Using these values, the radionuclides suspended in the room air will have a source intensity of 0.910 photon per Rn-222 disintegration per liter. For source 4, the concentrations of RaA, RaB, RaC and RaD were estimated considering the mass balance of radon progeny in the room to be at steady state. Under these conditions, the concentrations of RaB and RaC would be 1.67, and 1.96 pCi/cm^2, respectively. The concentration of RaD is conservatively taken to be the same as that of RaC. The intensity for the source was 4.16 photons per Rn-222 disintegration per liter.

RESULTS AND DISCUSSION

The specific exposure rates for each of the sources considered in this study are given in Table 1. All the statistical uncertainties of the Monte Carlo results are within 5%. The values

Table 1. Specific Exposure (μR/h) (for a single room of 5mx4mx2.8m)

	uncollided	scattered	total
source 1*	1.78E 0	1.74E 0	3.52E 0
source 2*	4.10E-2	5.28E-2	9.39E-2
source 3**	1.05E-2	1.08E-3	1.16E-2
source 4**	5.63E-4	1.59E-4	7.22E-4

* Unit activity is pCi of Ra-226 per gram of concrete.
** Unit activity is pCi of Rn-222 per liter of air.

for source 1 are about 91% of those that would have been calculated without considering the emanation of radon within the walls (named "original result"). The exposure rates for source 2 are less than 3% of that for source 1. For sources 3 and 4, the uncollided photons are much more significant than the scattered photons. This is due to the fact that many photons reach the receptor before being collided since the absorber between the receptor and the source is only air. If the air within the room is assumed to have a Rn-222 concentration of 100 pCi/l, the exposure rates for sources 3 and 4 are estimated to be 1.2 and 0.07 μR/h, respectively. If the cross sectional area of the room is assumed to be increased up to 20mx20m, keeping the height

at 2.8m, the volume of the room would increase to 1120m^3 but the exposure rates for sources 1 and 2 would show no significant change. For source 3, the exposure rate increase gradually with room size and reach 193% of the "standard" room for a room with a cross sectional area of 20mx20m. For source 4, the exposure rates normalized for unit activity on the wall surfaces increase gradually to 158% of those for the "standard" room.

It would be extremely difficult to confirm the calculations presented here through direct measurements. For source 1, the estimates are 91% of "the original results." They are presumably correct because 90% of the radon and 100% of its precursor, Ra-226 which has 1.8% of the total gamma intensity of uranium series, were dealt with as sources and the dose contribution would be expected to be a little higher than 90% of "the original result." For source 2, further verification may be necessary. For sources 3 and 4, there are no direct experimental results that can be used for comparison or evaluation. Calculations for infinite conditions, therefore, have been performed to compare with the estimates found in papers. The infinite condition can be satisfied with a room, of which size is greater than 1000mx1000mx1000m, in our calculations. Since our radionuclide source is not mono-energetic, an average source energy was obtained for comparison with the estimates for a mono-energy source. For source 3, of which average source energy is 775 keV, the results for the infinite condition were consistent (within 12%) with the values reported by Dillman (Di74) for uncollided and scattered photon fluxes for the source of 0.8 MeV photons. For source 4, of which average source energy is 858 keV, the exposure rate for the infinite condition were again consistent (within 13%) with the results reported by French (Fr65) for the source of 0.85 MeV photons.

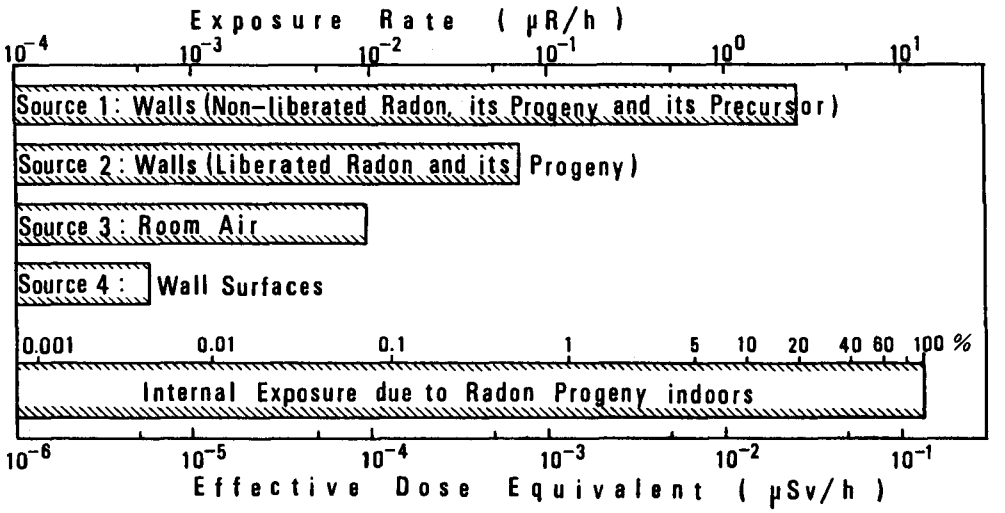

Fig. 1. Exposure Rates in "Normal" Condition in a Room of 5mx4mx2.8m.

Exposure rates for each source distribution for a "normal" situation are presented in Figure 1. "Normal" concentrations of Ra-226 in the concrete and Rn-222 in the air were assumed to be 0.75 pCi/g, and 0.81 pCi/l, respectively (UN82). Under these conditions, the contributions to the exposure rates from sources 3 and 4 are only 0.009 and 0.006 μR/h. Even in a very large room (cross sectional area of 20mx20m, and height of 2.8m), the exposure rates for sources 3 and 4 were only 0.02 and 0.002 μR/h, respectively. Most of the contribution to the external exposure rates arises through source 1. It must be acknowledged, however, that the exposure rates from this source are directly dependent upon the concentration of uranium and its decay products within the walls, including the rate at which radon is assumed to escape from the walls. Calculations based on the assumed conditions for source 2 show that most of the radon liberated within the walls escapes outside the walls. As a result, the exposure rates from source 2 are reduced to 25% of the total liberated radon and its progeny within the walls.

The effective dose equivalent from internal exposure due to inhalation of radon and its progeny by people who remain within the "normal" room, as described previously, would be 0.13 μSv/h. In the corresponding "normal" room, the effective dose equivalents due to external dose from sources 1, 2, 3, and 4 were estimated to be 20, 0.5, 0.07 and 0.004% of that from internal exposure (see Fig. 1).

CONCLUSIONS

A modified computer code has been developed which can be used to estimate indoor external exposure rates from the naturally occuring radionuclides in the walls and floor of a room which are redistributed due to the emanation of radon within the walls and floors of a building, as well as the ground. Accounted for in the code are also external exposures due to submersion within a radioactive "cloud" within the room as well as the plateout of radon progeny on the walls and floor. Applications of the code show that the relative contribution to external exposure rates from radionuclide sources arising through the emanation of radon within, and the escape of radon from, the walls of a room are quite small. The main contributors to external exposure rates are the radon and its progeny that remain within the walls of the room.

REFERENCES

Cu76 Culot,M.V.J., Olson,H.G. and Schiager,K.J., Health Phys. 30, 263 (1976).

Di74 Dillman,L.T., Health Phys. 27, 571 (1974).

Fr65 French,R.L., Health Phys. 11, 369 (1965).

Ko78 Koblinger,L., Health Phys. 34, 459 (1978).

Ko80 Koblinger,L., KFKI-1980-07, ISBN 963 371 630 6.

Mo76 Moeller,D.W., Underhill,D.W. and Gulezian,G.V., Natural Radiation Environment III, p.1424, CONF-780422(Vol.2) (1980).

UN82 United Nations Scientific Committee on the Effects of Ionizing Radiation, (New York,: United Nations) (1982).

STUDY OF THE RADIATION PROTECTION PROBLEM CAUSED BY THE PRESENCE OF 226-Ra IN BUILDING MATERIALS.

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Objectives

The inhalation of radon daughters is now generally considered as the major source of exposure to radiation of the general population. In some countries or regions, extremely high doses are received by a considerable number of inhabitants, exceeding even the annual dose limit for workers. This is not expected to be the case in Belgium where the soil activity is rather low (1). Still there was some concern about the widespread use of phosphogypsum and eventually other waste or byproducts (slags, flyash) in building materials. In combination with low ventilation rates, this could yield higher exposures, although probably below any action level. We feel that even low enhanced exposures should be justified in comparison to the differential cost of using more favourable materials, and that an appropriate legislation should be worked out. In view of this task a study of the contribution of building materials to the indoor exposure was started. This does not imply that we consider the soil to be a negligible source, nor that there are no other relevant parameters. On the contrary a parallel research project was started to study on the one hand the ventilation in houses, which affects the radon concentration in the first place, and the indoor aerosol on the other hand, which determines the equilibrium of radon daughters relative to radon (2). The results of such studies could allow to optimize the indoor climate in order to reduce the exposure to radon daughters (and to other pollutants) irrespective of the radon source. However we think that on the short term a more prudent use of specific building materials is more likely to be cost-effective.

Methods

The radon exhalation from building materials is obviously determined by their 226-Ra content. So the first step is to perform a γ-spectroscopic analysis of a sample in order to measure the specific activity of 226-Ra and other nuclides (40-K, 232-Th). However the radon exhalation must be measured directly since there is a wide variability of the emanating fraction of radon. The emanating fraction depends on the pore and grain structure of the material and on the moisture content. The exhalation rate of the investigated material is determined by enclosure of a sample in a hermetic chamber, filled with

radon-free nitrogen. After a few hours up to several days the emanated gas is accumulated on a charcoal trap and transferred to a Lucas scintillation cell. The exhalation rate of the sample is calculated from the measured activity divided by the time t after enclosing or by

$$[1-\exp(-\lambda t)]/\lambda t$$

for times comparable to the radon decay period (λ^{-1} = 3.82 days/ln2). For thick walls in particular the diffusion of radon to the surface must be taken into account, since possibly part of the emanated radon decays before reaching the surface. This requires the knowledge of the diffusion coefficient of the material. For a full description of the transport properties of the material, the porosity and the permeability (air flow in response to a pressure gradient) should be determined.

Results

Methods for measuring the specific activities of building materials and of their radon emanation and diffusion properties are currently in operation in our laboratory (3). The first results indicate that at present the most important material for Belgium is phosphogypsum, not only because of its rather high radium content but especially because its emanating fraction is much higher than for any other material (e.g. 0.1 - 2% for different bricks).

Table I: Specific 226-Ra content and radon emmanating fraction of some building materials.

Material		226-Ra (Bq/Kg)	Mass exhalation rate (Bq/Kg.s)	Emanating fraction (%)
Phosphogypsum		430	$1.2\ 10^{-4}$	14
Sulphogypsum		41		
Natural gypsum	blocks	<11	$6.3\ 10^{-7}$	
	board		$2.2\ 10^{-6}$	

For the effective diffusion coefficient D_e of phosphogypsum a mean value of 0.008 cm^2/s is found. The true porosity ϵ of the material has not yet been established but is close to and anyway smaller than unity so that the diffusion length $\sqrt{D_e/\epsilon\lambda}$ is larger than 60 cm. Fortunately phosphogypsum is currently applied only in plaster and plasterboard for wall finishing and no longer in the form of bricks for interior walls.

Future developments

A survey of the radionuclide content and of the radon exhalation from building materials will be performed on two levels:

a) representative samples of the materials used most extensively and covering nearly the entire Belgian building market.

b) the materials suspected to contain enhanced levels of 226-Ra mainly due to the application of identified active byproducts.

These studies will be performed with the cooperation of the Belgian research institute for the building industry (WTCB). More extensive measurements will be performed on identified radioactive materials from the second survey. The true exhalation rate will be measured on erected wall or floor structures. The sealing power of coatings will be studied. The effect of pressure gradients, humidity, temperature and moisture content of the material could be investigated.

A pilot survey of the radon concentration in Belgian houses is in course (4). A passive track-etch radon detector and a thermoluminescent dosimeter have been distributed in about hundred houses covering the Belgian territory. The results will be available in a near future. This survey will be repeated on a larger scale and with respect to the rules for representative sampling. The data from the survey (a) will be linked to the survey of building materials in order to verify a relationship between the radon exhalation rate of the building materials and the indoor radon concentrations.

The enhancement of the radon concentration by existing or new radioactive materials can be estimated for a reference value of the ventilation rate. The justification for the use of a given material should be based on two considerations:

- the magnitude of the contribution relative to the concentrations in houses with "traditional" materials.
- the increased risk for lung cancer induction for a given concentration increment, based on dosimetric models and the accepted radiological risk factor per unit dose equivalent. The dosimetry requires the knowledge of the radon daugher equilibrium and the physical properties of the daughters (e.g. the fraction not attached to aerosols).

The essential question in this justification problem is which cost could be assigned to the detriment. The problem is hardly comparable to the radiation protection of workers or the public in the vicinity of nuclear installations. One must be careful not to impose restrictions in view of the radon problem that would be excessive in comparison to other measures such as building codes affecting the quality of life.

References

(1) W. Slegers, L. Ghoos, Metingen en resultaten van straling in de omgeving. An. Belg. Ver. voor radioprotectie vol. 8, n° 2 (1983).
(2) F. Raes, A. Janssens, A. Declercq, H. Vanmarcke, Investigation of the indoor aerosol with an automated mobility aerosol spectrometer and its effect on the attachment of radon daughters, Proc. Int. Seminar on indoor exposure to natural radiation and related risk assessment, ENEA-CEC, Anacapri Oct 3-5 1983.
(3) A. Poffijn, R. Bourgoignie, R. Marijns, J. Uyttenhove, Laboratory measurements of radon exhalation and diffusion, ibidem.
(4) J. Uyttenhove, R. Marijns, A. Janssens, H. Vanmarcke, R. Jacobs, Survey on natural radiation in houses in Belgium, ibidem.

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On the Radioactivity of Building Materials

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Abstract

The specific activities of natural radionuclides of various main building materials used in Greece have been measured. Considering these as models, the radiation exposure from gamma rays and radon was evaluated. It was estimated that concentrations of 8pCi/g of ^{226}Ra, 6 pCi/g of ^{232}Th and 85 pCi/g of ^{40}K in building materials will give an increase of 50 mrad/y in gonad dose if only one of the above radionuclides is present. Health effects of radon are also discussed.

1.Introduction

It is well known that many building materials used in the construction of dwellings contain radioactive elements {Hu56},{Kr71},{St76},{OR77},{UN77},{Ko78}, {OE79},{Za80}. The radionuclides which are present in them belong to the natural radioactive elements, such as uranium-238, radium-226, thorium-232 and their decay products, as well as potassium-40. The radiological implications from the above nuclides are due to irradiation of the body by gamma rays and irradiation of the lung tissues from inhalation of radon-222 and its daughters.

2.Experimental Procedures and Results

We have picked and measured a lot of samples of various building materials used in Greece, such as, red bricks, cement, concretes, tiles, gypsum etc . The sampling was made from Thessaloniki (North Greece), Valey of Tembi and Volos (Central Greece) and Tanagra, Chalkida, Etolikon (South Greece), see Table 1 .

The measurements of radioactivity in the building materials were performed by several methods, as: direct gamma spectroscopy using Ge-Li and intrinsinc Ge detectors, neutron activation analysis and delayed neutron technic.

The results of our measurements are summarized in Table 1. In this Table the specific activities of ^{226}Ra, ^{232}Th and ^{40}K are presented. The absorbed dose rate in air was calculated assuming 4π-geometry and infinite thickness (a factor 2) , and it is presented in Table 1, last column. These values are an index allowing the comparison between the building materials.

Polyurethane, which is used in the present days, as thermoisolating material in Building Technology,does not show any radioactivity (below detection level) . Wood was not considered in this work, since its low level radioactivity {UN77}. However, mean values were: ^{226}Ra and ^{232}Th below detection level, ($2x10^{-2}$pCi/g and $1x10^{-2}$ pCi/g, respectively), ^{40}K = 1.15 pCi/g and ^{137}Cs = $1.4x10^{-2}$ pCi/g.

3. Exposure to Gamma-Rays

The increase of gonad dose from building materials may be found from the following formula by subtracting the annual mean outdoor dose, ≈36 mrad {Kr71}.

$$\Delta D_y = 11.05\ C_{Ra} + 14.6\ C_{Th} + 1.0\ C_K - 36 \text{ mrad} \quad (3.1)$$

where: C_{Ra}, C_{Th} and C_K are the radioactive contents of building materials (measured in pCi/g) of ^{226}Ra, ^{232}Th and ^{40}K, respectively.

In Table 2, the anual increase of gonad dose, ΔD_y, in dwellings, which would be built with high radioactive building materials examined, is presented. For the red brick BM-2, as well as for the tiles BM-4 and BM-7, ΔD_y is of the order of 25 mrads, while for the concrete BM-17, ΔD_y is negative. $\Delta D_y \approx 25$ mrads is about half of the ICRP recommended for the population {IC77}.

If only one of the radionuclides considered is present, a 50 mrad/year increase is caused in gonad dose with the following specific activities of building materials: 8pCi/g, 6 pCi/g and 85 pCi/g for ^{226}Ra, ^{232}Th and ^{40}K, respectively. Then, if all the above nuclides are present, their concentrations must fulfill the following expression:

$$\frac{C_{Ra}}{8} + \frac{C_{Th}}{6} + \frac{C_K}{85} \leq 1 \quad (3.2)$$

The denominators of (3.2) are slightly different for the same criterion suggested by Krisiuk et al {Kr71} and Stranden {St76}. This is due to the difference existing in models considered by Krisiuk et al (a hole in an infinite medium) and Zastawny (typical cubicoid room).

The sum of the three quotients of the formula (3.2) for the building materials examined, gives the higher values for the red brick, 0.73 (sample BM-2) and for the tiles, 0.70 (samples BM-4 and BM-7).

Measurements of the dose rate were made in modern, as also, in old dwellings. The measurements were performed by a Scintillometer NaI(Tl) 1"x1".5 crystal and a Cutie-Pie Survey meter with an ionization chamber of 580cc (2"7/8x5"1/2). About 50 houses were picked at the wide area of Thessaloniki to measure the dose rate in bedrooms and living rooms. A dose rate of 3.6 μR/hr for the cosmic rays {UN77}, were subtracted from each measurement.

The results are presented in Table 3. The Table 3 shows the mean as well as the minimum and maximum dose exposure that were measured. We observe that the maximum values of increase of gonad dose, are at the same level as those of Table 2.

4.Lung Dose Exposure from Radon and Radon Daughters and Health Effects of Radon.

Health effects of radon are expressed in terms of WLM (Working Level Months). Considering an occupancy factor of 0.8 indoors,{UN77},the annual time of exposure is 41 working months {Co80}. In Table 4, the annual indoor exposure, in WLM, is given in the last column, for a typical room 3.5x3.5x3.0 m^3 (A/V=1.81) with a wall of half thickness 15 cm, built by brick BM-2 (ρ=1.96 g/cm^3) or concrete BM-17 (ρ-2.35 g/cm^3), which are the most contributing in radon. In Fig.1 typical curves of annual lung dose exposure indoors for various ^{226}Ra concentrations (from 0.5 to 3 pCi/g) in bricks as a function of air rates are presented. The curves for concrete buildings have the same behavior and are slightly higher.

5·Conclusions

According to the ALARA (As Low As is Reasonably Achievable) Principle that the dose exposure should be as low as possible, and the above results and the discussion, we can suggest that the radioactive concentrations of the nuclides considered in the building materials should not exceed the values of 1.25, 0.5 and 15 pCi/g for ^{226}Ra, ^{232}Th and ^{40}K, respectively. These values give a sum of the quotients of the criterion, formula (3.2), equal to 0.4, i.e 40% lower that it was suggested in section 3. It is corresponding to an annual increase of gonad dose equal to 20 mrads.

Acknowledgment - We wish to thank the Commission of the European Communities for the support partly this work.

BEHAVIOUR OF TRITIUM UNDER ENVIRONMENTAL NON-EQUILIBRIUM CONDITIONS

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INDRODUCTION

Calculations of the ingestion dose from tritium contaminated foodstuffs in conformity with the guidelines applicable in the Federal Republic of Germany /1/ are based on conditions of radioecological equilibrium and constant rates of discharge from nuclear facilities using the specific activity model. This gives values convenient for work, although modifications to the specific activity model have been suggested because in some cases it may furnish underestimates /2/. With a view to the high tritium inventories of future reprocessing plants and installations for fusion experiments detailed knowledge is required of the regional distribution and behaviour, and major discharges resulting from an accident must be taken into account as well. Above all if account is taken of the organically bound tritium (OBT) in addition to the tritium bound in water (HTO), the static models are not suited. Therefore, besides the routine environmental monitoring programme, a systematic long-term study is being performed at the Karlsruhe Nuclear Research Centre on the consequences on a tree of the environmental impact of tritium discharges under realistic conditions.

METHODS

A beech tree (Fagus sylvatica) was selected for the investigation. Sample collection and recording of climatological data of measurement have been described earlier /3/. For the tritium measurement in the liquid-scintillation spectrometer the leaves were freeze-dried and plasma-ashed /4/. Therefore, all the tritium concentrations are expressed in mBq/ml of water and oxidation water, respectively. The detection limit for 100 minutes of measuring time and 10 ml of sample volume lies at approx. 5 mBq/ml.

RESULTS

At the location of the tree the HTO concentrations in the air humidity collected continuously were determined with a high resolution in terms of time (usually every three hours); they were also determined in the individual precipitations and, finally, in the moisture of the soil down to 1 m depth, i.e. in the region of root growth. The HTO values in the air humidity take an extremely variable course with temporary peaks above 500 mBq/ml, quite a number of values between 50 and 300 Bq/ml, and the majority of concentrations occurring in the range from 10 to 50 mBq/ml. Both in the summer 1981 and in the summer 1982 periods can be distinguished with lower values when the emissions due to operating installations are clearly reduced because of the holiday time. The HTO values of the soil water which were determined in a relatively close succession in 1982 exhibit, on the whole, a rather regular course although the introductions into the soil by rainwater differed considerably. A peak value of 430 mBq/ml was observed only once. In Table 1 the arithmetic means have been copiled of the concentrations measured during the 1982 vegetations period. The precipitations represent a mean value weighted by the amount. The tritium concentrations measured in the leaves (HTO and OBT) are shown in detail in Figures 1 and 2.

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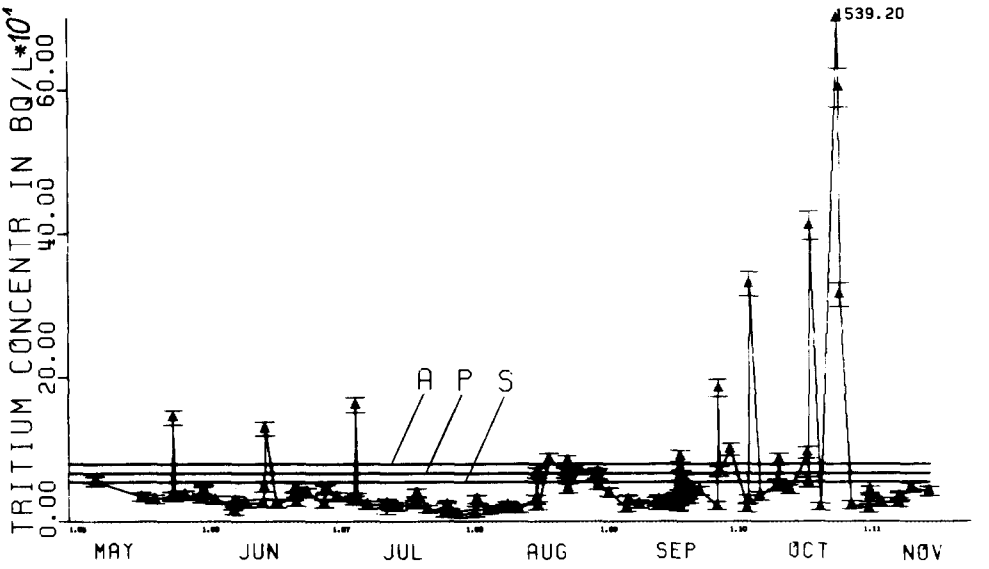

Fig. 1: HTO-concentrations measured in leaves during the 1982 vegetation period in comparison to mean HTO-concentrations in the humidity of the air (A), precipitations (P), and soil (S).

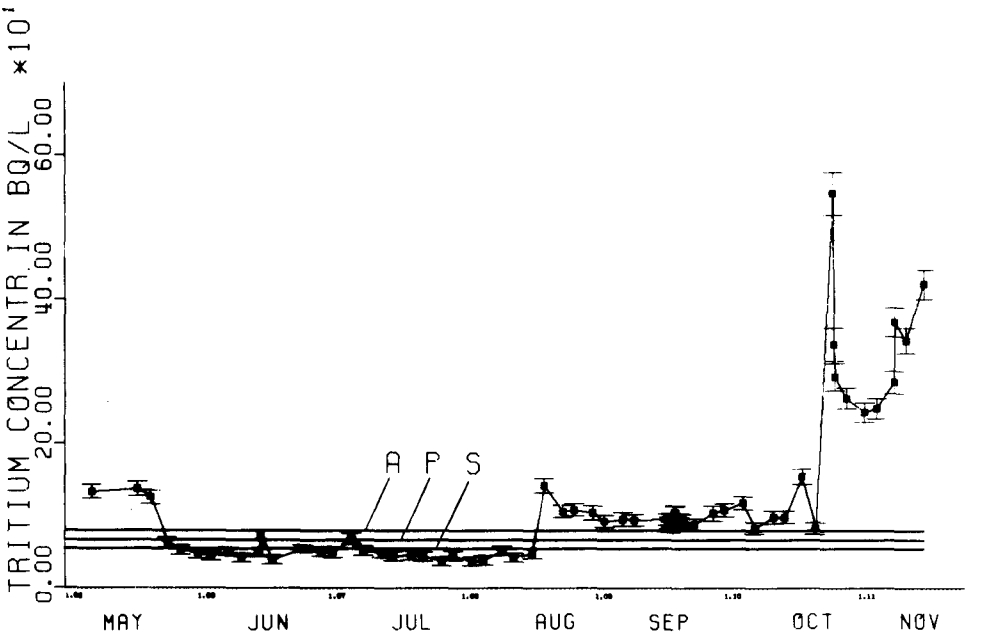

Fig. 2: OBT-concentrations measured in leaves during the 1982 vegetation period in comparison to mean HTO-concentrations in the humidity of the air (A), precipitations (P), and soil (S).

Table 1: Mean tritium concentrations during the 1982 vegetation period

	mBq/ml	(Number of measured values)
Air humidity	79	(1601)
Soil water	54	(142)
Precipitations	66	(55)
Leaves (HTO)	62	(131)
(OBT)	120	(68)

DISCUSSION

The measured values of environmental impact clearly show that the plants growing in the vicinity of the Karlsruhe Nuclear Research Centre are exposed under distinctly dynamic conditions because of variations in atmospheric discharges and varying wind directions. This, combined with a very quick equalization of concentrations via the stomata, causes temporary increases in HTO concentrations in the leaves during the day time. This is why an increased HTO value can be found with some probability only, if leaf samples are collected at random, and this applies to relatively frequent sampling as well. So, if individual samples are taken, e.g. ready-to-harvest green vegetables, the evidence of such HTO measured values is only poor.

The long-term study, in addition, makes evident quite well the behaviour of the OBT in the leaves. Contrary to the HTO-values, the dynamics of the OBT-values are greatly damped although they reflect the history of exposure. The tritium of the leaf water is incorporated in the organic substance due to photosynthesis or isotopic exchange with the concentration at the respective time and part of it is retained until the end of the vegetation period. Therefore, the specific OBT concentrations found in the samples are nearly always higher than the HTO concentration which means that the OBT to HTO ratio is greater than unity. However, this does not reflect enrichment or accumulation due to an isotope effect but only a retention effect for OBT under non-equilibrium conditions.

In our own studies these measured values serve to verify a dynamic model presently elaborated and the input parameters used. With the help of microclimatic data of measurement continuously recorded in which also the differences between day and night are taken into account, the tritium concentrations found in the leaves have been explained by this model for some periods /5/. Only by such a description mapping with high accuracy the actual situation it will be possible to make a statement on the extent to which estimates of the ingestion dose using conventional concepts based on equilibrium conditions will produce overestimated or underestimated results under realistic conditions.

REFERENCES

/1/ BUNDESMINISTER DES INNERN,
Allgemeine Berechnungsgrundlagen für die Bestimmung der Strahlenexposition durch Emission radioaktiver Stoffe mit der Abluft, Empfehlung der Strahlenschutzkommission,
Der Bundesminister des Innern (Okt. 1977).

/2/ Vogt, K.J.,
Modells for assessing the environmental exposure by tritium released from nuclear installations, in "Behaviour of tritium in the environment",
IAEA, Vienna (1979), p. 21-534.

/3/ Strack, S.,
Behaviour of tritium in the water pool and organic pool of the leaves of a beech tree,
Annales de l'Association Belge de Radioprotection, Vol. 7, 3-4 (1982), p. 213-227.

/4/ Strack, S., L.A. König,
Determination of organically bound tritium in environmental samples by application of the oxidizing plasma technique,
KfK-Report 3249 (1981).

/5/ Strack, S., L.A. König,
Untersuchungen des dynamischen Verhaltens von Tritium in Pflanzen nach Kurzzeitabgaben von tritiiertem Wasserdampf in die Atmosphäre,
European Seminar (CEC) on the transfer of radioactive materials in the terrestrial environment subsequent to an accidental release to atmosphere, Dublin, April 11-15 (1983).

INFLUENCE OF RADIONUCLIDE RESIDENCE TIME IN SOIL AND OF COMPETING ALKALINE EARTH ELEMENTS ON RADIUM UPTAKE BY EDIBLE PLANTS

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The uptake of radium by plants and its dependence on the residence time as well as on the concentration of other alkaline earth elements was studied with soil contaminated several decades ago. Only the use of a well weathered radium containing soil provides a realistic model for transfer factor determinations since the bioavailability of radium bound to soil particles is lower than in freshly prepared mixtures. The solubility of radium may depend on different sorption either on the surface or inside soil particles of various sizes or on chemical changes of the radium depending on its environment, such as the varying composition of other alkaline earth elements in the soil which may influence the uptake of radium. Although it behaves chemically different to the other members, such as calcium or magnesium both of which exert important functions in biological systems, it may still compete with these for the same binding sites (1).

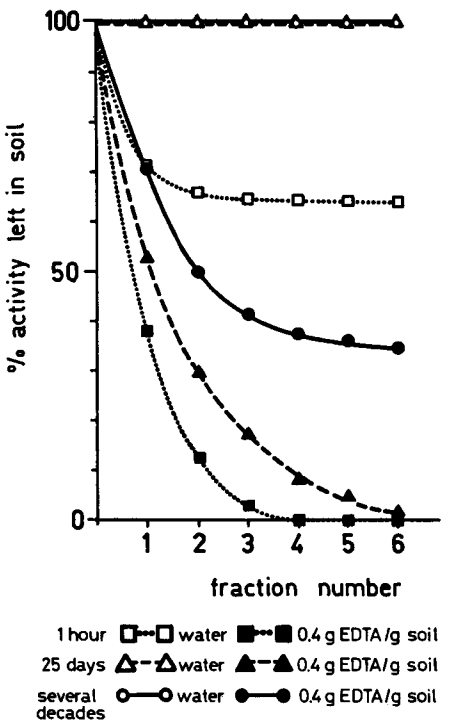
The soil samples, cultivation of the plants and the measuring techniques are described elsewhere (2). The plants investigated in this study are: tomatoes, cress, dandelions, radishes and mangold; they were chosen for their different behaviour towards alkaline earth elements, such as calcium. The contents of some elements taken up by the plants, especially the alkaline earths, were determined in their ashes by atomic absorption spectrometry.

One of the most important parameters which determine transfer factors between soil and plant is the bioavailability of a radionuclide which in turn depends on its chemical and physical forms which in the case of actinides, which are important in such considerations, are particles. Other important factors are sorption processes on the surface or deposits in pores throughout the soil particles leading to very slow exchange rates, depending on the matrix and the size of the particles involved. Figure 1 illustrates the dependence of the extractability of radium from soil on the residence time. Whereas 30 % of freshly deposited radium (1 hour storage) can be eluted with water, all of it can be removed by EDTA (ethylene diamine tetra acetic acid). After storage times of longer than one day none of the radium is dissolved in the water extract, but all the radium can be removed by a solution containing 0,1 g EDTA/g soil (3). Soil contaminated many years ago however cannot be decontaminated completely even with six treatments using a high concentration of 0,4 g EDTA/g soil. This treatment with EDTA effectively reduces the concentration of the radium available to plants as determined by extraction with ammonium acetate (4) (Table I).

For control soil containing only environmental levels of radium this reduction is higher. Plants (cress) grown on previously treated soil with EDTA show increased radium contents compared with untreated controls. This is probably due to a more thorough removal of calcium and magnesium leading to a higher radium to calcium ratio. Competition

between alkaline earth elements and radium was investigated as follows: different amounts of calcium were added in the form of very finely dispersed calcium carbonate to the soil, barium and strontium chlorides were added as solutions to form a slurry, the soil was then dried in warm air. Subsequent cultivation of plants on these soil-mixtures showed a clear trend to lower radium incorporation into the plant material with increasing alkaline earth content (figure 2). This effect can also be observed when the transfer coefficients for radium for untreated T soil are compared with those obtained with control W soil. The larger amount of bioavailable calcium in T soil results in relatively low transfer factors compared to those obtained with low calcium W soil. Both types of plants with the low and high calcium contents show this behaviour.

Fig. 1. Extraction of Ra-226 from contaminated soil with different concentrations of EDTA after different times



Transfer coefficient determinations between soil and plant for critical radionuclides produced or introduced into the biosphere during the nuclear fuel cycle are very important when the risks from nuclear waste repositories have to be assessed. Most of the scenarios for long term underground storage of nuclear waste assume that the conditioned waste is leached by ground water and that some of the inventory is brought to the earth's surface in this way. From the large number of possible exposure paths the ingestion through drinking water or via the food chain poses the biggest health hazard.

Literature values for the transfer from soil to plant of radionuclides are usually only subdivided into leaf vegetables and root vegetables irrespective of whether plants have a high or a low alkaline earth content and they show a considerable variation. Some more recent publications by Schüttelkopf and Kiefer (5) and by Tracy et al (6) list individual values for some vegetables and animal products.

A direct comparison of the results of different groups cannot usually be made since experimental conditions are not defined for all significant parameters and the plant species show too much variation. An example for this statement is that no distinction is made between

Table I Reduction of availability of radium after EDTA treatment

		T soil Ra^{++} (Bq/100g)	W soil Ra^{++} (Bq/100g)
Untreated soil:			
total Ra^{++} content		3970	4,78
available Ra^{++}		625,67 (16%)	0,27 (6%)
cress		7,56	0,15
transfer coefficient	1)	0,012	0,54
	2)	0,0019	0,031
Soil treated with EDTA (0,45 g/g soil):			
total Ra^{++} content		1640	4,54
available Ra^{++}		133,20 (8%)	0,04 (0,9%)
cress		4,64	0,21
transfer coefficient	1)	0,034	5,18
	2)	0,0028	0,046

1) based on available radium in soil
2) based on total radium in soil

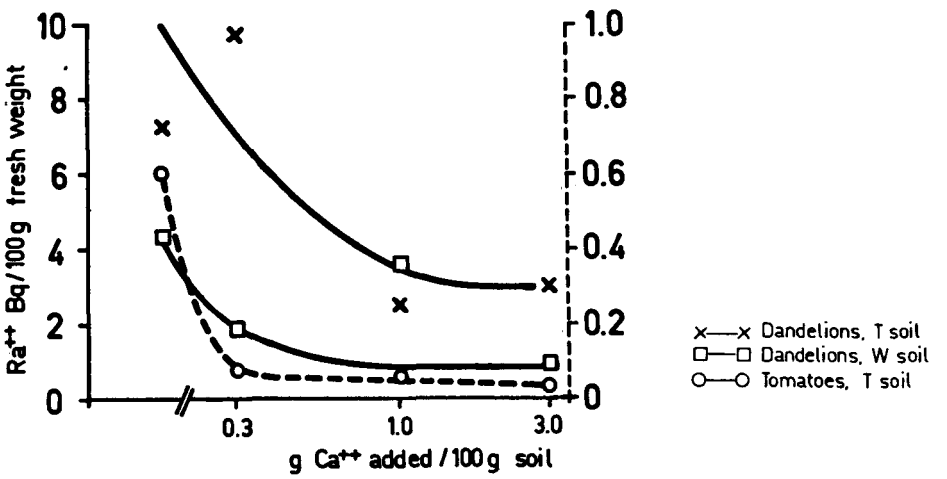
total radium content and bioavailable radium content of the soil, as determined by standard agricultural methods. Our results show that the radium uptake by dandelions and tomatoes vary greatly with the calcium content of the soil. When using soil with environmental radium contents, the radium impurity of the added alkaline earth elements has to be taken into account.

The Swedish group of Eriksson (7) carried out very extensive studies on the uptake of radium and other radionuclides taking into account differences in soil composition and structure, pH, as well as other regional differences. This Swedish study, Kirchmann's hydroculture model (1) and our experiments show that the alkaline earth elements compete for uptake by plants from the soil. The possible active transport mechanism and the involvement of specific cation carriers have not yet been characterized.

Further studies are needed to understand the solubility and the extractability of radium from soil as a function of the storage time. Our results so far obtained demonstrate that this is not an instantaneous process and that a definite "ageing" must occur. It is not yet clear whether the decrease in availability is due to the formation of some insoluble radium compounds or to the absorption of the radium inside the soil particles which act like a three dimensional ion exchange resin. This would lead to a new radium compartment with no response to

treatment with chelating agents and possibly very low exchange rates.

Fig. 2 Reduction of radium transfer into plants by calcium addition to the soil



Regarding the uncertainties outlined above it is not surprising that literature values for radium transfer vary over more than four orders of magnitude. Considering the multitude of soil parameters and their possible effects on the transfer of radium from soil to plant may lead to a better understanding of this problem and of the fundamental uptake mechanism of plants on the molecular level. This may allow more reliable predictions to be made.

Literature:
(1) KIRCHMANN, R., RONCUCCI, R., MOUSNY, J., "Isotopes and Radiation in Soil-Plant Nutrition Studies", IAEA Vienna (1965) 277
(2) KOPP, P., BURKART, W., GOERLICH, W.,Proc. 7th Int. Congr. Radiat. Res., Amsterdam (1983) E4-03
(3) BURKART, W., KOPP, P., GOERLICH, W., Proc. Seminar on the Environmental Transfer to Man of Radionuclides Released from Nuclear Installations, Brussels (1983) IAEA -SR-85/41
(4) ANDERSON, A., Swedish J. Agric. Res. 5, (1975) 125
(5) SCHUETTELKOPF, H., KIEFER, H., KfK 3367, Kernforschungszentrum Karlsruhe (1982)
(6) TRACY, B.L., PRANTL, F.A., QUINN, J.M., Health Phys. 44, (1983) 469
(7) ERIKSSON, Å., FREDERIKSSON, L., "Naturlig radioaktivitet i mark och grödor. Report SLU-IRB-52, University of Agricultural Sciences, Uppsala (1981)

A SYSTEMATIC INVESTIGATION ON THE EFFECTS OF THE MAJOR CHEMICAL AND CHEMICAL-PHYSICAL PARAMETERS ON Cs AND Sr DISTRIBUTION COEFFICIENTS IN A FRESH WATER SAMPLE.

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INTRODUCTION

The aim of this work was at determining the distribution coefficients (Kd, cm^3/g) of Cs and Sr and the evaluation in a fresh water sample of the major chemical and chemical-physical parameters that effect them.

There are lots of Kds in the literature which have been determined in different samples (sediments, soil, clay, silt, sand and synthetic materials), with different analitycal procedures and method. Obviously, these results should not be undiscriminatingly used in radionuclide transport studies using mathematical models to predict the environmental impact of conventional and non-conventional discharges.

It is quite simple to think or imagine that many variables or parameters may change the Kd values so that it could be extremely useful to investigate and understand this phenomenon.

It is obvious that the best sample for the investigation is a real sample. At the same time the investigating procedure should not interfere with the natural equilibria involved.

Experiments were carried out on a fresh water sample gathered from river Po making use of scintillation counting as the analytical method.

EXPERIMENTAL

Special care was taken in the collection, transportation and conservation of the water sample.

Because about twenty days were necessary for the lab-investigation, the stability of the sample was considered fundamental.

The sample, after the collection, was characterized from the chemical, mineralogical and granulometric viewpoints.

The stability, during the lab-activity was checked following granulometric distribution, some chemical parameters and the Kd values. The results were satisfactory so that all the Kds can be considered reliable.

^{137}Cs and ^{85}Sr were employed as radioactive indicators added to the 250 ml homogeneus samples in a quantity less than the natural concentration ($\simeq$ 0.015 ppb and 300 ppb respectively) just not to modify the natural equilibrium.

After addition the polyethylene capped bottles were positioned horizontally on a mechanical shaker in controlled temperature. Shaking frequency was 70÷90 cycle/minute.

On the basis of preliminary kinetic investigations the time necessary for reaching the equilibrium conditions was 3h and 30' for Cs and 8h for Sr.

The analytical determination was made with the scintillation counting technique in the soluble phase, obtained through a 0.45 μm Millipore filter.

All the procedure mentioned above was set up previously and when it was ready we made the research reported in this paper so that it will be easy to repeat the experiments and to compare the results.

The major parameters affecting Kd values and the related interactions between them was found out in a preliminary work in which we employed a factorial ANOVA (analysis of variance) model.

Kd values were determined in triplicate by batch procedure in polyethylene sealed bottles in controlled temperature.

The distribution coefficient was calculated from:

$$Kd = \frac{\frac{A_o - A_f}{P}}{A_f - B} \cdot 10^6$$

where A_o is the initial activity, A_f is the final activity in the water phase (< 0.45 um), P is the particulate load (mg/l), and B is the blank activity.

The parameters investigated were:

for Cs: carrier - on the untreated sample and on the separated granulometric fractions - temperature, pH and interaction pH-carrier, particulate load, granulometric fractions, TOC, clay concentration, UV radiations, freezing.

for Sr: effect of pH, temperature, carrier and related interactions, granulometric fractions, TOC, clay concentration, UV radiations.

RESULTS

The results of Kd were elaborated with ANOVA in order to find out the main effects and their interactions.

For Cs, effects: particulate, carrier, temperature and pH; interactions: pH-carrier and temperature-carrier.

For Sr, effects: particulate, temperature, carrier and pH; interactions: pH-concentration.

The relative residual variance, interpreted as an estimate of the general process error, is about 10%.

For Cs, Fig. 1 shows some typical dependence of Kd on the main factors: pH, temperature, Q_T (total amount of Cs in nano equivalents/l), P (particulate load mg/l), and in the following table some analytical dependence functions are shown.

TABLE

For Cs	For Sr
$Kd = 1.15 \cdot 10^4 \cdot Q_T^{(-0.408)}$	$Kd = 4.78 \cdot 10^4 \cdot P^{(-0.89)}$
$Kd = 2.45 \cdot 10^4 \cdot P^{(-0.230)}$	$Kd = 6.59 \cdot 10^3 \cdot d_m^{(-1.076)}$

where Q_T = total concentration of the element in nano equivalents/l
P = particulate load in mg/l
d_m = mean diameter of particulate fractions in μm.

The other results will be reported in the final paper. Also kinetic results are available for Cs absorption obtained with a proper equipment set up for the purpose.

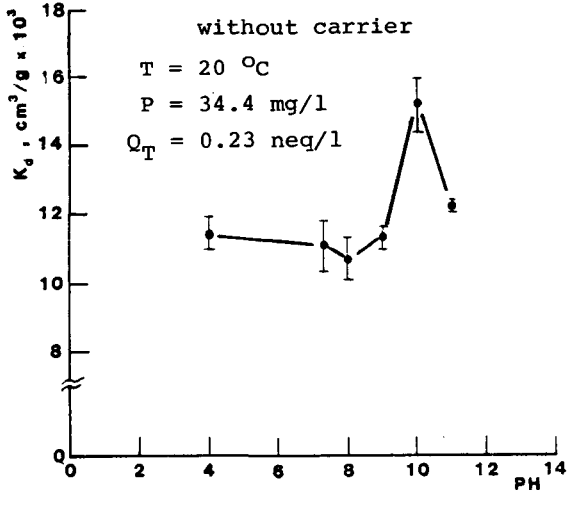
COMMENTS

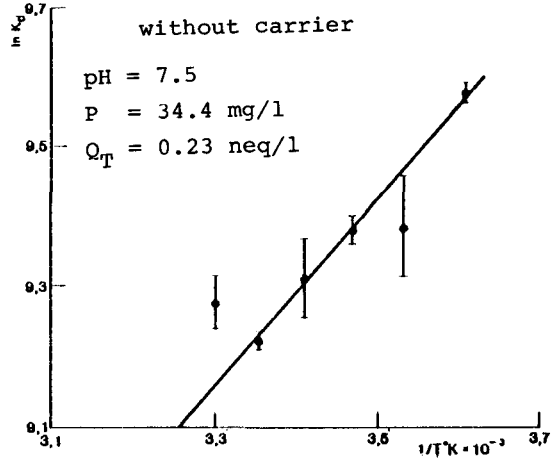
Considering the results obtained, the procedure adopted is recommended. At the same time as it is not complicated, it could be improved and standardized if the need for reliable Kds is considered necessary. Moreover, other interesting elements could be investigated.

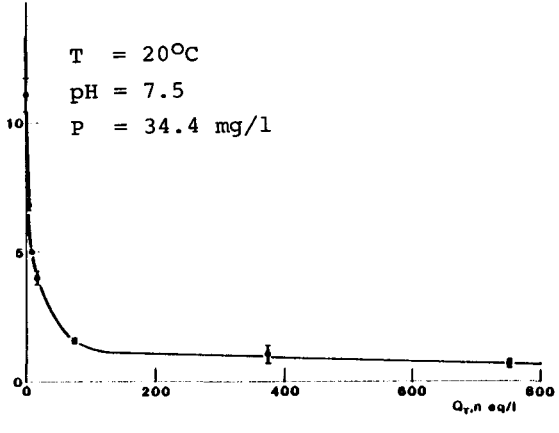
Such activity is under way in our laboratories.

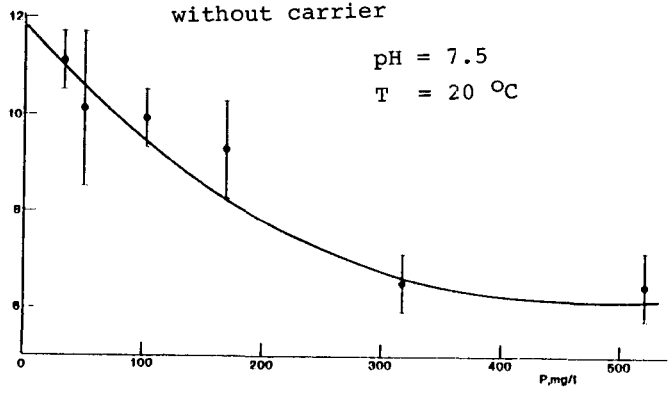
At the same time, because of the fact that a great quantity of Kds are available, coming from different samples and analytical procedures, we distinguish them as "natural", "isotopic", "mass" and "from leaching". This distinction is necessary for understanding the phenomena and for using Kd more correctly.

Fig. 1


without carrier
T = 20 °C
P = 34.4 mg/l
Q_T = 0.23 neq/l
K_d , $cm^3/g \times 10^3$
PH


without carrier
pH = 7.5
P = 34.4 mg/l
Q_T = 0.23 neq/l
ln K_d
$1/T°K \cdot 10^{-3}$


T = 20°C
pH = 7.5
P = 34.4 mg/l
Q_T,n eq/l


without carrier
pH = 7.5
T = 20 °C
P,mg/l

DEPOSITION VELOCITY AND WASHOUT COEFFICIENT FOR RADIONUCLIDES BOUND AT AEROSOL PARTICLES

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A part of the radionuclides that are emitted from nuclear facilities and others at normal operation or incidents as well as accidents, respectively, is bound at aerosol particles. The activity distribution of the emitted particle ranges of size is - in dependence of the filtering and operation method of the facility - in the diameter range of $d = 10^{-3} - 30$ µm and varies with time. Therefore, data for the calculation of the dry and wet deposition are required for this diameter range.

The aerosol particles that are emitted into the atmosphere are distributed due to turbulent diffusion. If they reach the vegetation or the ground surface, they are captured on the vegetation element or the ground surface due to inertia effects and Brownian diffusion, see fig. 1. Dry deposition exists, if aerosols remain attached after deposition due to electrostatic, Van der Waals and other forces. In the case of washout, the aerosols are trapped by the falling raindrop due to the same phenomena, mentioned above, see fig. 1, and reach the vegetation and the ground surface with the raindrop. In the ideal case, only aerosol particles which are in the residual moisture after the rain, remain on the above ground plant parts.

Physically the capture of aerosol particles at above ground plant parts and at raindrops can be described in the same way as the capture of aerosol particles on single fibres of fibre filters. The number of particles or the activity, respectively, that is deposited on the vegetation element per unit of time, is

$$\dot{\varphi}(d) = \int_0^h C(z,d) \cdot \bar{u}(z) \cdot f \cdot X_f(z) \cdot E(z,d)\, dz \qquad (1)$$

In this equation $\bar{u}(z)$ is the mean flow velocity in the height z, C(z,d) is the particle- or activity concentration with the particle diameter d, E(z,d) is the deposition rate, and $f \cdot X_f(z)dz$ is the vegetation surface in dz per unit ground surface. It is usual to describe the number of deposited particles with the deposition velocity $v_g(d) = \dot{\varphi}(d)/C(z_1,d)$. Equation (1) can be rewritten as follows, using the particle or activity concentration in the reference height z_1, which can be calculated from the product of the emission rate $\dot{E}(d)$ and the distribution factor $X(z_1)$:

$$\dot{\varphi}(d) = \dot{E}(d) \cdot X(z_1) \cdot v_g(d) = C(z_1,d) \cdot \frac{\dot{\varphi}(d)}{C(z_1,d)} \qquad (2)$$

When calculating the deposition according to this equation, it has to be taken into consideration that the curve of the particle or activity concentration in or above the vegetation must correspond not only to the curve of the activity, where the deposition velocity $v_g(d)$ has been determined experimentally, but also to the reference height z_1. Fig. 2 shows e.g. the onedimensionally calculated concentration curve in or above the vegetation /1/. Sometimes the experimental set-up causes concentration profiles different from those for emissions in great heights. Under such conditions the v_g-values have to be corrected with the concentration ratio C (vegetation)/$C(z_1)$ in order to obtain effective deposition velocities.

In /1/, /2/, /3/ the experimentally determined deposition velocities are compared with those theoretically calculated. A satisfying agreement can be stated, if differentiating between the kind of vegetation, the density of the particle material, the humidity of the atmosphere and the moisture of the surface. For grass, fig. 3 shows e.g. the good agreement between the theoretical curve and the experimental results for the used particle density range of $\rho_p = 1 - 3.6$ g/cm^3 in the experiments. The change of the

curves for small particle diameters below approx. 10^{-2} µm is caused by the high concentration decrease in the vegetation.

The deposition on the ground can be described with the transport velocity due to turbulent diffusion $v_{g,tur} = u_*^2/\bar{u}(z_1)$ and the sedimentation velocity $v_S(d)$ /1/.

$$v_{g,B}(d) = \frac{u_*^2}{\bar{u}(z)} \cdot E_B(d, u_*) + v_s(d) \tag{3}$$

$E_B(d,u_*)$ is the deposition rate on the ground. Here, the agreement between measurement and theory is satisfying, too /2/, /3/.

The trapping of aerosol particles on raindrops can be described in the same way as the capture on above ground plant parts, according to equation (1). The number of particles or the activity, respectively, that is transported to the ground with the precipitation intensity I per unit of time is:

$$\dot{\Psi}(d,I) = \int_0^\infty C(z,d) \cdot \Lambda(d,I)\, dz \tag{4}$$

with the washout coefficient $\Lambda(d,I)$ which is defined as follows:

$$\Lambda(d,I) = \int_0^\infty \frac{\pi X^2}{4} \cdot v(X) \cdot N(X,I) \cdot E(d,X)\, d(X) \tag{5}$$

v(x) is the terminal velocity of the raindrops with the diameter X and N(X,I) is the raindrop density distribution.

Fig. 4 shows the calculated curve of the washout coefficient for two rain intensities and various collection efficiencies due to impaction. The agreement with the experimental results is actually not jet satisfying.

/1/ Bonka, H., Horn, M.:
Review on the dry and wet deposition of aerosol particles.
CEC Seminar, Dublin 4(1983)

/2/ Bonka, H. et al.:
Weiterentwicklung der Ausbreitungs- und Ablagerungsrechnungen für radioaktive Emissionen aus kerntechnischen Anlagen und Einrichtungen mit der Abluft.
(in preparation)

/3/ Horn, M.:
Weiterentwicklung der Methoden zur Berechnung der Strahlenexposition um kerntechnische Anlagen.
Dissertation (in preparation)

This work was partly sponsored by the Federal Minister of the Interior of the Federal Republic of Germany under the number St. Sch. 729.

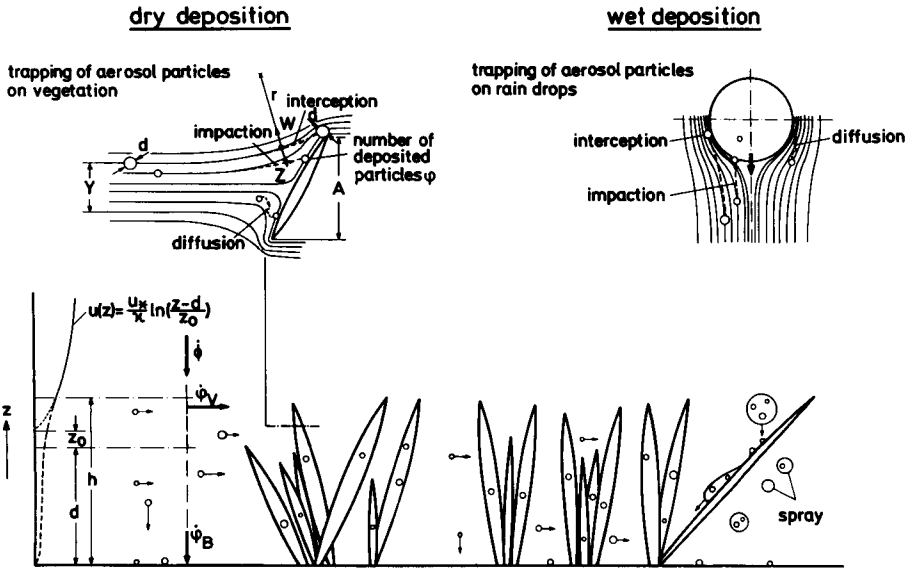

Fig. 1: Dry and wet deposition of aerosol particles on above ground plant parts and on the ground

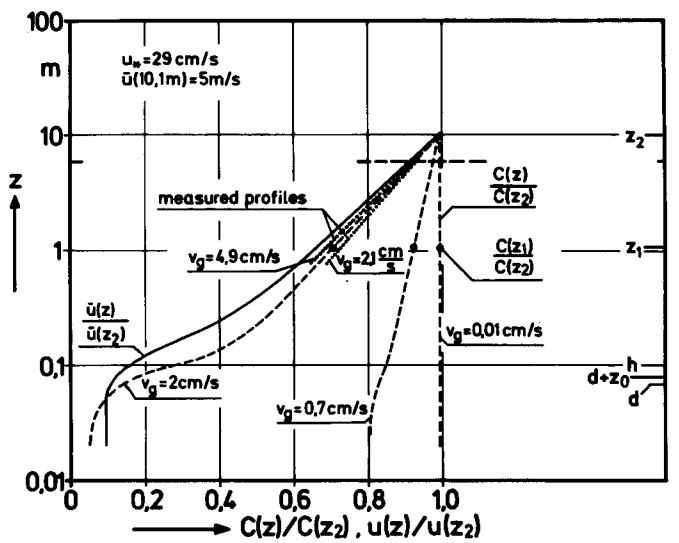

Fig. 2: Wind velocity- and concentration profile at different deposition velocities for grass

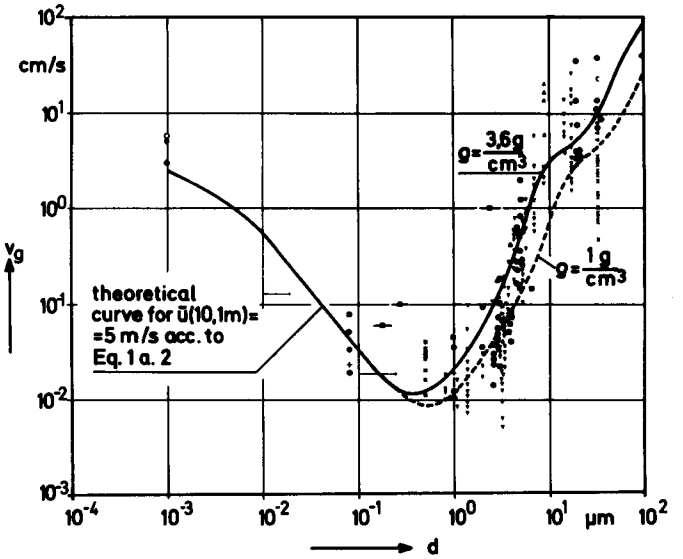

Fig. 3: Comparison of the calculated deposition velocities for aerosol particles on grass with measurements

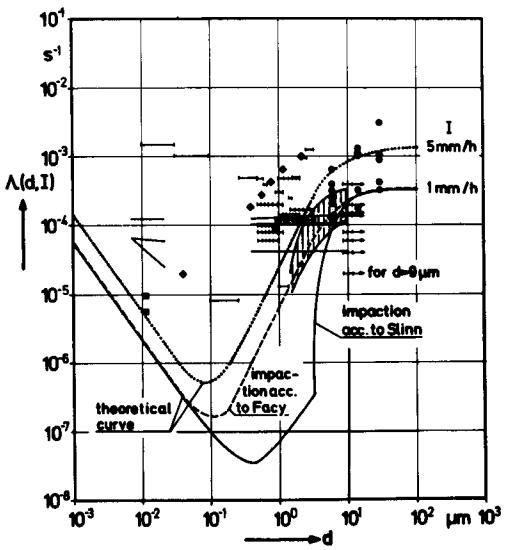

Fig. 4: Comparison of the calculated washout coefficient for aerosol particles with measurements

INVESTIGATION OF THE PATHWAY OF CONTAMINATED SOIL TRANSPORTED TO PLANT SURFACES BY RAINDROP SPLASH*

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2. Los Alamos National Laboratory, Los Alamos N.M. U.S.A.
3. Colorado State University, Fort Collins, CO. U.S.A.

INTRODUCTION

Previous studies have shown that, in most terrestrial ecosystems, over 99% of the inventory of refractory, long-lived radionuclides are ultimately associated with the soil. Therefore, environmental factors governing soil erosion, such as wind and water, are dominant transport mechanisms of radionuclides tightly bound to soil.[1]

An important pathway of radionuclide entry into food webs is the ingestion of surficially contaminated vegetation. Supporting this idea are observations of greater plant/soil concentration ratios in the field as compared to greenhouse studies. Concentration ratios from field studies are one to several orders of magnitude higher than those ratios found in greenhouse studies, indicating substantial surficial contamination of vegetation growing in natural environments.[2] Several investigators have suggested resuspension of soil by wind and water as a major source of surficial contamination on vegetation.[1,3]

Depending on their size, soil particles are selectively transported by wind and water erosion processes. Small particles are generally more easily transported and carried farther than larger particles.[1] Understanding relationships between soil particle size, contaminant distribution among particles, and soil-particle sorting by wind and water are essential for predicting the fate of soil-bound toxic and radiotoxic materials in terrestrial ecosystems.

We investigated the transport pathway of soil- therefore contaminants-to vegetation surfaces via raindrop splash.

The data collected has shown that soil transported by raindrop splash can significantly contribute to the contamination found on plants. This is a pathway that should be included in dose-assessment modeling endeavors. Further detailed study is needed to calculate the rate constant for the raindrop splash and retention pathways.

MATERIALS AND METHODS

A 9 x 13 m plot was established in Los Alamos National Environmental Research Park in New Mexico. The annual precipitation at Los Alamos averages 46 cm, with about 75% of it falling as late summer rain. Further description of the site can be found elsewhere.[4]

We partitioned the study plot into four subplots to separately study the accumulation and retention of radiotracer-labeled soil particles exposed to rain and no-rain treatments. A roof was constructed over half of the plot to create the no-rain (control) treatment. Two burlap cloth barriers (1 and 2 m in height) were nested around the plot to eliminate wind transport of soil as a significant experimental variable. Two anemometers were located within the plot, about 1 m above the ground surface, to verify the effectiveness of the

*Work done by LLNL under D.O.E. Contract W-7405-ENG-48 and by LANL under Contract W-7405-ENG-36.

wind barriers. Natural precipitation was measured at various locations within the plot by recording, tipping-bucket rain gauges.

Accumulation and retention of soil on vegetation surfaces were determined by labeling soil particles with radioactive tracers. About 545 kg of dry soil was collected from the top 2 cm of the plot and separated into 4 size classes using a mechanical ROTO-TAP sieve shaker: (<53 μm (silt-clay), 53-105 μm (fine sand), 105-495 μm (medium sand), 495-850 μm (coarse sand).

Each size fraction was batch-reacted with one of four gamma-emitting radionuclides dissolved in water. The tracer-soil slurry was leached 5 times with tap water to remove unfixed tracer. The size fractions were recombined to form a composite approximating the original soil. A more detailed description of this procedure has been published.[5]

We laid a 2-cm layer of tracer-labeled soil to the ground surface in two of the accumulation subplots. Uncontaminated tomato plants, direct from the greenhouse were then placed in those subplots.

To measure retention, we manually applied tracer soil directly to the tomato plant surfaces. Contaminated plants were then placed in the other two subplots, which had burlap-covered ground surfaces. The control (no-rain) and experimental (rain) subplots were treated identically.

We used tomato plants (Lycopersicon esculentum), Big Boy and Better Boy varieties, as the particle collection surface. We placed each potted tomato plant in a secondary container in the center of a 36 cm x 36 cm area in the plot so that the top of the pot was flush with the soil surface. Cotton placed on top of the pot prevented tracer from contaminating the potting soil. We considered the absorption of tracers by the foliage to be negligible because the fifth leachate of the soil slurry showed insignificant tracer concentrations.

Tomato plants in the plot were exposed to four natural rainstorms between August 8 and 24, 1979. Plants were harvested randomly from the plot after exposure to a single rainstorm or up to four rainstorms.

Each plant was sampled in two height strata: 0 to 40 cm and greater than 40 cm above the ground surface. The ashed and dissolved samples were counted in the same geometry for tracer gamma-ray emissions on GeLi detectors (80 cm^3 volume, 15% efficiency relative to a 3 by 3 NaI detector) with a resolution of 2.0 keV per channel at 1332 keV. A standard curve relating the net counts per minute of tracer in a sample to soil mass (mg) was constructed for each soil-size fraction. Estimates of the soil mass in plant samples are expressed as mg of soil per gram wet weight of plant to normalize for differences in plant size.

Visual observation of particles on leaf surfaces were also made using the scanning electron microscope to verify the labeled-particle technique.[6]

RESULTS AND DISCUSSION

Four separate rainstorms deposited from 1 to 67 mm of rain in the plot during the course of this study. The plants were exposed to multiple rainstorms to determine rates of soil-particle accumulation and retention as a function of time and soil-particle size. Soil concentrations on plants exposed to rain were statistically related to a variety of precipitation characteristics; however, empirical relationships could not be formulated from four rainstorms.

In the accumulation experiment, no significant quantities of soil particles greater than 105 μm were detected on any of the plants. No soil particles were found on the plants at heights above 40 cm. The greatest amount of soils detected on the lower strata after the plants were exposed to four storms was 2.4 mg/g wet plant of silt-clays. Fine sands reached a peak of 2.6 mg/g wet plant after exposure to 3 storms (160 hours). There is no statistical

difference between the results obtained after exposure to 2, 3, or 4 storms. The accumulation data for the silt-clay size fraction (± 1 standard error) is shown on Figure 1.

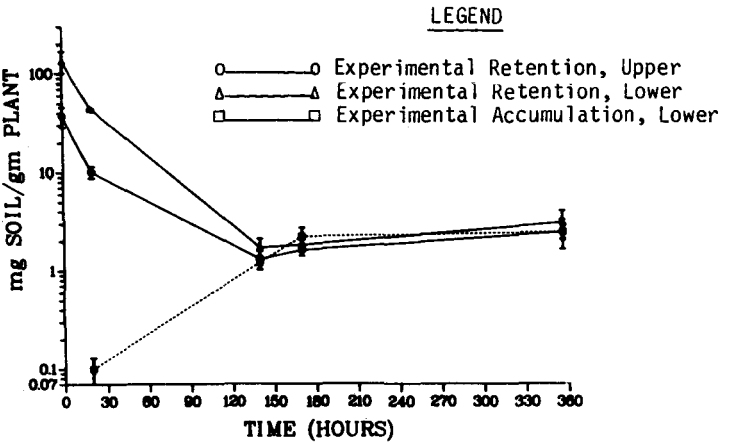

Figure 1. Accumulation and retention of silt-clay soil particles through time.

In the retention experiment, we did not consider soil-size fractions greater than 105 μm because insignificant concentrations of these soil-size fractions were found on contaminated plants by rainfall in the accumulation plot. We found that the quantity of soil per gram of wet plant on the two strata was biased by the contamination technique. Higher soil concentrations were found on the lower strata of the plants. However, the concentration of soil-size fractions less than 105 μm on the two strata did not differ after exposure to two or more rainstorms (140 hours). The retention data for the fine sand size fraction is also presented on Figure 1. The fine sands' size fraction followed the same general relationships as the silt-clay for both the accumulation and the retention experiments.

The majority of soil less than 105 μm on both strata was lost after the first rainstorm. Soil concentrations on both strata do not change significantly after exposure to two or more rainstorms. There appears to be a slight increase with time but this could have been due to data variability or possibly that some of the washed-off soil was resplashed onto the plants during subsequent storms.

The curves presented on Figure 1 show a definite plateau region. There is no statistical difference between the three curves in this region. This indicates that there is a equilibrum concentration of soil that is maintained on the plants through time.

Due to the few data points and the 120-hour span between the first and second storm, we were unable to calculate rate constants. However, we made estimates of the relative importance of raindrop splash in contributing to the plant/soil Pu concentration ratio, based on results of our study as well as past studies of radionuclide concentrations in vegetable crops at the same location.

Concentrations of Pu in garden samples collected in 1976 averaged 8.6 pCi/g dry soil and 0.45 pCi/g dry tomato plant for an average concentration ratio (plant/soil) of 0.05.[7] Concentrations of Pu in the silt-clay fraction of garden soil averaged about 17 pCi/g dry weight.[8] Based on a silt-clay concentration of 2.4 mg/g wet plant, the silt-clay particles on tomato plants would contribute 0.3 pCi Pu/g dry plant. The calculation was made as follows:

$$\frac{2.4 \text{ mg silt-clay soil}}{\text{g wet plant}} \times \frac{1 \text{ g wet plant}}{0.141 \text{ g dry plant}} \times \frac{1 \text{ g}}{1000 \text{ mg}} \times \frac{17 \text{ pCi Pu}}{\text{g silt clay}}$$

$$= 0.29 \text{ pCi Pu/g dry plant.}$$

Given a soil Pu concentration of 8.6 pCi/g, the silt-clay particles and associated Pu would account for a Pu concentration ratio of 0.034 as compared to a ratio of 0.05 that was actually measured in the study plot.

If the contribution of fine sands to the Pu concentration ratio is included (2.6 mg fine sand/g wet plant, 21 pCi Pu/g fine sand) then the combined ratio was 0.079, as compared to an average of 0.05 that we actually measured. Therefore, rain drop splash of soil particles to tomato plant surfaces could easily account for the Pu concentration ratio that was observed in the garden study.

REFERENCES

(1) Hakonson, T. E., Watters, R. L., and Hanson, W. C., 1981, "The Transport of Plutonium in Terrestrial Ecosystems," Health Physics 40, 63-69.

(2) Watters, R. L., Edgington, D. N., Hakonson, T. E., Hanson, W. C., Smith, M. H., Whicker, F. W., and Wildung, R. E., 1980, "Synthesis of Research Literature," in: Transuranic Elements in the Environment (Edited by W. C. Hanson), U. S. Department of Energy, Technical Information Center, Oak Ridge, TN, DOE/TIC-22800.

(3) Arthur, III, W. J., and Alldredge, A. W., 1982, "Importance of Plutonium Contamination on Vegetation Surfaces at Rocky Flats, Colorado," Environmental and Experimental Bot. 22 (1), 33-38.

(4) Nyhan, J. W., Hakonson, T. E., Miera, F. R., Jr., and Bostick, K. V., 1978, Temporal Changes in the Distribution of ^{137}Cs in Alluvial Soils at Los Alamos, Los Alamos National Laboratory, Los Alamos, NM, LA-7298-MS.

(5) Dreicer, M., Hakonson, T.E., Whicker, F. W., and White G. C., 1984, "Rainsplash as a Mechanism for Soil Contamination of Plant Surfaces," Health Physics. 22 (1).

(6) Wallwork-Barber, M. K. and Hakonson, T. E., 1981, Accumulation and Retention of Soil Particles on Plants in Environmental Surveillance at Los Alamos During 1980, Los Alamos National Laboratory, Los Alamos, NM, LA-8810-ENV, p. 59.

(7) Los Alamos National Laboratory, 1978, Environmental Surveillance Report of Los Alamos During 1977, Los Alamos Scientific Report, Los Alamos, N.M., LA-7263-MS, p 38.

(8) White, G. C., Hakonson, T. E., and Ahlquist, A. J., 1981, "Factors Affecting Radionuclide Availability to Vegetables Grown at Los Alamos," Jour. Env. Qual. 10, 294-299.

BIOGEOCHEMICAL INFLUENCES ON THE BEHAVIOUR OF RADIOIODINE IN SURFACE FRESH WATER AND SOILS

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The paper deals with some new detailed insights into the chemistry of iodine in aquatic environments. These may have some importance for understanding the radioecological behaviour of radioiodine, since the transfer of radionuclides in general is basically connected with their chemical states and reactions. In detail it was found:

- In surface water and in aerated soil water, iodide undergoes chemical reactions involving oxidation and tending to reach an equilibrium in which only a small part of the iodine remains in the form of iodide.

- The chemically converted iodine does not turn out to be iodate. In fact the evidence strongly indicates the formation of organic iodine compounds. Moreover iodate shows a tendency to be reduced and thus to be included in these processes.

- The above chemical transformations are instigated by microbiological activity; a great deal can be said about the mechanism of instigation.

- In surface water the formed iodine compounds are mainly dissolved, while in soils a nearly complete incorporation of the iodine into indissolved materials occurs.

- The exact nature of these organic iodine compounds has not yet been determined, but on the basis of the chromatographic techniques, electrophoresis, extraction and precipitation procedures used in these investigations a good deal about their chemical and physico-chemical properties has been revealed.

The observed effects can play a decisive role in the behaviour of radioiodine in the environment, e. g. they can explain the relatively strong retardation of radioiodine oberseved in soils. They also will have some importance for the performance of analytical procedures in radioecological studies.

UPTAKE OF RADIONUCLIDES BY FARM ANIMALS CLOSE TO A MAJOR NUCLEAR INSTALLATION

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INTRODUCTION

A field investigation of the transfer of artificially produced radionuclides in the pasture-cow-milk pathway has been made at a farm close to the nuclear fuel reprocessing installation at Sellafield on the north-west coast of England. The routine discharges from the plant have resulted in enhanced levels of several artificial radionuclides in the local environment. For example, during the period of this investigation the annual depositions of ^{90}Sr and ^{137}Cs at the farm were a factor of about five higher than the average deposition of these radionuclides in the United Kingdom (1). Even if extremely cautious assumptions concerning local eating habits are made, the consumption of meat and dairy products from this farm would give rise to an annual activity intake of less than one percent of the limit for adult members of the public (2). Nevertheless, the presence of enhanced levels of certain artificial radionuclides provides a rare opportunity to study their transfer in the environment.

METHODS

The methods of sample collection and analysis have been described in detail elsewhere (1, 3). Briefly, samples of airborne particulates, rainwater, herbage, supplementary feed and milk were collected at monthly or two-weekly intervals over a period of two years. Samples of soil and cattle faeces were collected at less frequent intervals. At the end of each year, a single cast dairy cow was sacrificed to provide tissue samples. In order to estimate and minimise the uncertainties due to variations between individual samples, multiplicate samples were taken of all media except airborne particulates and rainwater.

After appropriate pretreatment, e.g. concentration by evaporation or drying and milling, the content of ^{137}Cs and other gamma-ray emitting nuclides of all samples was measured with germanium or sodium iodide detectors. The ^{90}Sr content of bulked feed samples, representing intake in each month, and individual samples of rainwater, milk, soil and tissues was determined by an yttrium separation method followed by β particle counting. The stable calcium content of aliquots from the bulked feed, milk and tissue samples was determined by a colorimetric method. Samples of rainwater, feed and milk were each combined to form representative quarterly samples. These and individual samples of soil and tissues were analysed by a radiochemical separation technique followed by α particle spectrometry to determine the content of plutonium isotopes and ^{241}Am. The daily fodder intake of the herd was estimated from a consideration of the theoretical dry matter and metabolisable energy requirements of the cows (1).

RESULTS

Coefficients of transfer from feed to milk, F_m, were calculated by the formula

$$F_m = \frac{\text{concentration of radionuclide in milk (Bq } l^{-1})}{\text{daily intake of radionuclide (Bq } d^{-1})}$$

For ^{90}Sr and ^{137}Cs, this was calculated for each month over a two year period; the mean of the monthly coefficients for each major feeding period are shown in Table 1. The concentrations of plutonium isotopes and ^{241}Am in milk were very

low and usually did not exceed the limit of detection. Hence, only maximum values of F_m could be calculated for these radionuclides.

Similarly coefficients of transfer from feed to flesh (meat or liver) F_f, were calculated by the formula

$$F_f = \frac{\text{specific activity of radionuclide in flesh (Bq kg}^{-1})}{\text{daily intake of radionuclide (Bq d}^{-1})}$$

In this case the activities in flesh are those determined in tissues from the two slaughtered animals and the daily intake was taken to be the mean daily intake during the year preceding slaughter. Table 2 shows best estimates of the coefficients of transfer to meat, liver and milk for each of ^{90}Sr, ^{137}Cs, ^{239}Pu and ^{241}Am; for ^{90}Sr and ^{137}Cs these are equilibrium coefficients, for ^{239}Pu and ^{241}Am these refer to activity at the time of slaughter.

Table 3 shows mean specific activities of ^{239}Pu and ^{241}Am in tissues from two cast dairy cows, sacrificed at six years of age. Estimates of the total activities present in lung, liver, meat and bone are also given. During the two years of the study, the average annual intake by ingestion of each cow of the dairy herd was estimated to be 10 kBq a^{-1} of ^{239}Pu and 5 kBq a^{-1} of ^{241}Am, 90% of which was derived from the consumption of contaminated herbage and soil during the grazing season. The average annual intake by inhalation of each cow was estimated to be 0.16 Bq a^{-1} of ^{239}Pu and 0.07 Bq a^{-1} of ^{241}Am. This estimate is based on measurements of the concentration of these nuclides in air and assumes an inhalation rate of 1.5×10^{-3} m^3 s^{-1}. Measurements of titanium (a soil trace element) in lung, muscle and soil samples indicated that inhalation of soil did not account for the measured activities of ^{239}Pu and ^{241}Am in lung tissues.

DISCUSSION

Strontium-90

The calculated daily intake of ^{90}Sr by the herd varied markedly during each year and was, on average, a factor of three higher during the grazing season than during the winter periods due to the differences in specific activities of pasture herbage and the stored feeds that were supplied. However, the concentration of ^{90}Sr in milk was relatively constant and varied only by a factor of about 50%. The daily intake of stable calcium and concentration of stable calcium in milk were both found to be relatively constant and could not account for the changes in ^{90}Sr transfer coefficient. The observed ratio, OR, of $^{90}Sr/Ca$ in milk divided by $^{90}Sr/Ca$ in daily intake was found to be unusually high during the winter months (OR = 0.4) but lower during the summer months (OR = 0.1, that is within the range found by other investigators). It was concluded that the concentration of ^{90}Sr in milk was being supported both by recent intakes and by activity remobilised from bone; during the summer months a dynamic equilibrium was reached but during the winter months the activity in milk was supported mainly by remobilised activity.

Caesium-137

On a monthly basis, the concentration of ^{137}Cs in milk followed the calculated daily intake very closely with no evidence for a long term retention component. However, as shown in Table 1, coefficients of transfer obtained for the summer months were a factor of about two lower than those obtained for the winter months. The specific activity of ^{137}Cs in surface soil at the farm is relatively high (200 to 300 Bq kg^{-1} dry weight) so that small traces of soil

adhering to the surface of herbage can make a significant contribution to the total measured activity of the herbage. Measurements of herbage sample weight after ashing and measurements of titanium in herbage and soil samples were used to quantify the level of soil contamination on herbage and other feed samples. It was found that the presence of soil on feed samples accounted for between 30 and 60% of the total measured ^{137}Cs activity of herbage but less than 5% of the total measured ^{137}Cs of stored feeds. Measurements ofthe physical and chemical disposition of ^{137}Cs in soil from the farm showed that 85% of the ^{137}Cs was permanently bound to silicaceous mineral particles and would be totally unavailable for biological uptake. It was concluded that the inclusion in daily intake of ^{137}Cs associated with soil on herbage surface could account for the lower coefficients of transfer observed during the summer months. In order to calculate the coefficients of transfer for ^{137}Cs shown in Table 2 daily intake was recalculated excluding the fraction of ^{137}Cs that was estimated to be bound to mineral particulates.

Plutonium-239 and Americium-241

Even if a relatively low gut uptake factor and a relatively high fractional uptake from the respiratory tract are assumed it would appear that intake by ingestion was responsible for the bulk of the systemic activity of these radionuclides in cattle on the farm. The distribution of ^{239}Pu between liver and bone was very similar to that found after experiments in which dairy cows were orally dosed with ^{238}Pu in oxide and citrate forms but the proportion of activity present in meat is approximately twice that found following oral dosing with ^{238}Pu citrate and almost on order of magnitude greater than that found after oral dosing with ^{238}Pu oxide (4). The ratios of ^{239}Pu and ^{241}Am activities in meat to activities in liver and bone were approximately an order of magnitude higher than found in animals grazing on more heavily contaminated pastures in the area (5). It is possible that at these very low levels of intake a greater proportion of systemic intake does indeed become deposited in meat but it should be noted that the absolute levels of activity are very low and subject to analytical and sampling uncertainties. If it is assumed that there has been negligible systemic clearance of plutonium and americium from these animals and also that their intake in previous years was similar to that calculated for the two years of the study, then it is possible to estimate fractional gut uptake values for these animals of 8×10^{-5} for ^{239}Pu and 2×10^{-4} for ^{241}Am. These values are lower than found in oral dosing experiments (4) but higher than usually associated with these radionuclides.

REFERENCES

1. Sumerling T. J., Dodd, N. J. and Green N., The transfer of ^{90}Sr and ^{137}Cs to milk in a dairy herd grazing near a major nuclear installation. To be published in Sci. Tot. Environment.
2. Sumerling, T. J. and Crick M. J., A preliminary evaluation of a dynamic model for the transfer of radionuclides in the pasture-cow-milk pathway with data from a field investigation, in Proc. CEC Seminar. The transfer of radioactive materials in the terrestrial environment subsequent to an accidental release to atmosphere, Dublin 1983.
3. Green N., Radiochemical analysis of samples from an environmental pathway study, Laboratory of the Government Chemist 4th Symposium on the Determination of Radionuclides in Environmental and Biological Materials, 1983.
4. Stanley R. E., Bretthauer, E. W and Sutton, W. W., Absorption, distribution and excretion of plutonium by dairy cattle, in Radioecology of Plutonium and Other Transuranics in Desert Environments. Eds. White M. G. and Dunaway P. B., NVO 153 1975.
5. Popplewell, D. S. P., Personal Communication, 1983.

Table 1: Mean of monthly coefficients of transfer from feed to milk, F_m, for ^{90}Sr and ^{137}Cs.

Season	Feed	F_m, d l^{-1}	
		^{90}Sr	^{137}Cs
Winter 1980/81	hay and concentrates	3.8×10^{-3}*	6.4×10^{-3}
Summer 1981	grazed pasture	1.2×10^{-3}	3.8×10^{-3}
Winter 1981/82	silage and concentrates	2.9×10^{-3}*	7.2×10^{-3}
Summer 1982	grazed pasture	1.0×10^{-3}	3.4×10^{-3}
Winter 1982/83	silage and concentrates	---	7.4×10^{-3}

* not in equilibrium, see discussion.

Table 2: Best estimates of coefficients of transfer from feed to meat, liver and milk.

Product	Coefficient of transfer, d kg^{-1} or d l^{-1}			
	^{90}Sr	^{137}Cs*	^{239}Pu	^{241}Am
Meat	5×10^{-4}	1.4×10^{-2}	3×10^{-4}	5×10^{-4}
Liver	--	9×10^{-3}	1.6×10^{-3}	2.1×10^{-3}
Milk	1×10^{-3}	7×10^{-3}	$< 7 \times 10^{-6}$	$< 1 \times 10^{-4}$

* excludes intake of ^{137}Cs bound to mineral particulates, see discussion.

Table 3: Mean specific activities and total activities of ^{239}Pu and ^{241}Am in tissues from two slaughtered cows.

Tissue	Mean Weight kg	Specific activities, mBq kg^{-1}		Total activities, Bq	
		^{239}Pu	^{241}Am	^{239}Pu	^{241}Am
Lungs (2)	4.1	12	7	0.05	0.03
Liver	8.4	54	29	0.45	0.24
Meat	140*	6	4	0.8	0.6
Bone	70*	51	78	3.6	5.5

*estimated from dressed carcase weight, excludes head and hooves.

ASSESSMENT OF 210-Po AND 210-Pb ATMOSPHERIC CONCENTRATION IN THE ENVIRONMENT OF COAL FIRED POWER PLANTS

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The radiological impact of fossil fuel burning has been investigated for about twenty years. The ingestion pathway has been found to result in the highest individual doses, whereas the inhalation pathway makes the dominant contribution to collective doses. In a recent paper, it was shown that the individual dose from 210-Po and 210-Pb releases to japanese people was of the order of several tens of μSv/year. A field study has been carried out : measurements of Po-210 were made on air samples collected under the plume of an isolated coal fired power plant in relation with SO_2 concentration measurements used as a tracer of the release. The two pollutants were determined at several sampling locations and a good correlation was observed between their concentrations, showing that the stack is an important source of natural radioactivity.

DOUBLE TRACER EXPERIMENTS TO INVESTIGATE MODELS FOR THE CALCULATION OF GAMMA DOSES FROM A RADIOACTIVE CLOUD

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INTRODUCTION

This paper presents work from a series of atmospheric dispersion experiments in May 1981 at the Ringhals nuclear power plant in Sweden. The aim of the project was to obtain short-term observations of concentrations and gamma-ray exposures from stack effluents and to compare these results with corresponding values calculated from computer models.

Two tracers, sulphurhexafluoride (SF_6) and radioactive noble gases, were released from a 110-m stack and detected at ground level downwind at distances of 3-4 km. Calculations were made with two Gaussian plume models: PLUCON[1] developed at Risø National Laboratory and UNIDOSE[2] developed at Studsvik Energiteknik AB.

EXPERIMENTS

The Ringhals nuclear power plant is situated on the Swedish west coast 50 km south of Gothenburg. The radioactive noble gases used for the experiments were routine emissions from unit 1 (BWR). The experimental site, Fig. 1, is a rather level rural area reaching about 60 m above mean sea level at its highest. The vegetation comprised partly coniferious trees and partly agricultural fields covered with grass.

The experiments were based upon a good meteorological forecast, and required one hour for the setting-up of the sampling network. Via radio control the one-hour sampling of SF6-concentrations and of gamma radiation was synchronized for all positions.

An 11-m meteorological mast was set up in a fairly homogeneous flow field east of the reactor stack, Fig. 1. The mast gave information at different heights on wind speed, wind direction, and temperature. During each experiment, a radiosonde was launched from here yielding information on the vertical structure of temperature and humidity in the atmosphere. A 96-m mast is permanently set up close to the power plant. However, due to very inhomogeneous surroundings the measurements from this mast are unsuited to a detailed analysis of the local meteotological conditions during the individual experiments.

During 3 of the 4 experiments the meteorological conditions were near neutral corresponding to Pasquill category D with wind speeds of 8-14 m/s. The remaining experiment was made under stable conditions, Pasquill category E, with a wind speed of 3.5 m/s.

The tracer SF_6 was injected at a constant rate to the stack. Prior to an experiment the automatic tracer sampling units were transported to positions marked out in advance, Fig. 1, and distributed according to the actual wind direction. The distance between the sampling units was typically 150 m. A total of 25 units were available. Air samples in bags were collected immediately after each experiment and later analysed for their content of SF_6 by use of electron capture detector gas chromatography. The uncertainty of the absolute tracer concentrations is about 20% and the short-term reproducibility is about 2%.

The amount of noble gases emitted from the stack was monitored continuously at the power station. The radiation from the plume was detected with 11 GM-counters, 3 ionization chambers, and 3 mobile Ge(Li) spectrometer systems. These instruments were intercalibrated prior to the experiments with certified gamma sources. The GM-counters and the ionization chambers gave information on exposure rates, whereas the gamma spectrometers gave information on unscattered gamma radiation, which permitted identification and quantification of the individual radionuclides.

During the experiments each radiation detector was placed near an SF_6 sampling unit. Due to the relatively small number of radiation detectors these were placed along the crosswind measurement line at every second SF_6 sampling unit.

RESULTS

Using the recorded radioactive source terms and meteorological data, calculations were made with the two Gaussian models of concentrations, exposure rates, and unscattered gamma fluence rates to be compared to their measured counterparts.

Considerable care was invested to estimate the effective release heights of the tracer gases from the stack during the experiments. Momentum, buoyancy, and downwash effects were taken into account.

The gamma spectrometer results showed unexpectedly a consistent significant surplus of the radioactive decay products of the noble gases relative to the noble gases themselves. This was interpreted as dry deposition of the non-gaseous decay products on vegetation and on the ground. The deposition velocities estimated to account for the surplus of the noble gas decay products were relatively high (2-10 cm/s), but not unphysical considering the circumstances.

Table 1 shows the ratios of the measured to calculated values of concentrations and exposure rates for the four experiments. The two Gaussian models yield rather similar results, so for simplicity only one set of calculations was used. The results for experiment I are displayed in Fig. 2.

In experiment II made under stable conditions the plume did not touch the ground at the measurement positions. In the other experiments the measurement positions were situated near the point of maximum concentration. This may have contributed to the difficulties encountered to estimate the vertical dispersion parameters, σ_z, probably caused by non-Gaussian vertical profiles of the plumes. A detailed report on the experiments is in preparation[3].

REFERENCES

1. Thykier-Nielsen, S. The Risø Model for Calculating the Consequences of the Release of Radioactive Material to the Atmosphere. Risø-M-2214 (1980) 65 pp.

2. Karlberg, O. et al. UNIDOSE-A computer program for the calculation of individual and collective doses from air-borne radioactive pollutants. Studsvik Report 79/1 (1979), Studsvik Energiteknik AB, Sweden.

3. Risø-R-492.

Table 1. Ratios of the measured to calculated values of concentrations and gamma exposure rates in crosswind measurement positions for the four experiments.

Experiment I			Experiment II			Experiment III			Experiment IV		
pos	con	exp	pos	con	exp	pos	con	exp	pos	con	exp
52	1.0		72	BDL	4.5	0	0.7		-1	1.6	0.9
53	1.5	2.1	73	-		1	1.1	0.2	0	1.4	
54	2.2		74	-	2.1	2	1.4		1	1.4	0.9
55	1.7	1.2	75	-		3	0.9	0.2	2	1.3	
56	1.8		76	-	0.4	4	0.9		3	1.8	1.0
57	1.2	0.9	77	-		5	1.4	0.2	4	2.0	1.5
58	1.2		78	-	0.4	6	3.0		5	1.0	
59	1.2	1.4	79	-		7	1.0	1.0			
60	1.9		80	-	0.7						
61	2.3	1.9	81	-							
62	1.4		82	-	1.3						
63	1.0	2.0									
Mean	1.5	1.6	Mean		1.6	Mean	1.3	0.4	Mean	1.5	1.1

BDL below detection limit

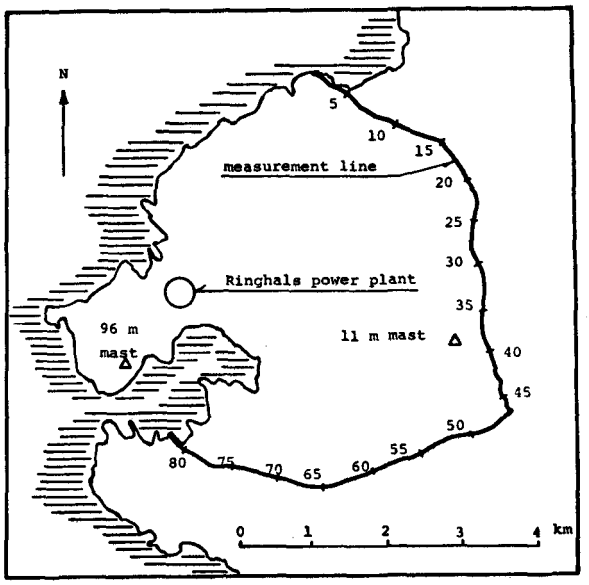

Fig. 1. Sketch of the experimental site.

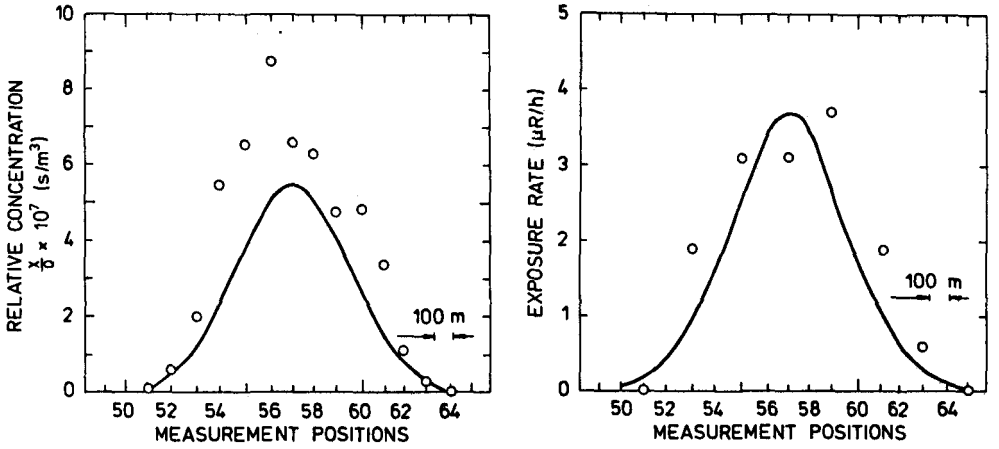

Fig. 2. Measured (O) and calculated (-) results from experiment I showing crosswind profiles of relative concentrations (upper figure) and of gamma exposure rates (lower figure).

STOCHASTIC MODELLING OF CONCENTRATIONS RESULTING FROM SHORT TERM RELEASES OF RADIONUCLIDES TO THE ATMOSPHERE

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1.Introduction.

The MESOS model for transport and dispersal of radioactive releases to the atmosphere over distances of up to 1000 km or more has been descibed at a previous IRPA meeting (1).

Databases of meteorological data (from synoptic stations and ships at 3 hourly intervals over Europe throughout 2 years - 1973 and 1976), have since been completed, and used with MESOS to provide realistic meteorological scenarios for accident releases. An extensive program of calculation has led to a large number of results for specific sources, nuclides and receptor points. These results are now being analysed to provide a simple stochastic model which for any source and receptor point, given the release duration, will provide cumulative probability distributions of exceeding different levels of contamination per unit release of a specified nuclide. Such a tool is of direct application in accident risk analysis; and also in particular to transfrontier consequences of a nuclear accident, relevant to requirements of Article 37 of the EURATOM treaty. Of particular interest are the highest levels of contamination predicted at remote receptor points and the atmospheric conditions leading to them. Proper statistical analysis of these highest values requires the application of extreme value theory.

2. Calculations with the MESOS Model.

The MESOS model has been described in detail in (2). It simulates dispersion of sequential 3 hour releases using meteorological data over W. Europe to estimate trajectories and dispersion according to evolving atmospheric conditions along those trajectories. A validation study simulating the Windscale release of 1957 was outlined in (1). Results for hypothetical unit releases of various nuclides (Kr^{85}, Xe^{133}. Xe^{135}, Cs^{137},I^{131}) over 3 hours have been generated for several source locations in Europe using both the 1973 and the 1976 data bases. Since 1973 was meteorologically a fairly average year, and 1976 rather extreme with several severe blocking situations, comparison gives an indication of the variation which may occur from year to year. Results for releases of longer duration have been obtained by combining sequential 3 hour releases, since release duration is an important parameter. Time integrated atmospheric concentration (TIC), and dry and wet deposition where relevant, have been calculated at a range of receptor points round each source, and used to generate predicted frequency distributions of different levels of contamination (see (1) for

3. Statistical Analysis.

In order to make these results more directly useful it was necessary to analyse them statistically and produce a simple stochastic model to predict probability distributions of contamination for a general source and receptor point. This simple model divides the calculation into 2 parts; the probability that a release leads to any positive contamination at the receptor (i.e. that it passes over or near that point), and the cumulative probability distributions of different levels of contamination when such exposure does occur, appropriate to the specified nuclide and release duration.

The exposure probabilities for 3 hour releases are independent of the nuclide but differ for wet deposition; they are estimated using a geostrophic wind rose at the source and the source-receptor distance via a straightforward log-regression equation estimated from the MESOS-generated data. For longer release durations probabilities of exposure are obtained from the 3 hour release probabilities using another empirical formula based on the data. An idea of the accuracy of the prediction equations is given by the comparison of MESOS results and statistically predicted probabilities in table 1. Overall agreement is good.

Receptor	Release duration (hours) 3	6	12	24	72	168
1 100 km N	10.6	15.1	21.8	31.7	56.5	85.2
	13.7	19.3	28.5	40.7	65.3	83.9
2 200 km N	7.4	10.0	16.1	23.8	46.2	72.8
	12.3	17.5	25.9	37.4	61.3	80.5
3 400 km N	5.3	8.0	12.1	18.2	38.1	62.4
	9.9	14.3	21.4	31.3	53.3	73.1
4 800 km N	4.1	5.9	9.1	15.1	27.0	41.6
	6.4	9.5	14.4	21.5	38.8	57.1
5 100 km E	16.4	23.4	34.4	48.3	76.1	94.7
	18.2	24.8	35.9	50.1	75.5	91.2
6 200 km E	15.5	22.0	31.7	44.7	74.0	91.6
	16.3	22.5	32.9	46.3	71.6	88.6
7 400 km E	12.3	18.4	27.3	40.8	69.7	89.2
	13.1	18.5	27.4	39.3	63.6	82.5
8 600 km E	11.2	16.2	25.3	38.0	62.9	83.2
	10.5	15.2	22.6	33.0	55.6	75.3

TABLE 1. Comparison of MESOS and statistically predicted %-probability of exposure at 8 receptor points for several release durations : Hanover dry deposition
MESOS
Statistical prediction } %-probability

The mean and variance of the distribution of contamination

at the receptors when exposed can be predicted from a few simple variables such as source-receptor distance, source geostrophic wind rose, and nuclide half-life, desposition velocity and wash-out coefficient. They are then used to prescribe the distribution itself, which is represented by a Weibull distribution. Figure 1 shows an example of distributions produced directly by MESOS and the corresponding cumulative exposure distributions predicted by the simple stochastic model; they agree very well.

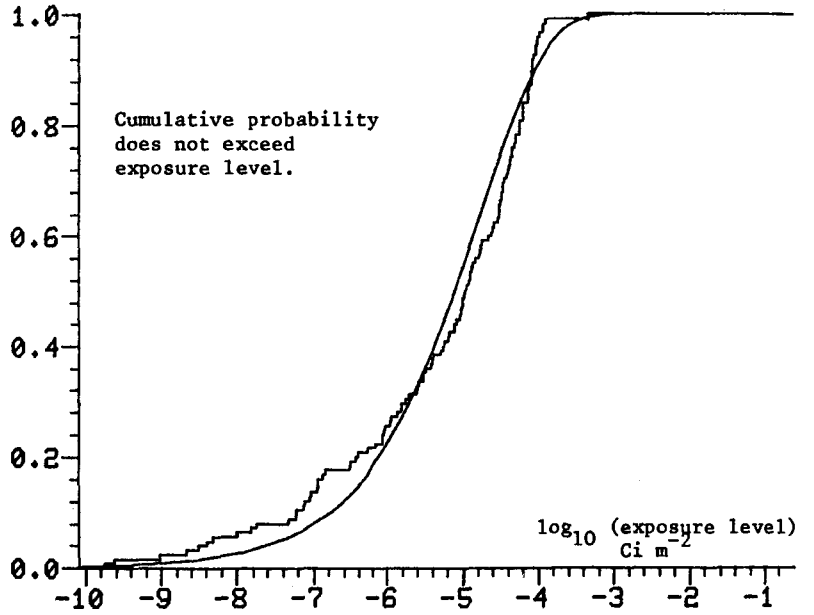

FIGURE 1. Comparison of MESOS and statistically predicted exposure cumulative probability functions for 3-hour release duration : Mol 1976 I^{131} (P) wet deposition, receptor 5, 100 km East.

The potential user of the stochastic model will require geostrophic wind roses at or near the source; these are being extracted at a grid of points over Western Europe from the MESOS databases.

4. Situations Leading to Relatively High Levels of Contamination

Of particular interest are those meteorological conditions which are likely to result in high levels of contamination at a receptor point. A special study has therefore been made of the situations leading to the highest values predicted with MESOS.

At distances up to the order of 100 km the same releases tend to give high TICs for all nuclides simultaneously; many of the most critical exposures were associated with slow anticyclonic conditions in late summer and autumn, and evening and night time releases in stable conditions predominated. At longer distances

time of release was less significant and critical conditions varied more with the decay and deposition characteristics of the nuclides. In several cases the wind speed dropped during passage over the receptor giving longer exposure times. Lateral spreading due to divergence in the synoptical scale windfield was often minimal with persistent trajectories in a stationary meteorological situation. High exposures tended to occur clustered in episodes, examples of which are discussed in (2). Situations leading to high wet deposition are rather different, since anticyclonic conditions are generally dry; they depend far more on the location of the receptor point relative to the source.

Statistical analysis of the highest levels of contamination is being undertaken by considering only the occurrence of excesses over high threshold levels. It is important to allow for the clustering of these high exposures; since ignoring this can lead to very misleading conclusions. For example, it is usually incorrect to take the few highest values and assign a probability proportional to the fraction of the total sample that they represent. The application of extreme value theory is appropriate, and a fairly simple method for describing these rare episodes can be applied. This is described in (3).

5. Conclusions

Results generated using the MESOS model have been used to derive a simple stochastic model which, given a release of a nuclide over a defined period from a specified source, may be used to derive a cumulative probability distribution of exceeding various levels of contamination at a chosen receptor point. Attention is now being focussed on the higher levels of contamination which may occur, and the application of extreme value theory.

6. Acknowledgements

This work has been supported by the Commission of the European Communities, DG V, Luxembourg. We are indebted to Mr. G. Fraser for many useful discussions.

7. References

1. H.M. ApSIMON, A.J.H. GODDARD, J. WRIGLEY. Long Range Transport of Radioisoptopes in the atmosphere and the Calculation of Collective Dose. Proc. 5th Int. Cong. IRPA, 1980. Vol. II, pp 51-54.

2. H.M. ApSIMON, A.J.H. GODDARD. Assessment of Collective Dose from Planned and Unplanned Releases to the Atmosphere. Final Contract Report to Assoc. EURATOM/CEA, 1983.

3. A.C. DAVISON. Modelling Excesses Over High Thresholds with an Application. Proc. NATO As I. Statistical Extremes and Applications. Portugal, 1983.

WEATHERING OF FISSION PRODUCTS DEPOSITED ON ASPHALT

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Introduction

The exposure rate level from radioactive materials deposited on outdoor surfaces will diminish not only with the physical half-life of the nuclei but also because of movements of the materials.

What can happen on a road is either that the materials penetrate into the road surface or it can be removed alltogether by going into a sewage with rainwater.

The dose rate's variation in time, D(t), can be described as:

$$D(t) = D(o)\ W(t)\ \exp\ (-\ \lambda t)$$

where W(t) can be splitted up into a part with short weathering halflife and one with a long halflife:

$$W(t) = A \cdot \exp(-\ \lambda_{w1} t) + (1-A)\exp(-\ \lambda_{w2} t)$$

λ : the decay constant corresponding to the physical halflife of the radionuclide.
λ_{w1} : the decay constant for the short term weathering.
λ_{w2} : the decay constant for the long term weathering.
A : the part that disappears with short term halflife.

The parameters that influences the weathering is:

1) The contaminating material (nuclide, chemical compound, concentration).
2) The surface (chemical compound, roughness porosity).
3) The way of deposition (wet or dry).
4) The history of the contaminated area since deposition (weather, traffic).

Road Experiments

At Risø we have contaminated asphalt areas with three different fission product nuclides and then followed the dose rate at 1 m above the road surface. The contamination was allways done with the radionuclide dissolved in water.

The amount of experiments done is not sufficient to give any clear indication on how the weathering depends on the different parameters involved. But at Fig. 1 it can be seen that there is a correlation between the short term weathering halflife and the part removed: when we have a very short halflife, up to 70% gets removed and when the short-term halflife, is long, almost nothing gets removed.

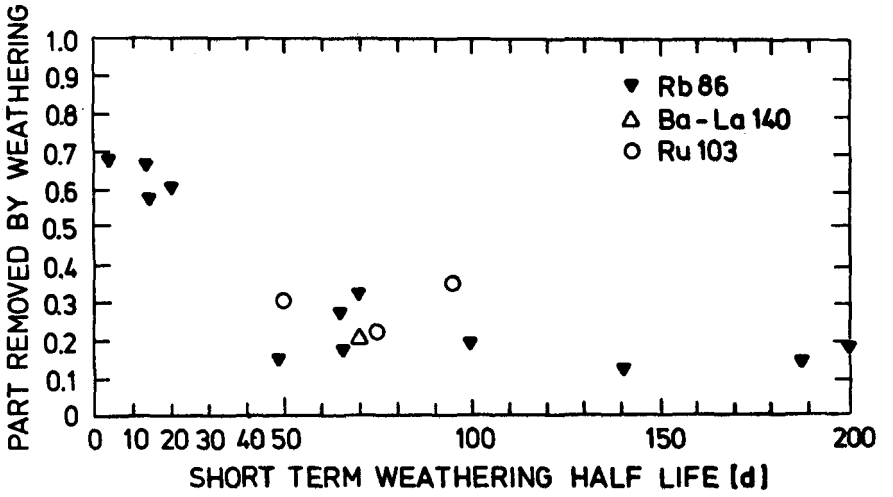

Fig. 1.: Weathering efficiency plotted against the weathering half-life.

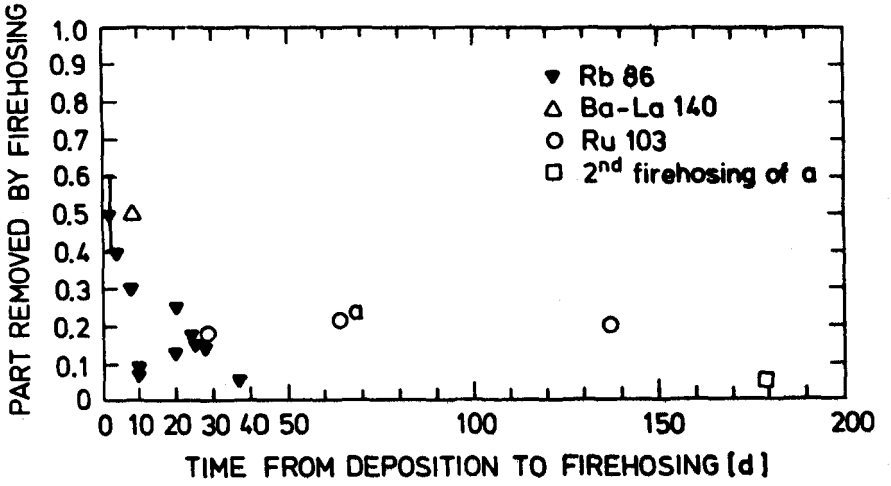

Fig. 2.: Efficiency of fire-hosing plotted against age of contamination.

Forced Decontamination

Several decontamination attempt were made, the most successfull being fire-hosing. On Fig. 2 is shown the results of the fire-hosings against the time at which it was done. It is clearly seen that about half of the Rb86 (representing Cs134 and Cs137) can be removed if the fire-hosing takes place very soon after deposition. Whereas it has no meaning to try removing Rb86 by fire-hosing after about 30 days.

The two other nuclides Ba-La140 and Ru103 seem to have a different behaviour. For Ru103 there is no time dependence to be seen, it allways seems possible to remove about 20%.

Conclusion

The results of the experiments vary a lot, but to get a general idea (and a mumber to use in consequence models) the linear average of the halflifes and removed part have been calculated and for model purpose we then get the dose rate time dependence to be

$$D(t) = D(o)\ \exp(-\lambda t)\cdot (0.3\cdot \exp(-3.2y^{-1}\ t) + 0.7\cdot \exp(-7.5\cdot 10^{-3}y^{-1}\ t))$$

($\lambda_{w2} = 7.5\cdot 10^{-3}y^{-1}$ is taken over from Gale, 1963-AERE-R4241).

The results indicate that Rubidium/Cesium after some weeks are fairely well bound to the asphalt surface, whereas Barium/Lanthanum and Ruthenium do not appear to react chemically with the surface and they are therefore more mobile.

EXPERIMENTAL STUDIES ON THE BEHAVIOUR OF TECHNETIUM IN THE MARINE ENVIRONMENT**

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Introduction

The presence of technetium in the biosphere depends largely on the most important isotope ^{99}Tc, which enters the environment as a result of nuclear fuel reprocessing, nuclear waste storage, nuclear weapon testing and, to a very limited extent, the use of ^{99m}Tc in radiodiagnostics (1,2).

The long-lived radionuclide ^{99}Tc ($T_{1/2}$ = 2.15x10^5 years) is produced in relatively high yield (6%) in both thermal and fast fission of ^{235}U, ^{238}U and ^{239}Pu. Because of its long half-life, ^{99}Tc contributes to a considerable extent in determining amounts of long-term radioactivity and hazards of high-level wastes from nuclear fuel. A critical view on the inventory of the most important radionuclides in spent fuel from light water reactors shows clearly that after nearly 1000 yeras of aging, ^{99}Tc represents the major contribution to the all-over radioactivity of waste, but in terms of hazard, ^{99}Tc is much less important even after 10^6years because it is not as radiotoxic as the actinides.

Despite projections showing future increase in the inventory of ^{99}Tc (170,000 kg equal to 1.06x10^{16} Bq by the year 2000)(3), little information is available on the biogeochemical behaviour of technetium in the marine environment. Previous studies of the behaviour of technetium in sea water and sediments have shown (4,5) that in agreement with thermodynamic considerations, the anion pertechnetate is the chemical form most stable in well oxygenated sea water and in aerobic sediments. In the presence of anoxic sediments, pertechnetate may be reduced and immobilized as highly insoluble compounds (5). Bacterial activity and organic matter content could play an indirect role in the fixation process of technetium by determining redox conditions favourable for its immobilization in the sediments.

A review of recent studies on the behaviour of technetium in marine biota shows that ^{99}Tc is accumulated by a variety of marine organisms (6). The present investigation will give attention to the biokinetic behaviour and fate of technetium in selected marine organisms.

** Contribution N. 2066 of the Radiation Protection Programme of the Commission of the European Communities, partially done under the contract BIO B322-I(S)

Material and Methods

The photon-emitting radionuclide ^{95m}Tc ($T_{1/2}$= 60 days) facilitated greatly the experimental procedure in as much as it permitted in vivo measurements of technetium in organisms. The different gamma energies are easily detectable with NaI(Tl) scintillation crystals or by Compton effect in a liquid scintillation medium. We used both well-type crystals and an Armac (Packard Instr. Co.) liquid scintillation detector for large samples. All measurements of ^{95m} Tc were corrected for physical decay.

Experimental organisms: the crabs Pachygrapsus marmoratus, Carcinus maenas and the benthic fish Lepadogaster lepadogaster, were maintained in aquaria with running seawater for sufficiently long times to achieve good acclimatation to the experimental conditions prior to being used in radiotracer experiments.

The accumulation of ^{95m}Tc as pertechnetate (37 kBq/L) from water by the mediterranean crab P. marmoratus was followed for one month using 20 specimens (15 females and 5 males) of 12.99±4.95 g and of a carapace width of 27.85±3.86 mm. The single specimens were whole-body counted after being rinsed for several minutes in a non-radioactive bath in order to get rid of the external radioactivity of the exoskeleton and the gills. The crabs were fed daily with fresh mussels for two hours. The technetium concentration in sea water was checked at different time intervals and readjusted to the initial value by adding new pertechnetate solution.

At the end of the accumulation period, 10 specimens were dissected and the incorporated radioactivity was measured in the organs and tissues. The other 10 specimens remaining were used to follow the release kinetics of ^{95m}Tc over a period of 110 days by placing the crabs in non-radioactive water which was changed each day. Measurements of the radioactivity remaining were made by whole-body counting. At the end of the release phase, with about 10% of initial radioactivity still incorporated, the crabs were dissected and the distribution of ^{95m}Tc measured in organs and tissues.

Attempts to determine the assimilation efficiency of ingested technetium and the subsequent behaviour of its different oxidation states have been made using crabs (Carcinus maenas) and benthic fish (Lepadogaster lepadogaster). Several individuals of both species previously acclimated in running sea water were fed a single ration of mussel digestive gland which had been injected with 10 µl ^{95m}Tc solution (activities ranging from 5.5 to 18.5 kBq) in either the VII or IV oxidation states.

Immediately following ingestion, the animals were whole-body counted and replaced in glass jars containing 1 L filtered sea water which was changed daily. This procedure was continued for 4 days, after which time the crabs were transferred to large basins with running sea water. The diet of the fish was supplemented with Artemia brine shrimp. During the first 3 days of excretion, fish fecal pellets were filtered from the sea water daily and also counted for radioactivity. Crab feces were monitored in a similar fashion until the 28th day of loss.

RESULTS

Bioavailability of Technetium

The bioavailability of technetium to crabs and fish was studied in two different ways. For P.marmoratus, technetium-95m as pertechnetate was taken up from water over a one month period, while for C.maenas and for the fish species ^{95m}Tc either in VII or in IV oxidation states was administered to the animals in single rations of pretreated food.

In the case of direct uptake from water, the crabs accumulated the isotope very slowly, reaching a concentration factor of 8 after 30 days. The relative short exposure time did not permit the animals to reach equilibrium conditions.

There were at least three different pools for technetium found in the crabs: the digestive gland with 53%, the gut with 20%, and the gills, muscle and others with 27% of the total radioactivity assimilated. The absorbed fractions in these organs turned over with biological half-lives of 13, 35, and 118 days,respectively. Concentration factors for these principle organs were found to be 115, 67, 28, and 16 for hepatopancreas, stomach, gut, and gills, respectively.

When technetium was ingested by fish with food, assimilation efficiences ranged from 55-75% and the absorbed fraction of both oxidation states subsequently turned over with a biological half-life of approximately 120days. About 50% of the ingested radioactivity was absorbed across the gut by the crab C.maenas, but the turnover rate for this fraction was much more rapid than in fish, as evidenced by a biological half-life of only 60 days.

Organ and Tissue Distribution

At the end of the accumulation phase more than 70% of the total radioactivity taken up from water by the crab P.marmoratus was localized in the digestive system with hepatopancreas, stomach, and gut accounting for 53, 16, and about 2%, respectively. Most of the remaining technetium (20%) was associated with muscle tissues of legs and pincers, while only a small fraction of 3% was found in the gills,which may act as a way of entrance and elimination of technetium.

After 120 days of release the distribution pattern of technetium in the different organs of P.marmoratus changed. Still 16% of the body burden was localized in the digestive gland, while stomach, gut, gills, and muscle accounted for 14, 2,10 and 38% of the total radioactivity. In C.maenas similar figures were found after 131 days of release. In this case, digestive tract, gills and muscles accounted for 25, 8, 17 and 37%, respectively. The majority of the technetium was associated with the muscles, while by contrast the hepatopancreas and the digestive tract accumulated technetium to the greatest degree.

No significant difference of the distribution of Tc among the tissues of fish was noted between the IV and VII oxidation state. The highest concentration were found in the liver. This organ contained 25% of the body burden after 250 days of release. The heart and gall bladder also showed a marked affinity for technetium. The presence of significant amounts of incorporated technetium in all tissues after several months of release demonstrates its strong binding capacity to the tissues of vertebrates and invertebrates alike.

Excretion

During the first 3 days of loss about 18% of the technetium eliminated by the fish Lepadogaster lepadogaster was associated with the feces regardless the oxidation states (IV and VII). Most of this fraction was excreted during 24-48 hours post ingestion. The amount lost due to defecation is a minimum estimate since sequential leaching experiments with fish fecal pellets showed that as much as 80% of the technetium content could be lost within 4 hours. Nevertheless, it appears that most of the technetium initially ingested by these fish is excreted in soluble form either as urine or directly across the gills. In this case the two oxidation states of technetium exhibited a different elimination behaviour. About 40% of the reduced form (IV) and about 50% of the pertechnetate (VII) were excreted as soluble forms. Despite the high fecal excretion of technetium over the first few days of depuration, some residual radioactivity was found in feces excreted as long as 21 days after ingestion.

With the crabs C.maenas, the fraction lost via fecal excretion appears to be much less (about 8%) than that from fish. Furthermore, unlike fish fecal excretion of technetium occurs over a longer period as evidenced by significant amounts of ^{95m}Tc measured in crab-fecal pellets collected periodically during the first month of loss. Given the low fraction (about 8%) lost with the feces over the 28 day period, it seems clear that defecation by crabs is a less important route of technetium excretion than in fish, at least during the early phases of radionuclide depuration.

REFERENCES

1) WILDUNG,R.E.,McFADDEN,K.M.,GARLAND,T.R., Technetium sources and behaviour in the environment, J. Environ. Qual. 8, 156 (1979).
2) TILL, J.E., HOFFMAN, F.O.,DUNNING, Jr., D.E., A new look at Tc-99 release to the atmosphere, Hlth. Phys. 36, 21 (1979).
3) BURKHOLDER, H.C., CLONINGER, M.O., BAKER, D.A., JANSEN, G., Incentives for partitioning high-level waste. USERDA Rep. BNWL-1927, Natl.Tech.Inf.Serv.(1975).
4) ALLARD, B., KIGATSI, H., TORSTENFELT, B., Technetium: reduction and sorption in granitic bedrock. Radiochem. Radioanal. Letters, 37, 223,(1979).
5) SCOPPA, P., SECONDINI, A., SCHULTE, E.H., Indagini sulla stabilità dell'anione pertechnetato nell'ambiente marino, Atti del XXII Congr. dell'Associaz.Ital. di Protezione contro le Radizioni, Vol. I, 503, (1983).
6) SCHULTE, E.H., SECONDINI, A., SCOPPA,P., Trasferimento del tecnezio attraverso le catene alimentari marine, Atti del XXII Congr. dell'Associaz.Ital. di Protezione contro le Radiazioni, Vol. I, 563,(1983).

COEFFICIENTS OF DISTRIBUTION AND ACCUMULATION OF K,Rb,Cs AND ^{137}Cs IN THE INTENSIVE POULTRY BREEDING CYCLE

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ABSTRACT

The concentration of K,Rb,Cs and the activity level of Cs-137 in samples from the intensive poultry breeding cycle (feed,meat,eggs), under the condition of chronic alimentary contamination is presented. Concentrations of Cs and Rb were determined by non-destructive neutron activation analysis,concentration of K by atomic absorption flame photometry and activity of Cs-137 by gamma spectrometric analysis. On the basis of these results,coefficients of distribution and accumulation were calculated. The distribution coefficients of the analysed stable isotopes in meat have values close to 1,whereas for various parts of egg these coefficients vary between 0.5 and 1.5. Significant differences in Cs-137 distribution in various parts of egg were established. The values of accumulation coefficients indicate that all analysed elements selectively accumulate in the meat of young birds (broilers), and Cs-137 accumulates in the egg white as well.

INTRODUCTION

Using the accumulation coefficients of microelements and radionucleides in the poultry and eggs,but depending on the category of poultry and kind of feed,it is possible to predict the concentration of microelements and activity of radionucleides in the meat and eggs. This prediction is achieved in a relatively fast way by measuring the microelements and activity of radionucleides in the feeds only. Besides other factors,the accuracy of this prediction depends mostly on the accuracy of measurements in the feeds. In addition,using the coefficients of distribution and accumulation, we learn about chemical and metabolic behavior of radionucleides and their isotopic and non-isotopic analogs within the organism. Having this in view, the purpose of our work was to determine the coefficients of distribution and accumulation of fission product Cs-137,its stable isotope and its two chemical analogs Rb and K in the meat and eggs of poultry in intensive breeding.

METHODS AND RESULTS

The method of collection,preparation of samples and determination of content of stable elements (NNAA,AA) and activity of Cs-137 (gamma spectrometric analysis) are given in previous papers (Djurić

and Ajdačić,1980;Djurić,1979). Using the obtained concentrations of K,Rb,Cs i.e. activity of Cs-137 in poultry meat,eggs and feeds, given in Table 1, we evaluated the following:

a) Distribution coefficients (Cd), which are the concentration ratios of K,Rb or Cs /g kg^{-1} f.s./, i.e. activity of Cs-137 /Bq kg^{-1} f.s./, in particular kinds of meat of the same category of poultry or parts of eggs; in the other case this was the concentration ratio of these elements i.e. activity of Cs-137 in the same kinds of meat of various categories of poultry (Fig.1);

b) Accumulation coefficients (Ca), which are the concentration ratios of K,Rb,Cs /g kg^{-1} f.s./ or activity of Cs-137 /Bq kg^{-1} f.s./, in different kinds of meat or in the edible parts of eggs and in the feeds used in the breeding process, or in production of eggs (Fig.2).

DISCUSSION AND CONCLUSIONS

On the basis of our results it follows that:

-the distribution of K is uniform for both kinds of meat at the broilers and at the laying hens, as it was expected;

-the distribution of Rb depends on the age of the birds and kind of meat: Rb is retained more in the white meat of the broilers than in the dark meat, whereas the hens the retention of Rb is stronger in the dark meat;

-the distribution of the stable Cs in the poultry meat is the same as the distribution of Rb; however, the retention of Cs in the white meat of broilers is more prominent with respect to Rb and to K. This is in agreement with the data in the literature on different metabolic behavior of these elements (Comar and Bronnen,1962; Phipps,1976);

-the retention of Cs-137 in the white meat is slightly larger than in the dark meat in both categories of poultry;

-the retention of Cs-137 is largest in the eggwhite, contrary to the stable Cs which is least retained in the eggwhite, among all three chemical elements.

Selective accumulation of K,Rb,Cs and Cs-137 is confirmed also by the accumulation coefficient (Fig.2). The accumulation of the studied elements from the feeds is more prominent in both kinds of meat at the younger birds (broilers), in agreement with more intensive metabolic processes. In this way, e.g., the Cs-137 is being accumulated about 5-6 times more in the meat of the broilers than in the meat of the hens. Most accumulation of all studied elements is in the eggs.

However, it should be pointed out that the largest accumulation of Cs-137 from the feed we find in the eggwhite, whereas we find most concentration of the stable Cs in the eggyolk . This points out different chemical forms of the stable and of the radioactive Cs-137.

Our results show that the eggwhite can be used as an indicator for the level of radioactive cintamination of poultry by the radionucleide Cs-137 introduced in the alimentary way during the intensive breeding .

REFERENCES

Comar C., Bronnen (1962) : Mineral Metabolism; Academic Press, New York.

Djurić G.,Ajdačić N. (1980) :Journal of Radioanalytical Chemistry, Vol.59, No. 2, p.435-443.

Djurić Gordana (1979) : Doctoral Thesis; Faculty of Veterinary Medicine, University of Beograd.

Phipps D.A. (1976) : Metals and Metabolism; Clarendon Press,Oxford.

TABLE 1. THE CONCENTRATION OF K,Rb,Cs [$g \cdot kg^{-1}$ FS] AND THE LEVEL OF ACTIVITY OF ^{137}Cs [$Bq \cdot kg^{-1}$ FS] IN FEED, MEAT AND EGGS

CHEMICAL ELEMENTS	OATS	FEED MIXTURE				POULTRY MEAT				EGGS	
		S	FMB	FMH	F	DB	WB	DH	WH	WHITE	YOLK
K	7,92 ±0,49	9,38 ±0,17	7,99 ±0,63	10,61 ±0,46	7,87 ±0,48	2,89 ±0,17	3,23 ±0,15	2,77 ±0,16	3,03 ±0,14	1,41 ±0,08	0,91 ±0,06
Rb $\times 10^{-3}$	7,4 ±0,9	7,5 ±0,9	9,1 ±0,8	10,4 ±0,7	7,6 ±0,9	4,3 ±0,6	5,7 ±0,7	5,1 ±0,8	4,5 ±0,6	1,0 ±0,2	0,98 ±0,15
Cs $\times 10^{-6}$	24 ±4	90 ±8	55 ±9	53 ±8	110 ±10	10 ±2	16 ±3	13 ±2	12 ±2	7 ±1	13 ±2
^{137}Cs	1,5 ±0,2	2,4 ±0,2	1,8 ±0,2	2,3 ±0,2	2,3 ±0,2	0,3 ±0,03	0,4 ±0,02	0,2 ±0,04	0,3 ±0,03	0,37 ±0,04	0,15 ±0,05

S - COMMERCIAL FEED MIXTURE „STARTER"
FMB - " " " " FOR BROILERS
FMH - " " " " LAYINGHENS
F - " " " „FINISHER"

DB - DARK MEAT OF BROILERS
WB - WHITE " " "
DH - DARK MEAT OF LAYINGHENS
WH - WHITE " " "

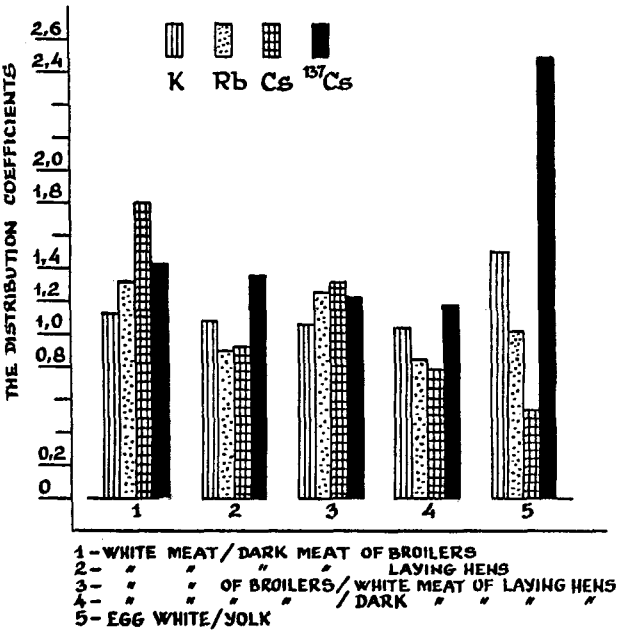

FIG.1 THE DISTRIBUTION COEFFICIENTS OF K, Rb, Cs AND ^{137}Cs SAMPLES FOR POULTRY MEAT AND EGGS

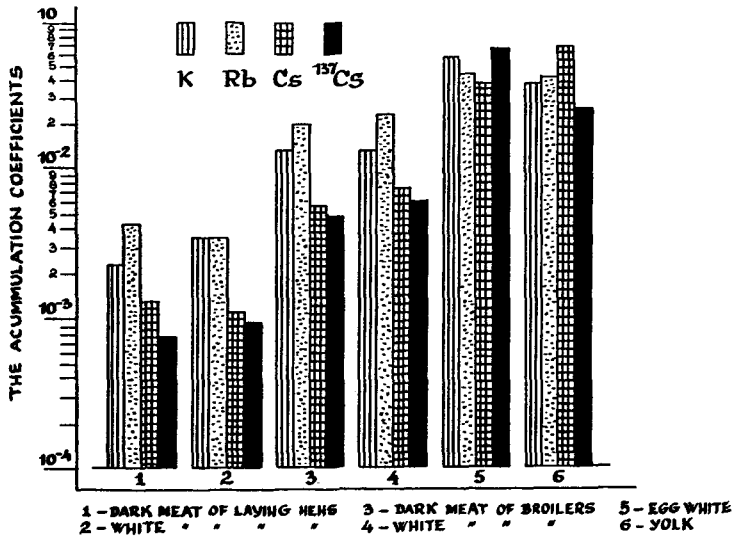

FIG.2 THE ACCUMULATION COEFFICIENTS OF K, Rb, Cs AND ^{137}Cs FOR POULTRY MEAT AND EGGS

ECOLOGICAL ASPECTS OF ATMOSPHERIC DISCHARGED TRITIUM IN THE VICINITY OF NUCLEAR FACILITIES IN JAPAN

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Environmental behavior of tritium in the form of water released into the atmosphere is strongly influenced by regional meteorological, geological, vegetational conditions and human activity. There are two groups of main tritium sources along the coast at the Tokai village, Ibaraki Prefecture in central Japan. One is composed of two heavy-water-moderated research reactors JRR2 and JRR3 which are neighboring in the site of Japan Atomic Energy Research Institute (JAERI). Another is a nuclear fuel reprocessing plant (NFRP) of Power Reactor and Nuclear Fuel Develpment Corporation(PNC). The JRR2-3 facilites are located at a distance of 1.8 Km north of the NFRP facility. The area concerned with the present study shown in Fig.1 includes resident area which is located between two groups of tritium sources and is likely to be subject to an influence of tritium released in the atmosphere mainly from the JRR2-3 facilities by north-east wind as annual average main direction in this region.

We have two rainy seasons every year in early summer and in early autumn, and have a rainy day every four days on the average and annual rainfall of approximately 1200 mm in recent years. Therefore, wash-out deposition of tritium may play an important role to understand the ecological behavior of tritium.

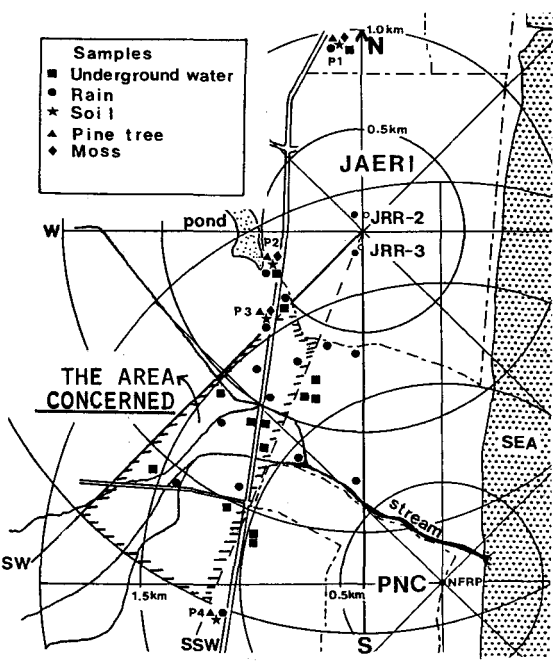

Fig.1
Sampling points around the area concerned

The uppermost of soil layers is composed of a loose sandy soil, covering almost the ground surface of the area concerned and ranging several meter in thickness. Then, water may move downward in the soil relatively fast. Most residents living in this area utilize underground origin tap water supplied by the village waterworks which is located 2.5 km SSW of the JRR2-3. Some of the residents have additional their own wells and use the underground water taken from 10 to 30 m depth for some purposes such as irrigation water. Therefore, the underground water may also be an important component for both ecological aspects and dose estimation of tritium in this area.

Tritium was released into the atmosphere amounting to 24.6Ci 1981 and 96.0Ci in 1982 as all vapor form from the JRR2-3 facilities, and amounting to 78.1Ci in 1981 and 120.6Ci in 1982 as a sum of HTO and HT forms from the NFRP facility.

For the purpose of investigating the consequences of tritium releases as well as the radio-ecological behavior of tritium, the periodical and simultaneous sampling was carried out at selected points inside and also outside of the area concerned from the beginning of 1982 and tritiated water content was measured by means of liquid scintillation counters for the samples of monthly rain, underground water, extracted water obtained by vacuum distillation from soils, pine needles and branches, and mosses. The detection limit was attained about 20 pCi/l by means of a large volume and low background liquid scintillation counter (LSC) ALOKA LB1 using a 100 ml PTFE vial and about 60 pCi/l by a low background LSC ALOKA 600LB using a 20 ml PTFE vial. The limit was improved to about 3 pCi/l by connecting the LS counting with 20 times electrolytic enrichment applied for some of fall-out level samples of monthly rain, underground and river waters.

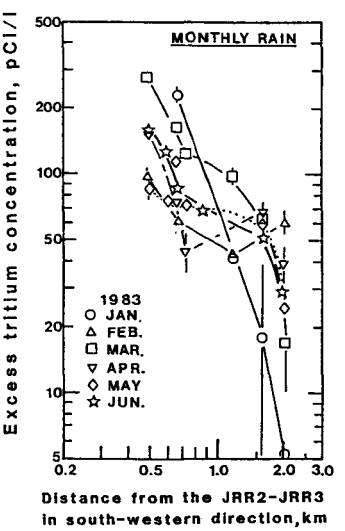

Fig.2
The dependence of the ^3H concentration of the monthly rain on the distance SW-SSW of the JRR2-3.

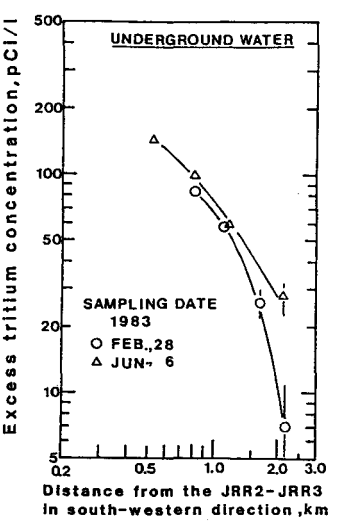

Fig.3
The dependence of the ^3H concentration of the underground water on the distance SW-SSW of the JRR2-3.

Results and Discussions

1. Tritium concentration-source distance relationships in rain and underground water

Chiba city is located in the distance of 100km south of the Tokai village and is considered to be subject only to the fall-out tritium of nuclear detonations. The fall-out level in Chiba was in the range from 25 ±3pCi/l at April to 42 ±3pCi/l at June during the first half of 1983. Excess tritium concentration of rain defined here is the differece of tritium concentration between rain collected at the area and Chiba and then it is considered to show the contribution of nuclear facilities. The excess tritium concentration in rain collected at the area decreased with the source distance SW-SSW of the JRR2-3 in most months of 1983 as shown in Fig.2. These observations agree with the usual dispersion theory and suggests that the area is mainly under the influence of the JRR2-3.

The exess tritium concentration of underground water collected twice in 1983 in the area decreased with the source distance SW-SSW of the JRR2-3 as shown in Fig.3. The excess tritium concentration described here is the differnce of the tritium concentration between underground water collected at the area and inland far more than several kilometer from the JRR2-3. The tritium concentration of the inland underground water is considered to be the fallout origin and was 68 ±3 pCi/l on Feburary 28 and 60 ±2 pCi/l on June 6, 1983. There was substantially no differnce in the concentration-source distance relationships in two times sampling of underground water. The excess tritium concentration of underground water decreased from about 150 pCi/l at the distance 0.5 Km to about 30pCi/l at the distance 2.0km in the same tendency as rain collected during the same period. These results suggest that tritium in the underground water may have its origin in the tritium released from the JRR2-3 before December 1982.

2. Relative correlations in the variation of tritium concentration among terrestrial environmental materials.

The variation of tritiated water concentration in some components of terrestrial ecosystem was observed by monthly and simultaneous sampling at selected points, 1.0km N (P1), 0.46km WSW (P2), 0.66km SW(P3) and 2.0km SSW (P4) of the JRR2-3. Some results are shown in Fig.4 obtained at the point P3 and in Fig.5 at the point P2, respectively. After the higher wet deposition of 421nCi/m^2 or 2570pCi/l of the rain concentration at the point P3 with the rainfall of 164mm in June, We had the lower wet deposition of 73nCi/m^2 or 570pCi/l with the rainfall of 126mm in July and 8nCi/m^2 or 42pCi/l with the rainfall of 184mm in August in 1982. The tritium concentration of the free water of pine needles collected at the point P3 decreased exponentialy as shown in Fig.4 from 2500pCi/l on June 28 to 300 pCi/l on August 31 passing through 500 pCi/l on August 5. This led to an apparent half residence time 14 days for tritiated water of pine needles which was comparable to the apparent half residence time of about 10 days obtained from tritium in the transpiration water of the pine needles at the same point. Both values seem to follow the decreasing rate of tritium in the rain described above. These observations suggest an important tritium pathway of precipitation-soil-plant and also rapid equiribrium in the tritium concentration between the free water of pine needles and the rain under the conditions of much rainfall including much tritium in the summer. On the contrary, under the circumstances of

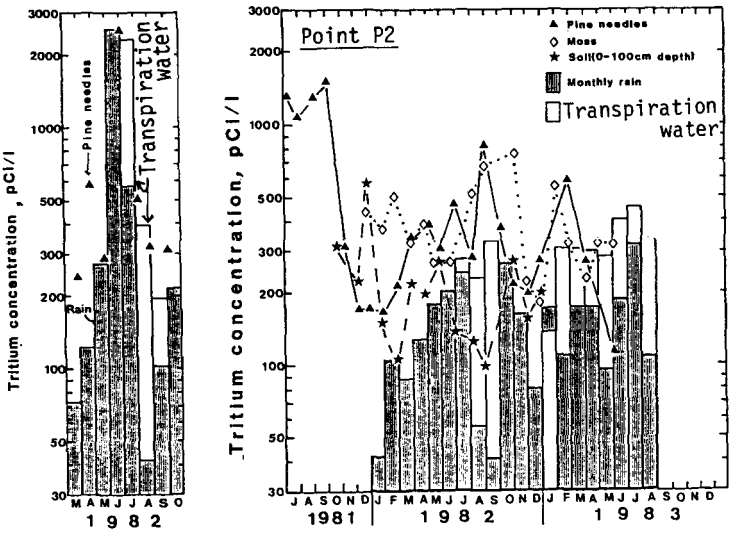

Fig. 4(left) and Fig.5(right)
The variation with time of the ^{3}H concentration of some components of terrestrial ecosystem collected at the point P3(left) and P2(right).

less tritium wash-out deposition at the point P2 as shown in Fig.5, the tritium concentration of monthly rain almost always lower than those of both the free water and the transpiration water of pine needles, which varied in coincidence with each other in most cases. This fact suggests that the atmospheric vapor of higher tritium concentration than that of the soil or rain plays an important role as an adjacent compartment of the tritium pathways to pine needles.

The tritium concentration of the free water of the moss showed higher than those of the monthly rain and the soil moisture and holded the same level as that of the pine needles as shown in Fig.5. The differnce in tritium concentration was observed sometimes between the moss and the pine needles. This result may be attributed to the fact that the roots of the moss distribute in the surface of the soil, then the moss is more sensitive to the atmospheric tritiated vapor. These result suggest that both the pine needles and the moss may become good indicators for the tritiated vapor in the atmosphere.

The depth profile of the tritium concentration of moisture of the soil taken by 20cm depth from the surface to 100cm below was observed during 1982 at P2. The mean concentration of tritium weighted by the moisture content was shown in Fig.5 and it seems to show almost the same level as the monthly rain, but not always to follow the vaiation of the monthly rain collected one month before the soil sampling. The profiles of the tritium concentration in the soil also gave less useful or analyzable informations on the tritium downward flow velocity or the residence time. The reason of these results is considered to be due to relatively fast exclusion of the tritiated moisture from the upper soil caused by the fast downward movement of the moisture through the porous sandy soil and by sucking up the soil moisture through plant roots enhanced by the transpiration.

MEASUREMENT CONCERNING THE CONVERSION OF ELEMENTAL IODINE DURING DISPERSION IN THE ATMOSPHERE

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1. Introduction

Pollutants reach vegetation, soil and surface waters via deposition from waste air plumes and thus enter into the food chain. Dry deposition (fallout) is today conventionally described by the deposition velocity and deposition by precipitations (washout) by the washout constant.

As a critical nuclide, I 131 plays a particular role both in normal operation and also during accidents in nuclear engineering facilities. In the case of radioiodine a differentiation must be made between elemental, aerosol (attached to aerosols) and organically bound iodine, since according to /1/ the deposition velocities and the washout constants, and thus also the ingestion doses, vary by approximately one order of magnitude for each of these forms of iodine. Elemental iodine accordingly proves to be the radiologically most important form of iodine.

The composition of the forms of iodine was measured during emission /2,3,4/. No answer has yet been found to the question in how far elemental iodine is converted into other forms of iodine during transport in the atmosphere. Experimental studies into this problem are carried out at the Jülich Nuclear Research Centre.

2. Test Method

Iodine in elemental form was released from the 50 m platform of the meteorological tower at the Jülich Nuclear Research Centre and air samples were taken downwind at distances between 200 and 1000 m with a tripartite filter system.

In order to produce elemental iodine a maximum of 100 mCi of I 123 (half-life = 13.2 h), supplied in 5 ml of carrier-free KI solution, was put into a reaction vessel in which 5 ml of saturated $K_2Cr_2O_7$ solution and 1 mg I_2 had previously been evaporated to dryness. At the time of emission the reaction flask was heated up to approx. 300 °C and the resulting iodine was expelled with a gas flow. The additional iodine concentration in air thus generated is comparable to the natural concentration.

The two sampling stations were switched on at the beginning of emission and shut down again 5 minutes after ending emission. The air first flowed through an aerosol filter at a flow rate of about 10 m^3/h, then through an I_2 adsorber (Südchemie DMS 11)

and finally through activated charcoal treated with KI. The residence time of the air in the adsorbers was 0.1 s. First the aerosols, then elemental iodine and finally the remaining gaseous iodine were removed from the air stream in this filter system and each form of iodine was separately measured by means of gammaspectrometry after completing the experiment.

The residual gaseous iodine collected in the activated charcoal filter mainly consists of organically bound iodine. However, it may also contain other iodine compounds with similar physico-chemical properties, thus causing similar radiation exposures. In the following this iodine fraction is designated organic iodine.

The series of experiments, which has not yet been completed extends over several years for the following reasons. In the first place, the number of experiments per year is limited to 5 by the licensing authority. However, this number cannot even be completely exploited since the possible production dates for I 123 seldom coincide with suitable diffusion conditions.

By using short-lived iodine 123 the potential radiation exposure to the public is greatly reduced in comparison to I 131. On the other hand, the short half-life of I 123 leads to the fact that due to the limited measuring capacity during one experiment air samples can only be taken and measured at a maximum of two locations.

3. Results and Discussion

The results obtained up to now and the meteorological parameters measured at the same time are compiled in Table 1 and Table 2. Apart from the wind direction and wind velocity, determined at release height of 50 m, all other meteorological parameters refer as usual to measurements at 2 m above ground level.

The results of the iodine measurements already show iodine conversions at a very short distance. It can be seen from Table 1 that, as a rule, a larger and larger fraction of elemental iodine is converted with distance and that the percentage of organic iodine increases. However, conclusions cannot yet be drawn from the few measurements available about whether there is a distance after which an equilibrium distribution of the forms of iodine is established, where this distance is and what form the equilibrium distribution takes.

A suspected correlation of the degree of conversion e.g. with the air temperature or reaction time (last column of Table 1) resulting from the distance of the field point from the source and the wind velocity has not been established. All the measurements have therefore been combined and the mean values, standard deviations and fluctuation ranges given below were determined. As an average over all the measurements

Table 1 Results of the Iodine Measurements

Experiment No.	Date	Time	Distance (m)	Relative Iodine Fractions (%)[1]			Reaction Time (s)
				aerosol	elemental	organic	
1	16.12.80	12.00-13.00	290	47	41	<12	88
2	14.07.81	14.00-15.10	250	8	39	53	46
3	13.10.81	12.00-13.05	220	59	22	18	25
			850	10	12	78	96
4	30.06.82	11.20-12.25	400	28	45	27	93
			1100	45	45	10	256
5	21.09.82	12.50-13.50	1000	12	86	< 2	123
6	20.10.82	11.15-12.15	430	15	83	< 2	84
			915	16	58	<25	179
7	15.06.83	11.45-12.32	230	16	66	<18	38
8	13.09.83	14.21-15.00	370	31	64	5	69
			450	29	60	11	83
9	20.09.83	12.50-13.20	400	47	51	2	67
			480	45	43	11	80

1) "<" smaller than detection limit

Table 2 Meteorological Conditions During the Experiments

Experiment No.	Wind Direction (°)	Wind Velocity (m/s)	Pressure (Torr)	Temperature (°C)	Humidity (%)	Diffusion Category
1	289	3.3	760	5.0	68	D
2	265	5.4	756	21	67	C
3	242	8.9	749	7.2	78	D
4	244	4.3	759	17.1	63	C
5	221	8.1	747	20.3	60	D
6	213	5.1	756	14.9	60	C
7	290	6.1	758	14	57	C
8	248	5.4	754	13.9	76	C
9	263	6.0	761	16.4	59	B

only about 50 % elemental iodine was detected, whereas about 30 % was accounted for by aerosol and about 20 % by organic iodine.

iodine form	relative fractions (%)		
	arithmetic mean	standard deviation	fluctuation range
aerosol	29	17	8 - 59
elemental	51	13	12 - 86
organic	20	22	<2 - 78

During transportation of iodine in the atmosphere every form of iodine can in principle be converted into every other. An equilibrium is finally established which can be dependent, amongst other aspects, on the temperature of the air, the intensity of cosmic radiation but also on the reactants or catalysts present in the air. It is therefore to be expected that each measurement of forms of iodine in the vicinity of a point source will provide different results. This explains the large standard deviations.

4. References

/1/ Der Bundesminister des Innern, Allgemeine Berechnungsgrundlage für die Strahlenexposition bei radioaktiven Ableitungen mit der Abluft und in Oberflächengewässer, Gemeinsames Ministerialblatt 1979, 371

/2/ H. Deuber, Die physikalisch-chemischen Radiojodkomponenten in der Abluft eines Druckwasserreaktors, KFK Report 3207, 1981

/3/ H. Deuber, Die physiaklisch-chemischen Radiojodkomponenten in der Abluft eines Siedewasserreaktors, KfK Report 3424, 1982

/4/ K. Heinemann, Messung von elementarem, aerosolförmigem und organischem Jod in der Abluft, Proceedings of 5th Fachgespräch "Überwachung der Umweltradioaktivität", 22-24 March, 1983, in Karlsruhe, to be published

A STUDY OF ELEMENTAL IODINE UPTAKE BY PLANTS, INFLUENCE OF SULFUR DIOXIDE AND MOIST LEAVES

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Due to the development of various industries in the vicinity of nuclear power plants, there may be some interaction between sulfur dioxide, water vapor and radioiodine released into the atmosphere.

This interaction may lead to a modification in the deposition rate of radioiodine onto vegetation. In order to investigate the processes that are involved in such an interaction, laboratory experiments were undertaken, in which plants (Helianthus annua and Phaseolus vulgaris L.) were exposed to low concentrations of radioiodine and sulfur dioxide, under high moisture conditions. Experimental methods are described and results discussed for application to real environmental situations.

INFLUENCE OF AGING EFFECTS ON THE SOIL-PLANT RADIONUCLIDE TRANSFER AFTER APPLICATION OF CARRIER-FREE Sr-90, Cs-137, Co-60 AND Mn-54 TO THE SOIL

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1. INTRODUCTION

The "aging effect" of the radionuclides ^{90}Sr, ^{137}Cs, ^{60}Co, and ^{54}Mn after their addition to soil as carrier-free solution /1/ has been studied by their uptake from two German soils into young barley plants. These plants have been used as biological indicator. Therefore, as well the biological status of the plants as the environmental conditions had been kept as constant as possible. Variations from the growth season and from the internal resources of the seed had been excluded by planting pregerminated seedlings into the soil.

2. EXPERIMENTAL PROCEDURE

Two parallel sets of experiments have been conducted: One using 0.37 kBq ^{90}Sr per kg of dry soil as nitrate, another using 0.37 kBq ^{137}Cs, 0.19 kBq ^{60}Co and 0.19 kBq ^{54}Mn per kg of dry soil as chloride. ^{90}Sr has been detected by its Cerenkov radiation, the others as γ-emitters by Ge(Li)-counting. The carrier-free solution was first applied to a small portion of the soil, and then mixed intensively with the final amount of the soil including 2 g NPP fertilizer per kg dry soil. The experimental pots contained 1.8 kg of dry soil and 10 barley seedlings were planted one week after sowing. The evaporation of the bare soil and the transpiration of plants were determined daily. The daily water loss was replaced by irrigation via a porous tube, which directed the water towards the bottom of the experimental pot.

The summer barley seedlings were planted 0, 1, 2, 5, 8, 14, 28 and 56 days after the contamination of the soil with the four radionuclides and was harvested 2, 4 and 6 weeks later (4 replicates always). The contaminated soil was used again after 9 and 22 (^{90}Sr) or 12 and 23 (^{137}Cs, ^{60}Co, ^{54}Mn) months, respectively. Proceeding each reuse 1 g NPP-fertilizer/kg dry soil was added.

Two soil types (Ap-horizon) have been tested: a podsolic soil from Gorleben (Lower Saxonia) and a degraded loess (parabraunerde) from Eschweiler (Northrhine-Westphalia) /2/. The experiment was conducted in a growth chamber: maximum water capacity of the soil 60 %, light/dark-period 10/2/10/2, day/night 22°/10°C, relative humidity 65 %/85 %, light 30 klux.

3. TEST OF THE EXPERIMENTAL CONDITIONS

Using as well the biomass as the uptake of ^{90}Sr as indicators a soil mass of about 1.8 kg had been demonstrated as sufficient, to get plants under non-limited conditions of growth and nutrient uptake.

The difference of the ^{90}Sr uptake between sown and planted barley seedlings is compared in Figure 1. The length of the plants have been taken as measure of their physiological status. The uptake of ^{90}Sr by germinating seeds from the nutrient-rich parabraunerde is reduced, while the planted seedlings show a more constant ^{90}Sr uptake. This difference could be explained either by plant-physiological differences in the mineral metabolism between seedlings and young plants or by the use of internal resources.

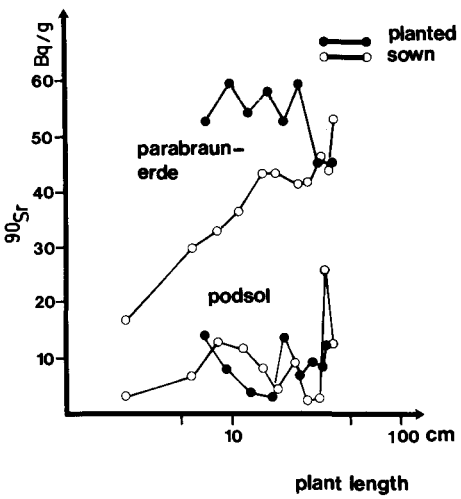

Figure 1: Comparison of the uptake of ^{90}Sr by young summer barley (Bq/g dry matter), which has either been sown or planted in two German soils (podsol from Gorleben, parabraunerde from Eschweiler). The length of the plant (excluding roots) has been used as an indicator of the status of development.

^{90}Sr is distributed homogeneously within each leaf of the young barley plants (radioautography). At this stage of development no enrichment at the site of intensive transpiration was detected. Only traces of ^{90}Sr could be detected in the guttation fluid.

4. NUCLIDE UPTAKE BY PLANTS FROM THE SOIL

The radionuclide uptake is expressed as radioactivity per g dry matter (plant parts above ground). Figure 2 compares the radionuclide uptake into barley seedlings, which were planted 9 (^{90}Sr) and 12 (other radionuclides) months after soil contamination, and harvested 2, 4 or 6 weeks after their plantation. No significant difference between the activities of the different harvests can be seen. Additionally, varying irrigation conditions (daily weight correction without or including the increasing water volume contained in the biomass) did not show any influence on the radionuclide uptake.

The comparison of the uptake of radionuclides from the two different soils can be seen in Figure 3 (only data of the 6-weeks harvested). If one considers only the first use of soil after initial contamination, generally a constant uptake of radionuclides can be seen. For some radionuclides the uptake curves show a

slight increase. This may be due to the increasing development of the root system and changes of the soil chemistry, which was indicated by pH-variations. The pH of the podsol remained almost constant after fertilization, while the pH of the degraded loess soil (parabraunerde) decreased during the first three weeks after fertilization from pH 6 to pH 4.8. After planting the barley restored the pH back to pH 6 within three weeks. It could not be elucidated whether the root development contributed to this change.

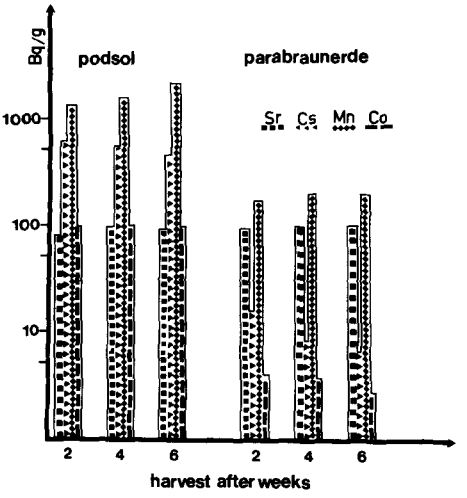

Figure 2: Radionuclide uptake of summer barley seedlings from two german soils. The plants were harvested 2, 4 and 6 weeks after plantation.

5. DISCUSSION AND CONCLUSIONS

Variations of the radionuclide uptake by barley seedlings were observed mainly by plants, which were grown in a degraded loess soil (parabraunerde). In this soil the most significant pH-variation has been observed. The decrease of the radionuclide uptake during the second and during the third use of the initially contaminated soil may be the result of chemical changes in the soil, indicated by the decrease of the pH. A supplementary fertilization and, in the third test an addition of 1 g $Ca(OH)_2$ per kg dry soil, was necessary to reestablish the pH and to ensure comparable growth conditions.

In summary, plant experiments on radionuclide uptake using small soil volumes may be used to interprete data from lysimeter or field studies, but one has to be aware of some problems: One can grow a large number of small plants, but only

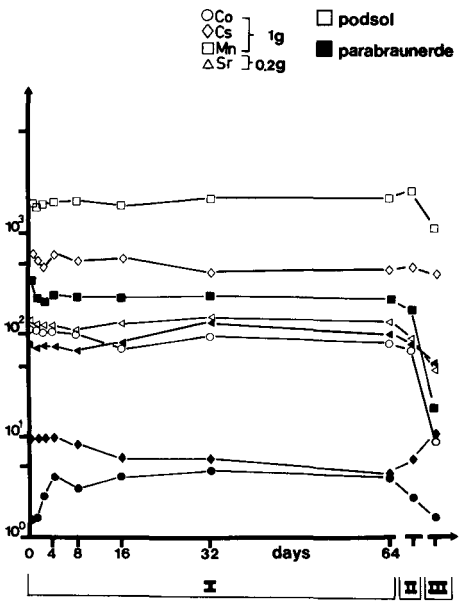

Figure 3: Radioactive uptake of summer barley seedlings, harvested 6 weeks after plantation. The symbols I, II and III indicate the repetitive use of the same soil (IInd use: 9 (^{90}Sr) and 12 (others) months after contamination /IIIrd use: 22 (^{90}Sr) and 23 (others) months thereafter). Preceding the second and the third use a fertilization of the soil was applied. The activity is given in Bq per 0.2 g (^{90}Sr) and 1.0 g (others) dry matter.

during a limited period of time. The roots trend to grow towards the bottom of the experimental pots. Due to the limited availability of water in the soil daily watering, sometimes twice a day, enforces desorption of the adsorbed radionuclides, which could lead to an increased root uptake. The rapid turnover of nutrients and the intensive growth of roots alter the chemical status of the soil. Therefore, an "aging effect" must be separated from variations of the soil chemistry by microorganisms and by roots.

5. LITERATURE

/1/ J.F. Cline: Aging effects of the availability of strontium and caesium to plants. Health Phsics 41 (1981), 293-296.

/2/ F. Führ: Die Aufnahme von Radionukliden aus dem Boden - Ermittlung praxisgerechter Transferfaktoren. Fachtagung Radioökologie des Dtn. Atomforums, Vulkan Verlag Essen (1975), p. 147.

RADIONUCLIDE TRANSFER OF SR, CS, CO, AND MN TO PLANTS GROWN ON SOILS WITH DIFFERENT PHYSICAL AND CHEMICAL PROPERTIES AND FROM DIFFERENT SITES AT ESCHWEILER, GORLEBEN, BIBLIS, AND STADE, F.R.G.

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In the vicinity of nuclear facilities, the soil types used agriculturally differ widely with respect to their physical and chemical properties. These soil properties, especially pH, clay and organic carbon content, and exchangeable calcium, influence the availability of radionuclides deposited on the soil, and subsequently also the soil-plant radionuclide transfer. Results from literature /1/2/3/ and from own lysimeter experiments /4/5/ show a considerable variation of the radionuclide transfer, mainly due to the different soil properties. To evaluate this variation for Sr-85, Cs-137, Co-60, and Mn-54 within one site and between several sites, a pot experiment with different soils from four different sites in the Federal Republic of Germany was conducted.

EXPERIMENTAL PROCEDURE

In this experiment 12 different soils (Ap-horizon) representing peaty, clayey, loamy, and sandy types were used: one soil each from the region of Eschweiler (soil No. 1.1) and from the future storage facility site at Gorleben (soil No. 2.1) and 6 and 4 soils, respectively, from the nuclear power station sites at Biblis (soils No. 3.1 - 3.6) and Stade (soils No. 4.1 - 4.4). These soils are differing widely with respect to their physical and chemical properties. The characterization of the soils and their major physical and chemical properties are presented in table 1.

The radionuclides, carrier-free Sr-85, Cs-137, Co-60, and Mn-54 in chloride form, were mixed into the soil uniformly. The radioactivity per kg of soil amounted to 206 KBq for Sr-85, 470 KBq for Cs-137, 394 KBq for Co-60, and 180 KBq for Mn-54. The contaminated soils were filled into dubble wall experimental pots: 8 kg per pot and 8 replicates per soil type. The soils were fertilized with a complex fertilizer (10 % N, 10 % P_2O_5, 17 % K_2O, 2 % MgO) before cultivation of each crop at a rate of 1 g/kg. Soil humidity was controlled at least three times per week and regulated at 50 % of maximum water capacitiy.

The experiment was conducted in a cold green house. The following plants were grown in a 2 years crop rotation: Spring wheat, lettuce, potatoes, and beans (phaseolus vulgaris). The crops were harvested being ready for consumption, dryed and ground. Aliquots of this plant material were measured in a well type Ge(Li)-detector. The results presented as radioactivity concentration (Bq/g dry plant material) were computed statistically. The significance of differences was checked using t-test.

RESULTS AND DISCUSSION

Normally, the transfer factor is used to indicate the entrance of radionuclides from soils via plants into food chains. However, transfer factors based on results from pot experiments generally do not deliver realistic values to estimate the irradiation of man due to ingestion /6/. Therefore, the magnitude of radionuclide transfer from soil to plants is reported as radioactivity concentration measured in distinct plant parts and presented as Bq/g dry plant material. In spring wheat grain and potatoe tubers a lower radioactivity concentration was measured for Sr-85, Cs-137, Co-60, and Mn-54 than in lettuce (Table 2-4). Considering the radioactivity amount applied per kg of dry soil, the lowest concentration was found for Co-60. For the other radionuclides increasing values were registered in the range Cs-137 - Mn-54 - Sr-85.

Table 1: Origin, characterization, and major physical and chemical properties of the soils (Ap horizon)

Soil No.	Origin and site		Characterization	Clay <2 µm %	Silt 2-60 µm %
1.1	Eschweiler (NRW)		Typic hapludalf	12.0	82.6
2.1	Gorleben		Plaggeptic haplohumod	2.6	5.1
3.1	Rhein, km 453,	Biblis	Mollic udifluvent	13.3	25.2
3.2	Griesheim,	"	Alfic udipsamment	6.4	10.4
3.3	Weilerhof,	"	Typic eutrochrept	24.4	22.5
3.4	Wattenheim,	"	Aquic hapludoll	17.1	26.3
3.5	St. Stephan,	"	Lithic udipsamment	3.4	8.5
3.6	Nuclear power Stn.,	"	Typic haplaquoll	17.6	20.7
4.1	Schloß Agathenburg,	Stade	Histic humaquept	0.7	22.7
4.2	Barnkrug,	"	Aquic udifluvent	19.8	60.2
4.3	Hammah,	"	Plaggeptic udipsamment	8.6	20.0
4.4	Brobergen,	"	Plaggeptic haplohumod	2.4	1.9

Soil No.	Fine sand 60-200 µm %	Med./coarse sand 200-2000 µm %	Total C %	pH($CaCl_2$)	Exchange capacity meq/100 g of soil	Ca^{++}	K^{+}
1.1	4.0	1.4	1.4	6.2	11.2	11.8	0.8
2.1	32.1	60.2	1.1	4.7	6.2	0.8	0.2
3.1	23.9	37.6	1.4	7.5	15.7	12.2	0.7
3.2	70.9	12.2	0.8	6.7	6.8	5.6	0.5
3.3	42.5	10.6	1.3	7.4	14.3	12.3	0.9
3.4	23.1	33.5	1.1	7.4	11.9	10.1	0.8
3.5	74.1	14.1	0.4	7.6	3.3	2.0	0.3
3.6	19.5	42.2	1.1	7.4	13.2	8.7	1.3
4.1	25.0	47.7	14.7	5.2	66.2	55.0	0.3
4.2	19.0	1.0	2.5	7.2	17.6	15.3	0.1
4.3	36.6	34.9	2.1	4.1	10.0	1.7	0.2
4.4	19.2	76.5	2.7	4.3	6.6	2.6	0.1

The mean values for the radioactivity in the edible plant parts computed from 8 replicates of one soil show a low variation with one exception (Co-60 in lettuce, Table 2-4). However, a partly considerable variation can be observed between the radioactivity concentrations in plants from different soils of one site. For Sr-85 and Cs-137 this variation is in the same order at the sites of Biblis and Stade, but for Co-60 and Mn-54 the variation is considerably higher in plants grown on soils from the site of Stade. Probably this is due to the wide variation of the physical and chemical soil properties (Table 1).

Between the different sites, for Sr-85, the radioactivity concentration in plants shows only a low variation, whereas for the other radionuclides increasing differences in the range Co-60 - Mn-54 - Cs-137 are determined (Table 2-4). Comparing all radioactivity concentrations of one radionuclide determined in edible plant parts produced on 12 different soils from 4 different sites, considerable differences can be registered reaching factors of 9-15 for Sr-85, 36-122 for Mn-54, 23-233 for Co-60, and 119-507 for Cs-137. For all radionuclides differences amounting a factor of 2 or more are significant (P = 1 %) or highly significant (P = 0,1 %).

Table 2: Transfer of Sr-85, Cs-137, Co-60, and Mn-54 from soils with different physical and chemical properties to spring wheat grain

Soil No.	Dry matter %	Bq/g plant dry matter							
		Sr-85		Cs-137		Co-60		Mn-54	
		$\bar{x}$	s	$\bar{x}$	s	$\bar{x}$	s	$\bar{x}$	s
1.1	94.0	23.6 ±	1.64	2.86 ±	0.27	1.04 ±	0.11	109 ±	8.99
2.1	95.2	43.2 ±	8.22	104 ±	7.61	56.0 ±	10.2	1613 ±	125
3.1	94.1	13.0 ±	0.83	9.42 ±	0.60	0.42 ±	0.04	35.1 ±	4.01
3.2	93.9	35.5 ±	4.64	3.32 ±	0.28	1.43 ±	0.20	62.5 ±	4.75
3.3	94.5	9.33 ±	0.95	0.46 ±	0.07	1.15 ±	0.32	37.7 ±	5.90
3.4	93.0	10.4 ±	0.62	1.12 ±	0.13	2.65 ±	0.16	53.6 ±	5.63
3.5	95.5	31.0 ±	5.47	4.72 ±	0.37	3.53 ±	0.32	79.9 ±	9.25
3.6	94.5	4.98 ±	0.73	1.13 ±	0.11	1.51 ±	0.24	62.1 ±	8.65
4.1	95.7	12.0 ±	1.65	97.6 ±	7.60	1.03 ±	0.19	156 ±	16.3
4.2	97.1	10.7 ±	1.34	47.7 ±	2.89	0.24 ±	0.03	17.6 ±	1.23
4.3	96.9	37.9 ±	3.19	14.6 ±	1.33	18.8 ±	5.07	399 ±	34.3
4.4	96.5	45.2 ±	5.14	233 ±	11.1	7.32 ±	1.84	970 ±	60.5

Table 3: Transfer of Sr-85, Cs-137, Co-60, and Mn-54 from soils with different physical and chemical properties to lettuce

Soil No.	Dry matter %	Bq/g plant dry matter							
		Sr-85		Cs-137		Co-60		Mn-54	
		$\bar{x}$	s	$\bar{x}$	s	$\bar{x}$	s	$\bar{x}$	s
1.1	5.58	792 ±	109	40.1 ±	11.9	6.38 ±	1.65	137 ±	12.0
2.1	6.01	2399 ±	115	549 ±	73.2	241 ±	39.3	2251 ±	332
3.1	4.40	498 ±	27.8	76.6 ±	6.27	7.61 ±	0.82	33.6 ±	4.60
3.2	5.65	951 ±	121	38.2 ±	6.55	15.2 ±	4.20	113 ±	16.7
3.3	5.43	378 ±	46.3	6.75 ±	1.39	5.13 ±	0.77	28.2 ±	3.72
3.4	5.14	708 ±	67.3	24.3 ±	7.35	14.1 ±	3.02	79.6 ±	9.76
3.5	5.68	1054 ±	119	56.5 ±	5.71	34.5 ±	12.2	128 ±	20.8
3.6	4.76	156 ±	9.97	13.7 ±	1.84	10.1 ±	1.32	55.4 ±	5.10
4.1	5.38	313 ±	33.7	142 ±	23.0	12.4 ±	3.18	235 ±	33.3
4.2	5.32	336 ±	24.4	251 ±	46.0	5.04 ±	0.93	18.4 ±	1.87
4.3	5.76	1601 ±	249	130 ±	12.3	50.0 ±	7.85	597 ±	158
4.4	4.71	1828 ±	248	802 ±	140	93.3 ±	16.6	1970 ±	314

The highest radioactivity concentrations for all radionuclides investigated are measured in crops grown on sandy soils with low pH values, low clay and exchangeable calcium content and low exchange capacity (soils No. 2.1, 4.4, 4.3, Table 2-4). In crops from soils with high pH values, high contents of clay and exchangeable calcium, and high exchange capacity low radioactivity concentrations are found (soils No. 3.3, 3.4, 3.6, 4.2, Table 2-4), as it is also observed in crops grown on a peaty soil with high content of organic carbon, low clay but high exchangeable calcium content and exchange capacity (soil No. 4.1, Table 2-4). As a fourth crop beans (phaseolus vulgaris) are grown on these soils. These results will be presented in the poster.

Table 4: Transfer of Sr-85, Cs-137, Co-60, and Mn-54 from soils with different physical and chemical properties to potatoe tubers

Soil No.	Dry matter %	Bq/g plant dry matter							
		Sr-85		Cs-137		Co-60		Mn-54	
		$\bar{x}$	s	$\bar{x}$	s	$\bar{x}$	s	$\bar{x}$	s
1.1	21.4			9.66	0.91	6.55	0.63	9.50	1.04
2.1	21.3			148	19.1	74.0	9.78	96.8	11.1
3.1	22.6			11.4	1.33	8.84	1.20	5.33	0.39
3.2	21.5			14.5	0.96	11.4	1.22	7.73	0.77
3.3	21.4	not		2.89	0.36	7.68	1.43	4.15	0.78
3.4	22.0	detectable		8.58	1.20	11.7	1.36	5.38	0.86
3.5	22.6	due to decay		15.8	1.22	16.5	2.54	11.5	1.86
3.6	23.2			4.11	0.69	6.24	1.04	3.47	0.73
4.1	22.5			23.3	2.84	5.55	0.56	10.1	0.46
4.2	20.9			37.6	5.91	3.26	0.31	2.72	0.20
4.3	20.8			43.2	4.38	32.7	4.25	35.1	2.35
4.4	20.8			358	40.7	54.3	5.91	67.6	8.41

CONCLUSIONS

According to these results the availability of radionuclides for plant roots in the soil is variing in dependence on pH, clay, calcium, and organic carbon content,and exchange capacity. In all crops the highest variation can be observed for Cs-137, and a lower variability for the other radionuclides decreasing in the range Co-60 - Mn-54 - Sr-85. Furthermore, the results of this experiment indicate, that within one nuclear facility site and between two different sites the possible variation for the radionuclide transfer from soil to plant might amount one order of magnitude each due to the different soil properties. In addition there is a variation of one order of magnitude caused by the plant species. These findings are in accordance with results from literature /1/2/.

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REFERENCES

/1/ Biesold, H.: Berichtsband des Radioökologiesymposiums, Stuttgart 15./16.10.1981.
/2/ Ng, Y.C., C.S. Colsher, S.E. Thompson: NUREG/CR-2975, UCID-19463, Lawrence Livermore National Laboratory, Livermore, CA, USA.
/3/ Wiechen, A.: Forschungsvorhaben St.Sch. 702c, Abschlußbericht, Institut für Chemie und Physik der Bundesanstalt für Milchforschung, Kiel, 1983.
/4/ Steffens, W., F. Führ und W. Mittelstaedt: Sonderdruck aus Jahresbericht 1979/80 der Kernforschungsanlage Jülich GmbH.
/5/ Steffens, W., W. Mittelstaedt, H.G. Ehrlich und F. Führ: Berichtsband der Gemeinsamen Strahlenschutztagung des Fachverbandes für Strahlenschutz e.V. und der Société Francaise de Radioprotection, Lausanne/Schweiz, 30.09.-02.10.1981.
/6/ Steffens, W., F. Führ, and W. Mittelstaedt: Proceedings of the 5th Congress of the International Radiation Protection Association, Jerusalem, 1980, Vol. 2.

VERTICAL MIGRATION OF HTO AND ^{137}Cs IN THE SOIL

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Abstract

From depth profiles of the HTO and ^{137}Cs concentrations in the soil the vertical migration rate was determined for these two radionuclides. For HTO the values assessed ranged from 2 to 5 m/a, for ^{137}Cs they were between 0.4 and 1.6 cm/a. The values found on the site of the Karlsruhe Nuclear Research Center are so low in absolute terms that they are insignificant from the radiation protection point of view.

Introduction

Radionuclides released into the atmosphere reach the surface of the earth as fallout and washout. The question of particular interest to the radioecologist is to which extent such deposits penetrate into the soil and may consequently enter the ground water. To investigate this question in more detail, a number of boreholes were drilled on the premises of the Karlsruhe Nuclear Research Center (KfK) /1/ and depth profiles determined for various radionuclides. Results of HTO and ^{137}Cs measurements will be reported here.

Tritiated water

In Figs. 1, 2 and 3 three selected depth profiles of tritium concentrations have been represented. To measure the tritium concentration boreholes were drilled down to the ground water. In the Karlsruhe Nuclear Research Center the ground water level lies at about 5 m depth. The selected depth profiles exhibit a clear dependence on depth of the tritium concentration. On the other hand, other boreholes yielded but insignificant variations of tritium concentrations with the depth. Not only the measured values have been entered in the figures but also the 2σ error bands of tritium measurement. The rate of migration, v_m, was determined from the figure for various pairs of measured values.

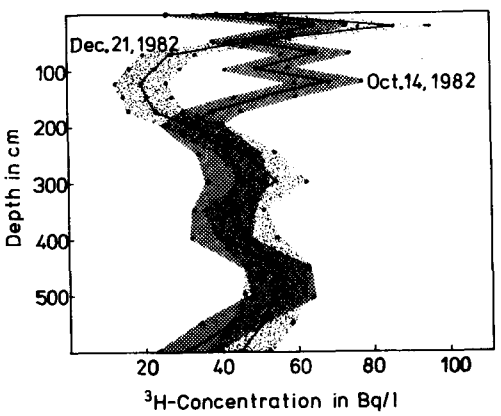

Fig. 1: Depth profile at the sampling point B3: v_m = 2.7–4.7 m/a.

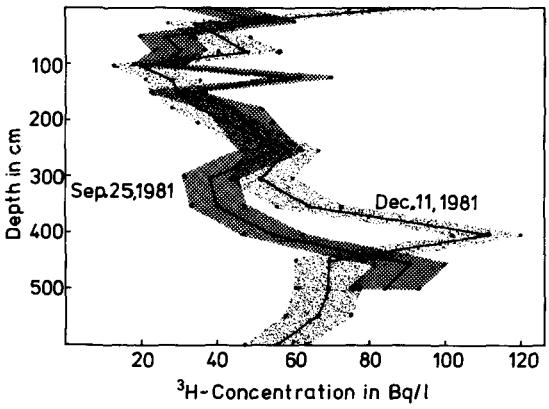

Fig. 2: Depth profile at the sampling point B3: v_m = 2.1-2.4 m/a.

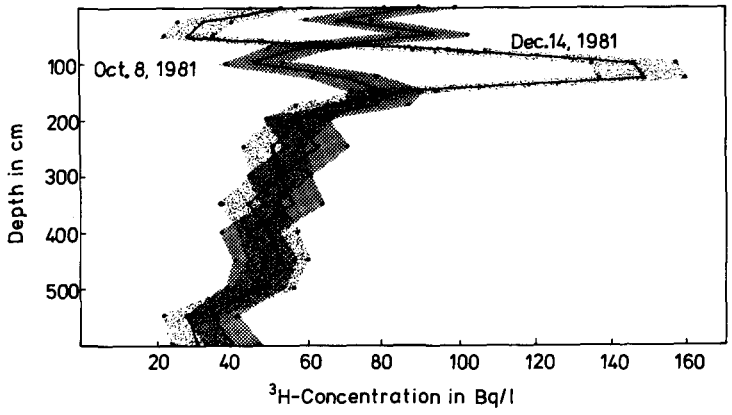

Fig. 3: Depth profile at the sampling point B1: v_m = 2.2-3.1 m/a.

In those cases where considerable variations had been found of tritium concentrations with the depth, there was a chance to investigate the vertical diffusion of tritium by repetition of the measurement at a later date. The examples described show that this type of investigation is not an easy task since the vertical movement of the humidity of the soil is superimposed by a diffusion blurring the depth profile. Moreover, experience has shown that, given the situation on the Karlsruhe site, the interval between two drillings must not be too long so that an observed peak of tritium concentration cannot reach the ground water in the time between drillings or disappear through diffusion which would inhibit an estimation of the vertical migration rate. By selection of conspicuous points on the plot and comparison with the results of measurements made in the hole drilled later values can be determined of the vertical rate of migration on the basis of two depth profiles. The migration rates so evaluated range from 2-5 m/a and have been indicated

individually in the legends to the respective figures. The values indicated for the migration rate have been extrapolated. It should not be expected at all that the vertical migration proceeds at a constant rate. On the other hand, the measured values agree well with those taken from /2/. An important result of the measurement for HTO in samples collected in the borehole is that the HTO concentration in the topmost ground water layer corresponds to that in the overlying soil.

The total tritium inventory down to the ground water level is estimated at 10-20 kBq/m^2 /1/. Assuming a migration rate of 3 m/a, HTO supply to the ground water amounting to 6-12 kBq/a is consequently obtained. If this activity were diluted with not more than 100 l ground water, the HTO concentration would be 60-120 Bq/l, i.e., an insignificant concentration from the point of view of radiation protection. The same result would have been obtained from a comparison of the tritium stored in the soil with the tritium uptake in the water and diet of 5.8 MBq/a, the value permitted in the Federal Republic of Germany /3/. Nevertheless, a possible effect on the ground water should be examined in case of major tritium releases.

^{137}Cs

^{137}Cs can be detected in many aerosols released from the Karlsruhe Nuclear Research Center. It is therefore of interest to measure the depth profiles for this radionuclide as well. The interpretation of the results of measurement in this case is complicated by the fact that a substantial portion of the ^{137}Cs detectable in the soil has its origin in nuclear weapons fallout. Figs. 4 and 5 show two of the measured depth profiles. It is evident from the figures that the vertical migration rate of ^{137}Cs is smaller by two to three orders of magnitude than that of HTO. If one relates the maximum depth of ^{137}Cs penetration into the soil of about 30 cm to the point of time of maximum activity deposit by nuclear weapons fallout, i.e., 1963, a maximum migration rate of about 1.6 cm per year is obtained for ^{137}Cs. If, instead, the maximum surface load per 3 cm of layer thickness had been attributed to the maximum exposure, a migration rate of 0.4 cm per year would have been derived for ^{137}Cs which is in agreement with the value published in /4/.

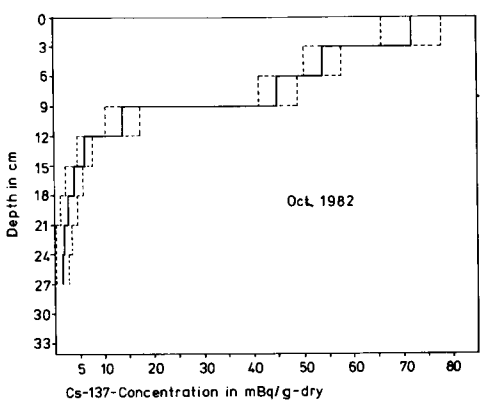

Fig. 4: Depth profile at the sampling point B1.

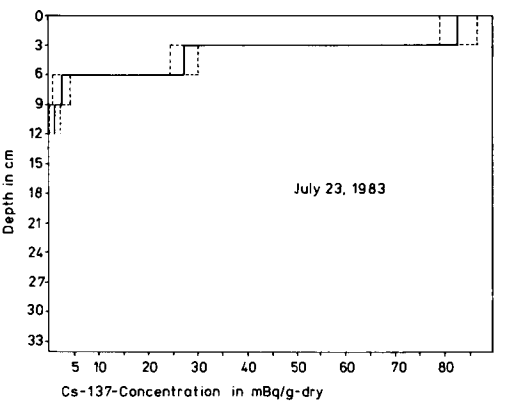

Fig. 5: Depth profile at the sampling point B1.

In this context reference should be made to the depth profiles published in /5/ for $^{239+240}$Pu. These depth profiles exhibit in conformity with /1/ that the vertical migration rate of plutonium does not greatly differ from that of ^{137}Cs. According to information from the literature (cf., e.g., /4/) a ^{137}Cs gross activity integrated over the depth of about 3 kBq/m^2 is to be anticipated on the KfK site for the time being. If one compares this value with the values of about 5 kBq measured on the premises of the Karlsruhe Nuclear Research Center, it gets apparent that the KfK contribution to the ^{137}Cs deposit is smaller than the contribution by nuclear weapons fallout.

Likewise, the comparison of deposits on the soil with the ^{137}Cs incorporation of 27 kBq/a, the value permitted in the Federal Republic of Germany shows, that the depositions are insignificant from the point of view of radiation protection.

Concluding Remarks

Considerable variations in the rates of migration have been found both for HTO and for ^{137}Cs. The results cannot yet be described by models since not all the parameters necessary are known. Nevertheless, the measured values are useful for estimating the radiation exposure via the air path resulting from the contamination of the ground water.

Literature

/1/ L.A. König, H. Fessler, K.-G. Langguth, D. Papadopoulos, A. Radziwill
Radioökologische Auswirkungen des langjährigen Betriebs der kerntechnischen Anlagen des Kernforschungszentrums Karlsruhe auf dessen nähere Umgebung - Befristete Einzeluntersuchungen in Erweiterung und Ergänzung des routinemäßigen Überwachungsprogramms
KfK-3459 (February 1983).

/2/ U. Zimmermann
Umschau in Wissenschaft und Technik, 75 (1975), p. 249.

/3/ Verordnung über den Schutz vor Schäden durch ionisierende Strahlen, issued on October 13, 1976 (BGBl. I, p. 2905) with the amendments of January 21, 1977 (BGBl. I, p. 1984) and of February 1, 1977 (BGBl. I, p. 269) taken into account.

/4/ M.J. Frissel, A.T. Jakubick
Transport and Accumulation of Radionuclides in Soil, Contribution to "Radioökologie", Berichtsband der Fachtagung des Deutschen Atomforums e. V., October 2-3, 1979, Wissenschaftszentrum Bonn, p. 131.

/5/ M. Pimpl, H. Schüttelkopf, K. Bender, G. Hefner, H. Bailer
Contribution to the Annual Report 1982 of Hauptabteilung Sicherheit (edits.: H. Kiefer, W. Koelzer, L.A. König), KfK-3535 (April 1983), p. 170.

OXIDATION OF HT-GAS IN THE SYSTEM AIR/SOIL

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1. INTRODUCTION

Tritium is a nuclide with a natural geochemical pathway: It is produced nearly exclusively in the higher atmosphere, oxidised at least to water and buried once in the groundwater and oceans until it decays. The calulation of the global inventory depends on the velocity of the supplying reactions: The range between 10^{18} to 10^{19} Bq seems to be the best one /1/. As a rough number about 10 kg tritium seems to be the natural global tritium inventory. It is predominantly bound to water and therefore part of the natural water cycle.

The major anthropogeneous tritium sources are bomb produced tritium, reactor technology and special applications (dyes, lamps). Further sources of HT and HTO are reprocessing of fuel elements, pilot projects in fusion technology and solid-state physics experiments.

Ten years ago a remarkable high content of tritium has been found in the atmospheric hydrogen compared to the water vapour. From experiments with inactive hydrogen and geochemical considerations it has been concluded that the soil organisms are the most important sink for gaseous tritium under natural conditions /2/. Further studies /3/ and experiments /4/ had focussed again the attention to the soil as the main sink of HT. In an air volume, which contains HT, its oxidation to HTO seems to be a very slow process, especially if the radioactivity is low /5/. The next step of interest in the radioecological pathway of tritium is the contact with the surface of plants and of soil. A first set of soil experiments was conducted to study the velocity of HT-conversion under well-defined conditions in a gastight experimental setup.

2. EXPERIMENTAL PROCEDURE

HT is taken from a glass container (1.1 L) with a stock supply of about 4×10^9 Bq HT (Amersham TRG 1). The container is sealed by a silicone stopcock, which is covered with mercury. The gas volume must be kept free from oxygen. The HT is taken by a syringe and fed into the reaction circuit through a silicone sealed injection stopcock. The volume difference is replaced by helium. About 2×10^6 Bq are used for each test run (reproducibility ±5 %).

The soil samples were taken from an air-dry stored parabraunerde (loess-type; Eschweiler, North-Rhine Westphalia, Fed. Rep. Germany). Some characteristic data are /6/: pH($CaCl_2$) 5.9/1.4 % C(organic)/12.0 % clay, 28.4 % silt, 58.3 % fine sand. The maximum water holding capacity is about 46 %, during the experiment it was adjusted between 18 and 19 %. The moist soil was transfered into the reaction vessel (12 cm diameter). While increasing the amount of soil the surface area remained constant. Four soil treatments were compared: sterilisation in an autoclav (120 °C), moistening with deionized water, addition of either a filtrate or a suspension of a garden soil. Moist soil has been stored two days in darkness.

The closed experimental circuit (see Figure 1) has a volume of 2.6 L and is made from glass and stainless steel. Some fittings contain either teflon or grease sealings. The soil in the reaction vessel (up to 1.0 kg) is kept under constant conditions (temperature 22°, relative humidity 75 %, air turnover in the circuit 40 l h^{-1}). After its contact with the soil the HT-concentration

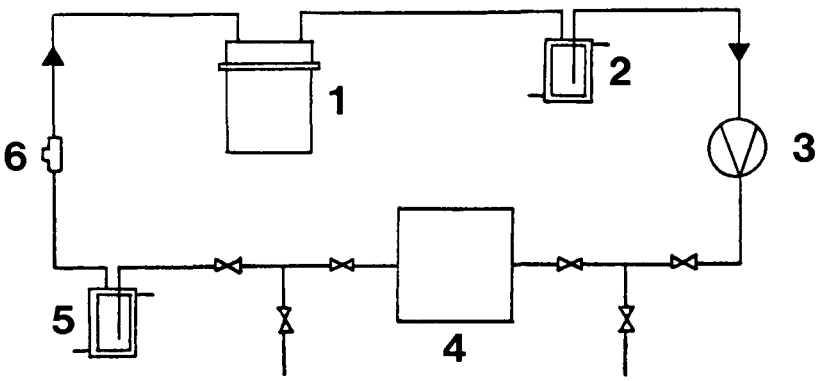

Figure 1: Scheme of the experimental circuit (1: reaction vessel containing soil, 2: cooling trap, 3: pump, 4: ionisation chamber, 5: dew point trap for air humidity, 6: injection point).

of the air is measured contineously by an ionisation chamber. Before entering the chamber the air must be dried by cooling traps, to separate HT and its reaction product HTO. The self-constructed ionisation chamber (stainless steel, 500 V, 65 mm diameter, 0.25 L volume, aluminium barrier to lengthen pathway) has an efficiency of 95 %, a sufficiently short response time, and no noticable memory effect. The resistance of the ionisation chamber can be read from a teraohmmeter Knick H 12 (range: 5×10^6 to 10^{16} ohm, detection range: 0.2 to 1×10^6 Bq l^{-1}). The air stream (driven by a metal bellow pump) is detected by a flowmeter. Leaving the ionisation chamber and the flowmeter the relative humidity of the air is adjusted by a dew point trap.

The tritium activity of all samples (air, water, organic material in the soil) is measured as water by LSC. Gas and organic material are burned in a stream of oxygen. The tritium content of HTO and of the organically-bound fraction are defined in the following manner: HTO is the portion, which can be extracted by a twofold vacuum distillation of water (second distillation after addition of inactive fluid). The remaining tritium burned in the oxygen stream is assumed to be the organically-bound fraction. After calibration of the oxidation all samples can be compared directly to get a tritium balance, to detect unnotified losses or sinks within the reaction circuit.

3. RESULTS

After a twofold sterilization of the soil the HT-concentration in the experimental setup shows only a very slow decrease during the first hours after HT-injection. Afterwards the HT-content of the air remains constant.

The loss of HT from the gas above the non-sterilized soil can be described by a simple exponential function (A : HT-activity):

$$A_t = A_o \, e^{-kt}, \qquad (1)$$

$$t_{1/2} = \frac{\ln 2}{k} . \qquad (2)$$

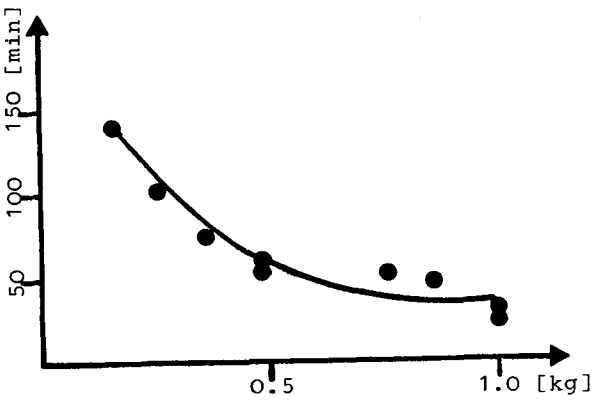

Figure 2:
Dependence of the half-time from the amount of soil in the reaction vessel (theoretical curve by least square method drawn as solid line) (soil moistened with deionized water).

This means, the HT-loss is proportional to the HT-concentration in the air. Also, only one reaction should be responsible for the HT-turnover. For there are different values of A_o (e.g. individual adjustment of the pressure in the circuit, differences in the amount of injected HT) only the half-time $t_{1/2}$ was taken as a measure of the turnover velocity. The half-time of the reaction in the presence of the biologically active soil should be

$$t_{1/2} \propto 1/Q, \qquad (3)$$

where $t_{1/2}$: half-time (min), Q : amount of soil (kg). The experimental results, using soil simply moistened with deionized water after storage under air-dryness, fit the equation (4):

$$t_{1/2} = 7.8 + 25 \times 1/Q \qquad (r = 0.98). \qquad (4)$$

The equation proposed above assumes a well-mixed gaseous atmosphere in the soil and consequently a homogeneously distributed reaction in the soil column. The limit of 7.8 minutes is significant from zero at the level of 80 %. This may be due to an incomplete mixing with the soil gas volume, or may be the limit of the whole experimental arrangement.

To support this consideration the profile of the resultung HTO in a soil column was analyzed (Table 1). The data show only a slow decrease of concentration in HTO with increasing soil depth. This leads to the conclusion, that the constant in equation (4) of about 8 minutes can be considered as the result of experimental limitations.

Table 1: HTO-activity (kBq/g dry soil) after reaction of soil (moistened with deionized water) with HT gas. Four segments of soil were taken in the center and near the border of the vessel from top (1) towards the bottom (4).

	sample from center		sample from border	
segment	activity	%	activity	%
1	2.1	100	2.1	100
2	1.8	86	1.6	76
3	1.8	86	1.5	71
4	1.6	76	1.5	71

Only less than 1 % was found as organically-bound fraction. Nearly all of : tritium gas was converted to tritiated water as the primary reaction pro- ɔt. This water remains mainly at its reaction site during the observation pe- ɔd. Only 1.7 ± 0.9 % of the total tritium could be detected in the cooling ap finally.

The addition of a filtered soil extract did not enhance the half-time of ıe reaction (Table 2), while a test run with a suspension of garden soil ıowed a rapid reaction velocity, which was in the range of the experimental ɜtection limit.

able : Half-time of the HT-conversion (min) in dependence from different treatments of the soil.

amount of soil (kg)	deionized water	garden soil filtrate	garden soil suspension
0.5	26 28	37	–
1.0	52 58	53	10.5

4. DISCUSSION AND CONCLUSIONS

Under laboratory conditions a rapid conversion of HT to HTO from air moving over a soil has been observed. This is in accordance with former results /3/, /4/. The HT is lost into the whole soil column, remarkable loss of HTO back to the air humidity is observed.

Our results are in accordance with the literature /7/: The most intensive HT/HTO-conversion has been observed by warm, moist loam. The addition of formaldehyde vapour lowered, the addition of nutrient solution or microorganismic cultures enhanced the HT/HTO-reaction in the soil. These results also confirm the role of soil organisms as major site the HT/HTO-conversion. Our observation did not include plants. For beans preliminary results reported a conversion of HT to HTO, but not to organic material, too /8/. In the next future one should test the dependence of the HT/HTO-turnover in soil from environmental conditions (e.g. temperature), and should confirm the role of microorganisms in contrast to pure chemical reactions by stimulation and inactivation procedures.

5. LITERATURE

/1/ W. ROETHER: Natürliche Produktion von Tritium. In: F.E. STIEVE und G. KISTNER [Eds.] STH-Berichte 12/1980, Dietr. Reimer Verlag, Berlin (1982).
/2/ D.H. EHHALT: On the uptake of tritium by soil water and groundwater. Water Resources Res. 9 (1973), 1073-1074.
/3/ IAEA: Tritium in some typical ecosystems. IAEA Techn. Rep. Ser. 207, IAEA, Vienna (1981).
/4/ J.C. MC FARLANE et al.: Environmental tritium oxidation in surface soil. Environm. Sci. and Technol. 12 (1978), 590-593.
/5/ P.F. SAUERMANN et al.: Fifteen years in handling tritium problems in connection with low-energy particle accelerators. Behaviour of Tritium in the Environment, IAEA, Vienna (1978).
/6/ F. FÜHR: Die Aufnahme von Radionukliden aus dem Boden - Ermittlung praxisgerechter Transferfaktoren. Radioökologie, Dt. Atomforum, Vulkan-Verlag, Essen (1979).
/7/ C.E. MURPHY et al.: Ther conversion of gaseous molecular tritium to tritiated water in biological systems. Savannah River, Aiken S.C., DP - 1422
/8/ J.F. CLINE: Absorption and metabolism of tritium oxide and tritium gas by bean plants. Plant Physiol. 28 (1953), 717-723.

MODELISATION DU MOUVEMENT DU TRITIUM DANS LE SYSTEME SOL-PLANTE-ATMOSPHERE

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INTRODUCTION

Le transfert du tritium aux végétaux est généralement estimé en utilisant des modèles très simples qui reposent sur l'hypothèse d'un équilibre stationnaire entre la plante, le sol et l'atmosphère. Ces modèles ne peuvent être utilisés dans le cas d'une exposition unique, ou d'expositions intermittentes très espacées dans le temps, où l'équilibre stationnaire est loin d'être atteint. Dans un tel cas, l'évolution dans le temps de la concentration du tritium dans le sol et les plantes, ne peut être estimée qu'au moyen d'un modèle dynamique, décrivant le mouvement du tritium dans le système sol-plante-atmosphère.

DESCRIPTION DU SYSTEME

La figure 1 donne une représentation idéalisée du transfert du tritium entre le sol, la plante et l'atmosphère. L'eau tritiée apportée au système arrive par voie atmosphérique dans le cas de rejets gazeux ou par la voie du sol en cas d'irrigation avec de l'eau contaminée. On remarquera que le système est traversé par un flux d'eau ascendant qui correspond au flux de transpiration et qui est contrôlé par l'ouverture des orifices stomatiques à la surface des feuilles. Par ailleurs la surface du sol, aussi bien que le surface des feuilles, constituent des surfaces d'échange entre l'eau du sol ou des feuilles d'une part, et l'eau atmosphérique d'autre part. Le compartiment 1 contient l'eau du sol accessible aux racines ; le compartiment 2 contient l'eau circulante des racines et des tiges ; le compartiment 3 l'eau des feuilles ; le compartiment 5 est réservé à la matière organique totale de la plante et son contenu s'exprime en eau de combustion de la matière organique.

EQUATIONS DU MODELE

Pour modéliser les échanges d'eau tritiée entre l'atmosphère et le sol, nous avons découpé le profil du sol en n couches élémentaires d'épaisseur identique h (m) de contenu en eau identique m (kg m^{-2}). On suppose en première approximation que l'eau perdue par évapo-transpiration est réapprovisionnée par les précipitations et que le contenu en eau du sol reste sensiblement constant dans le temps et dans l'espace et est égal à la capacité au champs (field capacity). Les échanges entre les couches du sol, et les échanges à l'interface sol-atmosphère, se traduisent par un premier système de n équation différentielle linéaires :

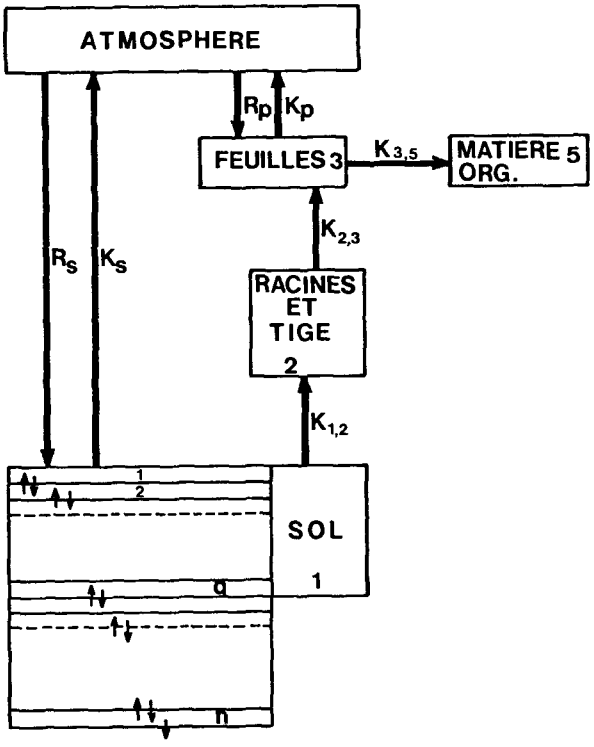

Fig 1 : Représentation schématique du système sol-plante atmosphère

$$\frac{dA_1}{dt} = R_s\, C_v - (K_s + K_d)\, A_1 + K_u\, A_2$$

$$\frac{dA_p}{dt} = K_d\, A_{p-1} - (K_u + K_d)\, A_p + K_u\, A_{p+1} \qquad (1)$$

$$\frac{dA_n}{dt} = K_d\, A_{n-1} - (K_u + K_d)\, A_n$$

où $A_1, \ldots A_p, \ldots A_n$ (pCi m^{-2}) sont les activités surfaciques des couches de sol précédemment définies ; C_v (t) (pCi kg^{-1}) est la fonction d'entrée du système représentant l'activité massique du tritium dans la vapeur d'eau atmosphérique ; R_s (kg m^{-2} s^{-1}) est le flux d'eau tritiée qui entre dans le système au niveau de l'interface sol-atmosphère ; K_s (s^{-1}) est la fraction du tritium de la couche superficielle du sol qui retourne à l'atmosphère par unité de temps. A l'intérieur du sol, les couches successives sont connectées entre elles par les coefficients de transfert K_u (s^{-1}) pour les transferts ascendants, et K_d (s^{-1}) pour les transferts descendants.

Les équations correspondant aux transferts dans la plante se traduisent par le deuxième système d'équations :

$$\frac{dN_2}{dt} = K_{1,2} \sum_{1}^{q} A_p - K_{2,3} N_2$$

$$\frac{dN_3}{dt} = R_p C_v + K_{2,3} N_2 - K_p N_3 \qquad (2)$$

$$\frac{dN_5}{dt} = K_{3,5} N_3$$

où N_2, N_3 et N_5 sont les quantités de tritium dans les compartiments 2, 3 et 5, exprimées par unité de surface de sol (pCi m^{-2}), sans préjuger de la répartition spatiale réelle à l'intérieur de chaque compartiment. R_p (kg m^{-2} s^{-1}) est le flux d'eau tritiée à l'interface feuilles-atmosphère. K_p (s^{-1}) est la fraction du tritium aqueux de l'eau des feuilles qui retourne à l'atmosphère par unité de temps. Les coefficients $K_{i,j}$ représentent la fraction de la radioactivité du compartiment i transférée au compartiment j par unité de temps.

La résolution des systèmes (1) et (2) permet d'obtenir les activités surfaciques N_i, connaissant les conditions à l'origine, la fonction d'entrée C_v (t) et les coefficients de transfert exprimés en fonction des paramètres caractéristiques du sol et des végétaux. Les concentrations moyennes du tritium C_i dans l'eau des divers compartiments sont ensuite obtenues par les relations $C_i = N_i/M_i$ où M_i est la masse surfacique de l'eau dans chaque compartiment.

EXPRESSION DES COEFFICIENTS DE TRANSFERT

Les flux de tritium dans le végétal, et au niveau des interfaces peuvent être exprimés en fonction des coefficients de transfert introduits dans les équations (1) et (2). Ces mêmes flux peuvent être aussi exprimés en fonction de paramètres caractéristiques du sol et des végétaux. En identifiant les deux expressions différentes de chacun des flux, on obtient les formules du tableau I.

Tableau 1 : Coefficients de transfert des équations (1) et (2) en fonction des paramètres caractéristiques du sol et des végétaux.

Sol	Végétaux
$R_s = \rho_a v + 0.2 \rho_w w$	$R_p = \rho_a G$
$K_s = \rho_\ell v / \rho_w \theta h$	$K_p = \rho_\ell G / M_3$
$K_u = D/\theta h^2$	$K_{1,2} = (\rho_\ell - \rho_a) G / M_1$
$K_d = D/\theta h^2 + w/\theta h$	$K_{2,3} = (\rho_\ell - \rho_a) G / M_2$
	$K_{3,5} = 0.2\ P/M_3$

Dans le tableau précédent les paramètres caractéristiques du sol et des végétaux sont : le coefficient de diffusion du tritium dans le sol (D m^2 s^{-1}) ; la conductance du couvert végétal (G m s^{-1}) ; l'épaisseur d'une couche élémentaire de sol (h m) ; le contenu en eau de chacun des compartiments (M_i kg m^{-2}) ; l'activité photosynthétique du couvert (P kg CO_2 m^{-2} s^{-1}) ; la vitesse d'échange à l'interface sol-atmosphère (v m s^{-1}) ;la vitesse d'infiltration de l'eau de pluie (w m s^{-1}); la concentration de la vapeur d'eau dans l'atmosphère (ρ_a kg m^{-3}); sa concentration dans une atmosphère saturée à la même température (ρ_ℓ kg m^{-3}) ; la masse volumique de l'eau liquide (ρ_w kg m^{-3}) et l'humidité volumétrique du sol (θ m^3 m^{-3}). On suppose en première approximation que la masse d'eau des végétaux et la conductance du couvert augmentent linéairement avec le temps. Les premiers calculs ont été effectués pour les valeurs des paramètres données dans le tableau 2.

Tableau 2 : Valeurs standards des paramètres caractéristiques du sol et des végétaux.

Symbole	Valeur		Unité
D	3×10^{-10}		$m^2 s^{-1}$
G	2×10^{-2}	(1)	m s^{-1}
M_1	90	(2)	kg m^{-2}
M_2	1	(2)	kg m^{-2}
M_3	5	(2)	kg m^{-2}
P	$4,3 \times 10^{-7}$	(3)	kg CO_2 $m^{-2} s^{-1}$
v	1×10^{-2}		m s^{-1}
w	$2,5 \times 10^{-7}$	(4)	m s^{-1}
ρ_a	$0,9 \times 10^{-2}$		kg m^{-3}
ρ_ℓ	$1,3 \times 10^{-2}$		kg m^{-3}
ρ_w	1×10^{3}		kg m^{-3}
θ	0,3		m^3 m^{-3}

(1) valeur nocturne : 2×10^{-3} ; (2) valeur atteinte après 60 jours de croissance ; (3) valeur nocturne : 0 ; (4) en période de pluie.

L'épaisseur h de chacune des couches élémentaires du sol, et le pas de temps Δt du calcul, ne peuvent pas être choisis arbitrairement. Pour que le calcul numérique soit stable nous avons pris h = 10^{-3} m et Δt = 0,1 h.

DOSISABSCHÄTZUNGEN FÜR DIE AUSLEGUNG VON KERNKRAFTWERKEN

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Einleitung

Während der Planungsphase eines Kernkraftwerkes (KKW) ist die Kenntnis über die Strahlenexposition des Personals bei zukünftigen Instandhaltungs- und Instandsetzungsarbeiten eine wichtige Grundlage für die Auslegung der Anlage. Die Anlage soll so konzipiert und ausgelegt werden, daß sowohl Kollektiv- als auch Individual-Dosen bei solchen Arbeiten so gering wie möglich gehalten werden. In einer BMI-Richtlinie /1/ werden diesbezügliche Anforderungen für die Planungsphase der Kernkraftwerke spezifiziert.

Um die Strahlenexposition für eine zukünftige Arbeit im Kontrollbereich abschätzen zu können, ist man auf Erfahrungswerte der im Betrieb stehenden Anlagen angewiesen. Unter Zugrundelegung der Detailplanung einer Instandhaltungsarbeit sollen die zu erwartenden Ortsdosisleistungen, Arbeitsaufwände und Häufigkeiten der Arbeiten festgelegt, und die sich daraus ergebenden Kollektivdosen abgeschätzt werden /1/.

In einer früheren Veröffentlichung /2/ wurden einige Näherungsverfahren für die Auswertung von Dosisleistungsmeßdaten zur Dosisabschätzung dargelegt. In diesem Vortrag werden die Dosisabschätzungen anhand eines Beispiels aus unseren umfangreichen Datenzusammenstellungen und Auswertungen aus mehreren KKW's, im Hinblick auf die Erfüllung der BMI-Richtlinie /1/ im einzelnen dargestellt und die Bedeutung solcher Dosisabschätzungen diskutiert.

Dosisabschätzungen für typische Arbeiten im Kontrollbereich

In der BMI-Richtlinie /1/ werden typische Instandhaltungs- und Instandsetzungsarbeiten zusammengestellt. Gemäß dieser Richtlinie ist für jede typische Arbeit eine Detailplanung unter Abschätzung der Kollektivdosis erforderlich. Tabelle 1 gibt ein Beispiel für eine zusammengefaßte Darstellung einer solchen Detailplanung in zugehörigen Tätigkeiten für die Demontage der primärseitigen Mannlöcher eines Dampferzeugers wieder. Die in Tabelle 1 angegebenen Ortsdosisleistungen sind Ergebnisse der Auswertungen gemäß /2/ und basieren auf Erfahrungswerten aus vergleichbaren 6 KKW. In Abbildung 1 werden beispielhaft die in Tabelle 1 verwendeten Ortsdosisleistungsmeßdaten für untere und obere Ebenen der Dampferzeuger dargestellt. Die in Tab.1 aufgeführten Arbeitsaufwände in Mannstunden sind Schätzwerte der Verfahrenstechniker. Die Kollektiv-Dosisabschätzung für jede Tätigkeit in Tabelle 1 wurde durch einfache Multiplikation von Ortsdosisleistungs- und Arbeitsaufwandswerten vorgenommen.

Die während der Planungsphase eines KKW durchgeführten Dosisabschätzungen sollen zur Beurteilung der zu treffenden Strahlenschutzvorsorge bei der Auslegung der Anlage dienen /1/. Auf solchen Dosisabschätzungen basierend kann man prüfen, ob gravierende Änderungen in der KKW-Planung und/oder bei der Durchführung der Instandhaltungsarbeit vorgenommen werden müssen, um eine weitere Dosisreduzierung zu erreichen.

Tabelle 1: Dosisabschätzungen für die Demontage der unteren und oberen primärseitigen Mannlochdeckel eines Dampferzeugers

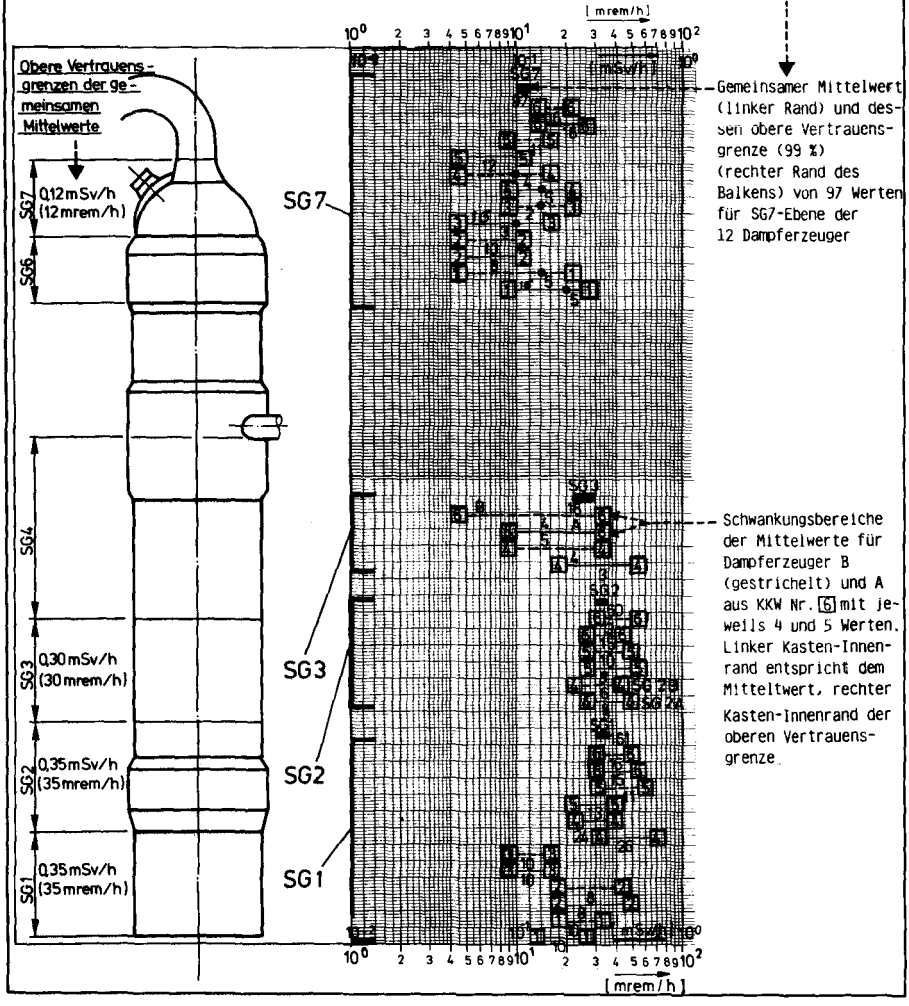
Arbeitsablaufplan (Detailplanung) in zugehörigen Tätigkeiten (1)	Arbeitsort/ und Arbeitsaufwand (Mannstunden) (M•h) (2)	Mittlere Ortsdosisleistung mit Ortscode / mSv/h / (3)	Kollektivdosis /man.mSv/ (4)	Dosisanteil /%/ (5)
1 Schraubenspannvorrichtung (SSV) mit Kran auf Hubwagen transportieren und vor Mannlochdeckel positionieren	-3,8 m Bühne (2•2)	0,35 (SG1)	1,40	15,1
	+16,3 m (2•2) (Deckel zu)	0,15 (SG7)	0,60	6,5
2 Bolzen verlängen Muttern lösen und SSV vom Hubwagen entfernen	- 3,8 m (2•2) (Deckel zu)	0,35(SG1)	1,40	15,1
	+16,3 m (2•2)	0,15(SG7)	0,60	6,5
3 Hubwagen an Mannlochdeckel befestigen, Muttern abschrauben und	- 3,8 m (2•1)	0,35(SG1)	0,70	7,6
Deckel auf Hubwagen ablegen	+16,3 m (2•1) (Deckel zu)	0,15(SG 7)	0,30	3,2
4 Mannlochdeckel mit Hubwagen wegfahren und innere austenitische Platte entfernen	- 3,8 m (2•0,5)	2,50(SGMW001)	2,50	27,0
	+16,3 m (2•0,5) (Deckel auf)	1,75(SGMW002)	1,75	18,9
Summe:			9,25	100 %

So zum Beispiel kann man zusätzliche Abschirmungen und/oder Trennung der Systemkomponenten mit hohen Dosisleistungen an den zukünftigen Arbeitsorten im Kontrollbereich vorsehen. Weiterhin können,soweit möglich, die dosisintensiven Arbeitsschritte an andere Orte mit geringerer Ortsdosisleistung verlegt werden. Wenn dies nicht möglich ist, können z.B. Einsatzmöglichkeiten von Fernbedienungseinrichtungen (z.B. mit TV-Kamera) untersucht werden.

Einzelne Tätigkeiten einer Instandhaltungsarbeit werden im allgemeinen an verschiedenen Orten im Kontrollbereich bei unterschiedlichen Ortsdosisleistungen und Arbeitsaufwänden durchgeführt (z.B.Tätigkeiten bzgl. Vorbereitung, Durchführung der Wirbelstromprüfungen der Dampferzeugerrohre). Es ist deshalb notwendig, die Dosen für die einzelnen Tätigkeiten abzuschätzen und daraus eine Gesamtdosis für die Instandhaltungsarbeit zu bilden (siehe z.B. Tab.1).

Der dadurch ermittelte Dosisanteil einer Tätigkeit an der Gesamtdosis ist dann ein Maßstab für die Notwendigkeit der Maßnahmen zur weiteren Dosisreduzierung, wenn diesbezüglich infrage kommende Maßnahmen anlagen- und verfahrenstechnisch möglich und sinnvoll sind. Wenn man die in Tab.1 angegebenen Dosisanteile mit den zugehörigen Ortsdosisleistungen und Arbeitsaufwänden zusammen betrachtet, sieht man eindeutig, welche von diesen Ein-

Abb. 1: Schwankungsbereiche der Langzeit-Mittelwerte von Ortsdosisleistungen für verschiedene Ebenen (z.B. SG7) der 12 Dampferzeuger (Geraderohrtyp) aus vergleichbaren 6 KKW, markiert mit [1] bis [6]



flußgrößen sinnvoll zu reduzieren sind (z.B. die in der Tab.1 eingekreisten Dosisanteile sind ungefähr gleich, wobei beim ersten ein größerer Arbeitsaufwand aber beim zweiten eine höhere Ortsdosisleistung für das Zustandekommen der Dosen verantwortlich sind).

Bedeutung der Dosisabschätzungen

Die während der Planung eines KKW's durchgeführte Dosisabschätzung für eine Instandhaltungsarbeit im Kontrollbereich legt zugrunde:

- einen Arbeitsablaufplan (oder Detailplanung) der zugehörigen Tätigkeiten
- eine Projektion der ausgewerteten Erfahrungswerte der Ortsdosisleistungen aus vergleichbaren KKW's /2/
- eine Abschätzung der Arbeitsaufwände für die geplanten Tätigkeiten

Die so abgeschätzten Dosiswerte können als "Erwartungswerte der Langzeitmittelwerte" betrachtet werden, wenn man bei der Projektion der Ortsdosisleistung und des Arbeitsaufwandes Langzeitmittelwerte verwendet. Der Zuverlässigkeitsgrad des abgeschätzten Dosiswertes ist von dem Zuverlässigkeitsgrad dieser Werte sowie von der Objektivität bzw. Realisierbarkeit der Instandhaltungs-Detailplanung abhängig.

Andererseits sind diese Erwartungswerte spezifisch für das geplante KKW und mit den Erfahrungswerten der im Betrieb stehenden Anlagen direkt nicht vergleichbar, da weder die Durchführungsmethode einer Instandhaltungsarbeit noch die zugehörigen Tätigkeiten und anlagenspezifischen Verhältnisse ein und desselben KKW im Laufe der Betriebsjahre immer identisch bleiben können. Dies gilt um so mehr im Vergleich verschiedener KKW untereinander. Außerdem stellen die veröffentlichten Dosiserfahrungswerte der KKW meistens Gesamtdosen einer Instandhaltungsarbeit mit den Dosen aus begleitenden sonstigen Tätigkeiten dar, deren Aufschlüsselung in die zugehörigen vergleichbaren Tätigkeiten nicht nachvollziehbar ist.

Da jede Projektion für die Zukunft mit Fehlern behaftet ist, darf man die absolute Höhe der abgeschätzten Gesamtdosis nicht überbewerten. Die abgeschätzten Dosiswerte für die einzelnen Tätigkeiten sowie deren Anteile an der Gesamtdosis können allerdings, wie oben dargestellt wurde, wertvolle Entscheidungskriterien für die Änderungen in der Planung eines KKW sowie der Instandhaltungsarbeiten zum Erreichen einer geringeren Strahlenexposition des Personals liefern.

Literaturstellen

/1/ BMI-Richtlinie über die während der Planung der Anlage zu treffende Vorsorge, GMBL 1978, Nr. 28

/2/ Atakan, Y., Evaluation of Dose Rate Data for Use in Nuclear Power Plant Design,
Nuclear Safety, 24(1), 66-74 (Jan.-Feb.1983)
oder
Atom + Strom, Jg. 29 (1983), Heft 4

TEN YEARS EXPERIENCE OF OCCUPATIONAL EXPOSURE AT SWEDISH LWRs

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1 INTRODUCTION

The Swedish light-water reactor (LWR) program contains twelve units, see Table 1, distributed on four coast-situated sites. Up to 1983, ten of them have been in production and during 1982 not less than 39 per cent of the electrical energy produced in Sweden originated from the LWRs.

To follow the occupational exposure, each power station performs its own dosimetry service. All the nuclear power stations use the same type of thermoluminescent dosimeters and registered doses are transferred and kept in one joint central dose register for the industry. Furthermore, all the stations are equipped with whole-body counters for internal contamination measurements. The experiense is, however, that the dose-equivalents from internal contamination have been insignificant compared with the dose-equivalents from external exposure.

TABLE 1 SWEDISH LWRs PROGRAM

Station unit and type	Electric power ($MW_{(e)}$)	First start-up
Barsebeck		
B1 BWR	570	1975
B2 BWR	570	1977
Forsmark		
F1 BWR	900	1980
F2 BWR	900	1980
F3 BWR	1060	1984
Oskarshamn		
O1 BWR	440	1971
O2 BWR	570	1974
O3 BWR	1060	1985
Ringhals		
R1 BWR	750	1974
R2 PWR	800	1974
R3 PWR	915	1980 a)
R4 PWR	915	1982 b)

Not in production before a)1981, b)1983

BWR-Boiling Water Reactor
PWR-Pressurized Water Reactor

2 COLLECTIVE DOSE-EQUIVALENT DISTRIBUTIONS

Figure 1 shows that since 1977, with 6 or more units in operation, the annual collective doses have been in the interval 8 to 13 manSv. The distributions of the annual collective doses have been such that the contractors' personnel on average received about 75 per cent of these doses.

The major portions of the radiation exposure have occurred, as was expected, during the annual routine outage periods and have then mostly amounted to from 70 to 80 per cent of the annual collective dose-equivalents.

On average, as can be seen in Table 2, the annual collective dose-equivalents per unit have been below 2 manSv.

In comparision with internationally accepted recommendations for the limitation of exposures to the individual, there are no recommendations or guidelines to restrict the occupational collective dose-equivalents for different practices. To establish a tentative guideline for such a restriction of the occupational exposure at the Swedish LWRs, the National Institute of Radiation Protection have suggested 2 mmanSv per installed MW electrical capacity and year as a level of ambition to which the

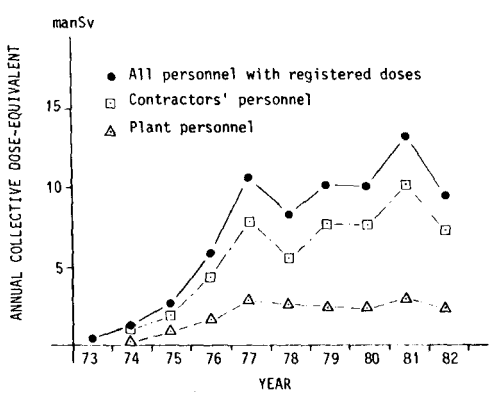

Figure 1. Annual collective dose-equivalents for the Swedish LWRs, total and divided between plant personnel and contractors' personnel, 1973-1982

collective dose-equivalent on average ought to be limited. As can be seen from Table 2, column 5, this value has been exceeded for some individual years, but not for the time period as a whole. The collective doses per unit energy generated, column 4, have been in the range 2-6 mmanSv per MW(e)·a, which is below reported values for most countries with LWRs. Among these countries, the United States have reported average values in the range 8-25 mmanSv per MW(e)·a.

TABLE 2 ANNUAL NORMALIZED COLLECTIVE DOSES AT SWEDISH LWRs

Year	Number of reactors	Average manSv per unit	[a] mmanSv per MW(e)·a	[b] mmanSv per MW(e) and year
1973	1	0.30	1.3	0.7
1974	4 [c]	0.35	6.3	0.5
1975	5	0.53	2.0	0.9
1976	5	1.18	3.4	1.9
1977	6	1.77	4.9	2.9
1978	6	1.37	3.2	2.2
1979	6	1.69	4.2	2.7
1980	7	1.43	3.5	2.2
1981	9	1.46	3.2	2.1
1982	9	1.06	2.3	1.5

Collective dose-equivalent per unit:
a) energy generated,
b) installed capacity and year
c) Units 02, R1 and R2 were commissioned during the last three months of 1974

3 INDIVIDUAL DOSE-EQUIVALENT DISTRIBUTIONS

In Figure 2, the individual annual dose-equivalent distributions for workers at Swedish LWRs are shown for the years 1981 and 1982. These graphs, which can be regarded as representative for the whole period of time considered, show that about 85 per cent of the workers received an annual individual dose of less than one tenth of the 50 mSv annual dose limit and that on average only about 3 per cent of the exposed workers each year received annual doses above 15 mSv. As can be seen from Figure 3, the number of contractors' personnel has been three times the number of plant personnel. However, the annual average doses

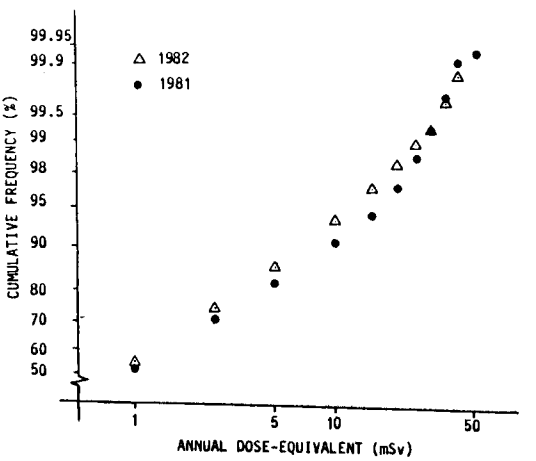

Figure 2. Log-probability plot of annual dose-equivalents to workers at Swedish LWRs for 1981 and 1982.

since 1975 have been in the range 2.3-3.3 mSv for both these groups.

Some attempts have also been made to divide up the collective dose-equivalents between occupational groups. The figures in Table 3 can, with the exception of those for the Insulation Personnel, be regarded as representative for the last six years.

The group Mechanical Repair Personnel, which have received about half the collective dose, contains smaller sub-groups such as steam generator workers and specialists performing control rod drive service and these sub-groups have received higher average doses than the total for the whole group.

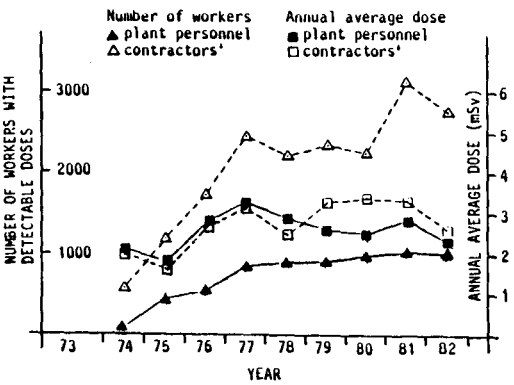

Figure 3. The number of workers with detectable doses and the annual average dose-equivalents for plant personnel and contractors' personnel at Swedish LWRs, 1974-1982.

TABLE 3 DOSE DISTRIBUTIONS FOR OCCUPATIONAL GROUPS AT SWEDISH LWRs 1982

Occupational group	Number of persons [1]	Proportion of total collective dose (%)	Average dose (mSv)	DRT [2]
Health physicists	131	8.3	6.0	1.9
Service personnel	578	11.7	1.9	0.9
Insulation personnel	138	6.4	4.4	0.4
Material testers	186	3.6	1.8	0.6
Operations personnel	483	6.5	1.3	0
Mechanical repair personnel	1323	53.1	3.8	1.9
Electrical and instrument technicians	413	6.1	1.4	0
Chemists	57	0.5	0.9	0
Others	457	3.8	0.8	0
Totals:	3766	100.0	2.5	1.3

1) Number of persons with detectable doses
2) DRT=Distribution Ratio Term, is the ratio of the fraction of the collective dose above 15 mSv for the observed group to the same fraction for the reference distribution defined in the 1982 UNSCEAR report, normalized to one for the reference distribution.

4 PREDICTION OF LIFETIME DOSE-EQUIVALENTS

An attempt to predict the lifetime dose-equivalents for workers who have received the highest individual doses at the LWRs during 1975-1982 is summarized in Table 4. The dose data are taken from the central dose register, which at the end of 1982 contained 10 037 workers with detected doses during that period. However, it should be noted that this prediction only includes those workers who received accumulated dose-equivalents of at least 50 mSv during the years 1975-1982 with the received dose distributed over 5 or more years during this period. This means that registered dose-equivalents to 158 individuals or about 1.5 per cent of all workers with registered doses have been considered. Excluded from this prediction were 47 individuals who have registered dose-equivalents exceeding 50 mSv for the period considered, but with the dose accumulation time less than 5 years. Of these 47 individuals four were Health Physicists, 30 Mechanical Workers, eight Service Personnel and five Insulation Personnel. To calculate the cumulative lifetime dose-equivalents the following mathematical extrapolation formula has been used:

$$H_{40} = \frac{40}{n} \sum_{i=i}^{n} \bar{H}_i$$

$\bar{H}_i$ = average annual dose-equivalent for each of the n years for which doses were registered

H_{40} = predicted dose-equivalent for a 40-year employment period

TABLE 4 PREDICTED LIFETIME DOSE-EQUIVALENTS FOR THE MOST EXPOSED WORKERS IN VARIOUS OCCUPATIONAL GROUPS

Occupational group	Number in the group	Dose-equivalents predicted over 40 years (Sv) Average H_{40}	Maximum H_{40}	Median H_{40}	75-procentile H_{40}
Health physicists	23 (171) [1]	0.58	1.00	0.53	0.70
Mechanical workers	105(4628)	0.48	1.02	0.45	0.55
Service personnel	5(1126)	0.63	0.80	0.68	-
Insulation personnel	8 (302)	0.55	1.24	0.46	-
Operation personnel	4 (787)	0.42	0.52	-	-
Material testers	3 (409)	0.42	0.51	-	-
Electrical and instrument technicians	5 (961)	0.32	0.35	0.33	-
Others	5(1653)	0.46	0.71	0.50	-

1) Number of workers with registered doses within the group.

As can be seen in Table 4, the predicted lifetime dose-equivalents on average range from 0.32 to 0.63 Sv and with a highest value of 1.24 Sv for the group Insulation Personnel. Of the groups Health Physicists and Mechanical Workers about 32 individuals, which corresponds to about 0.3 per cent of all workers with registered doses, have predicted lifetime dose-equivalents of 0.55 Sv or more.

The use of 40 years as an extrapolation time means that the lifetime dose-equivalents so predicted can probably be regarded as conservative values corresponding to the circumstanses at present prevailing at the Swedish LWRs.

ANALYSIS OF OCCUPATIONAL EXPOSURE IN A NATURAL-URANIUM HEAVY-WATER REACTOR

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INTRODUCTION

The Argentine Nuclear Programme contemplates the installation of six nuclear stations before the end of this century. Presently, two nuclear stations are in operation: Atucha I, 360 MW_e, and Embalse, 600 MW_e. A third station -Atucha II, 745 MW_e- is in its construction step. Additionally, uranium extraction, concentration and purification, and the manufacture of fuel elements are being performed within the country, while a heavy-water production plant and a semi-industrial reprocessing plant are in an advanced stage of construction.

Atucha I is a plant of the pressure-vessel reactor type, moderated and refrigerated with heavy water. Its operation started in 1974 and it has produced until mid 1983 2.5 $GW_e a$. This station was the first one of this type and capacity that was built in the world and this prototype condition has made the number of persons occupationally exposed higher than that in equivalent reactors. Accordingly, the use of cobalt alloys in various components of the reactor made dose rates relatively high along the primary circuit and, consequently, so were the doses resulting from the operations carried out in this sector. The prototype condition did also provoke a significant contribution of occupational exposures as a result of specific repairs about which no previous experience was available.(1)

An analysis is made herewith of the occupational doses at the Atucha I Nuclear Station, while the tasks with a greater contribution to the collective dose commitment were identified. Finally, an evaluation was made of the detriment expected during the whole useful life of the installation.

DOSIMETRIC EVALUATION

For the purposes of this paper, it was considered that the doses due to external irradiation, obtained as from readings of the personal dosimeters, are representative of the effective dose equivalent (further on doses). The doses due to internal contamination were calculated taking into account ICRP Publication 30.(2)

The United Nations Scientific Committee on the Effects of Atomic Radiations (UNSCEAR) (3) has provided a few approaches for the evaluation of the collective effective dose equivalent (further on collective dose) as from the distribution of the individual doses. For the case of Atucha I, the dose distributions due to external radiation deviate from the log-normal function at individual doses above $1.5\ 10^{-2}$ Sv. Opposedly, the dose distributions due to tritium

inhalation follow the log-normal function quite closely at any value.(4)

Dose values below 0.20 mSv for external radiation and far below such value for internal contamination have been recorded as zero. The assignment of zero dose to all those workers receiving doses below the registered level brings along errors in the evaluation of the collective dose. However, the error introduced for this reason is, in this case, below 1% of the total collective dose. Therefore, the collective dose introduced herewith was calculated as from the summation of the individual doses due to external radiation and to internal contamination.

Table 1 shows the collective dose incurred by all the personnel in the Station, while distinctions have been made among the doses incurred by personnel in three areas: plant operation, radiological protection and maintenance. Such dose includes that due to personnel training for future stations, which was estimated in approximately 20%. Also, distinction has been made between doses received by permanent and temporary personnel. 60% of the total collective dose is incurred by the maintenance staff.

TABLE 1

Collective dose incurred by personnel in Atucha I Nuclear Station (man Sv)

YEAR	TOTAL	OPER-ATION	RADIO-PROTEC.	MAINTE-NANCE	PERMA-NENT	TEMPO-RARY
1974	0.9	0.2	0.2	0.5	0.6	0.3
1975	1.6	0.5	0.2	0.9	1.3	0.3
1976	1.8	0.7	0.3	0.8	1.6	0.2
1977	6.8	1.5	0.9	4.4	4.9	1.9
1978	4.1	1.2	0.8	2.1	3.5	0.6
1979	5.9	1.3	1.0	3.6	4.7	1.2
1980	9.9	2.2	1.4	6.3	6.1	3.8
1981	5.8	2.0	1.0	2.8	4.9	0.9
1982	11.2	2.7	1.3	7.2	8.0	3.2

TASKS IMPLYING A HIGHER COLLECTIVE DOSE

The Atucha I station has showed a relevant performance and has operated with a load coefficient above 85% during long periods of time. However, maintenance tasks scheduled to be performed every 30 months, plus shutdowns involving interventions in the primary circuit, were the reasons for 40% of the collective dose incurred by the personnel during the last six years.

The tasks showing a greater contribution to the collective dose are shown in Table 2. Those tasks have been individualized as resulting from scheduled shutdowns, from non-scheduled repair tasks and from inspection operations.

TABLE 2

Maintenance tasks with greater contribution to the collective dose (man Sv)

YEAR	TOTAL	SCHEDULED TASKS		NON-SCHEDULED TASKS			
		INSPECT.	OTHER	QM	QF	QP	OTHER
1977	3.4	0.6	1.3	1.2	0.3		
1978	0.8			0.7	0.1		
1979	2.8			1.2	0.5	0.6	0.5
1980	5.2	1.6	3.1		0.4		0.1
1981	0.8				0.4		0.4
1982	6.4	2.0	4.4				

QM = Moderator circuit piping QF = Cooler circulation pump
QP = Moderator pump

ASSESSMENT OF THE RADIOLOGICAL DETRIMENT

The concept of radiological detriment was introduced by the International Commission on Radiological Protection (ICRP) and its evaluation is an essential step in the optimization of radiation protection against radiation sources. The linear dose/effect ratio allows for evaluating the health detriment produced by a given source by adding the doses incurred by each individual (collective dose) due to the operation of such source.

The detriment produced among occupationally-exposed workers at the Atucha I Station was mostly due to maintenance, repair and inspection tasks performed during the plant operation (Table 2). The expected detriment resulting from the operation of Atucha I during its whole useful life will depend on the frequency of the tasks performed in the vicinity of the primary circuit, on the evolution of the dose rates in the rooms where those tasks are carried out and on the way in which the maintenance tasks are performed.

The frequency of non-scheduled interventions kept increasing during the first years in operation and has stabilized during the last six years. Presently, maintenance tasks are scheduled for every 30 months and inspection operations are performed once a year.

The exposure rate in the vicinity of the primary circuit suffered a 100% increase during the last six years. This was mainly due to the deposition of cobalt activation products resulting from the corrosion of elements in the reactor. The tendency of the dose rates will be attenuated during the next few years by means of decontamination operations.

The experience collected during the operation of Atucha I has provided the necessary elements for planning maintenance operations towards decreasing the number of man-hours necessary for their accomplishment and, consequently, decreasing radiological detriment (5).

The occupational detriment that would result from the operation of Atucha I between now and the end of its useful life (the

latter estimated in 30 years) was evaluated as from the tendency observed in the annual collective dose during the last six years. The resulting collective dose would be 160 man Sv and the collective dose per unit of practice would be 27 man Sv/GW_ea. These values may be substantially reduced if special intervention systems are introduced, such as robots and special shields.

CONCLUSIONS

One of the factors that mostly contribute to a relatively high collective dose per unit of practice at Atucha I is that cobalt activation products, settled along the primary circuit, maintain dose rates at relatively high levels. This should be avoided in the future.

Maintenance tasks are the ones mostly contributing to the total collective dose and this is where most efforts must be made toward reducing the doses. Considering that the Argentine Authority has adopted a detriment cost value of US$ 10,000/man Sv, a significant investment has to be made in order to reduce detriment as far as it is reasonably achievable. It must also be considered that some of the radiation protection systems could be later used in both Atucha I and Atucha II stations. This would justify a considerably higher investment in radiation protection.

BIBLIOGRAPHY

(1) Gonzalez, A.J. IAEA-CN-42/158.

(2) ICRP Publication 30. Oxford, Pergamon Press, 1979. Annals of the ICRP v.2: N° 3/4, 1979.

(3) United Nations. Scientific Committee on the Effects of Atomic Radiation. 1982 Report to the General Assembly, with annexes. New York, United Nations, 1982. 773 p.

(4) Palacios, E.; Campos, J.M. and Curti, A. 1981 CNEA-NT 11/81.

(5) Gonzalez, A.J. et al. IAEA-SM-258/54.

EXPERIENCES OF CONTROLLING INTERNAL CONTAMINATION IN NUCLEAR POWER PLANT WORKERS

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INTRODUCTION

In Finland there are two nuclear power stations, one at Loviisa with two pressurized water reactors and one at Olkiluoto with two boiling water reactors. Both stations have been visited twice a year, once during the annual maintenance shutdown and once during normal operation, to determine internal contamination in nuclear power plant workers with the mobile whole-body counter of the Institute of Radiation Protection (1) and thus to find out which types of jobs cause internal radiation doses.

MATERIAL AND METHODS

The whole-body counting system used for these measurements was installed in a truck and the measuring geometry was that of a modified chair made of lead. Two alternative detectors, a NaI(Tl) detector (diameter 20.3 cm and height 10.2 cm) or a HPGe detector (efficiency 27 per cent), were used. (1). For the whole-body counting the workers were selected by the radiation protection officer at the nuclear power plant according to the instructions given by the Institute of Radiation Protection. These workers represented different types of jobs involving risks of internal radioactive contamination. Each time about fifty workers were measured. Since 1982, those selected for measurement during the annual maintenance shutdown, took a shower and changed into clean clothes before entering the truck.

The radiation doses were calculated assuming that the contamination nuclides were inhaled. The effective dose equivalent commitment of each nuclide was obtained from the annual dose limit by comparing the activity in the body at the time of measurement with the respective ALI value given in the ICRP30 (2).

RESULTS

During the annual maintenance shutdown detectable amounts of ^{51}Cr, ^{54}Mn, ^{58}Co, ^{60}Co, ^{110m}Ag and ^{124}Sb were observed in workers at the Loviisa plant. The same nuclides, except ^{110m}Ag and ^{124}Sb, which seem to be typical of Loviisa plant, were detected in workers at Olkiluoto plant. The highest contamination levels were found in workers participating in the mechanical maintenance, waste handling and decontamination. Figure 1 shows the mean activity found in employees participating in the mechanical work of opening and closing the reactor. In the control measurements made during normal operation, fewer persons were found to be contaminated and the amounts detected were smaller.

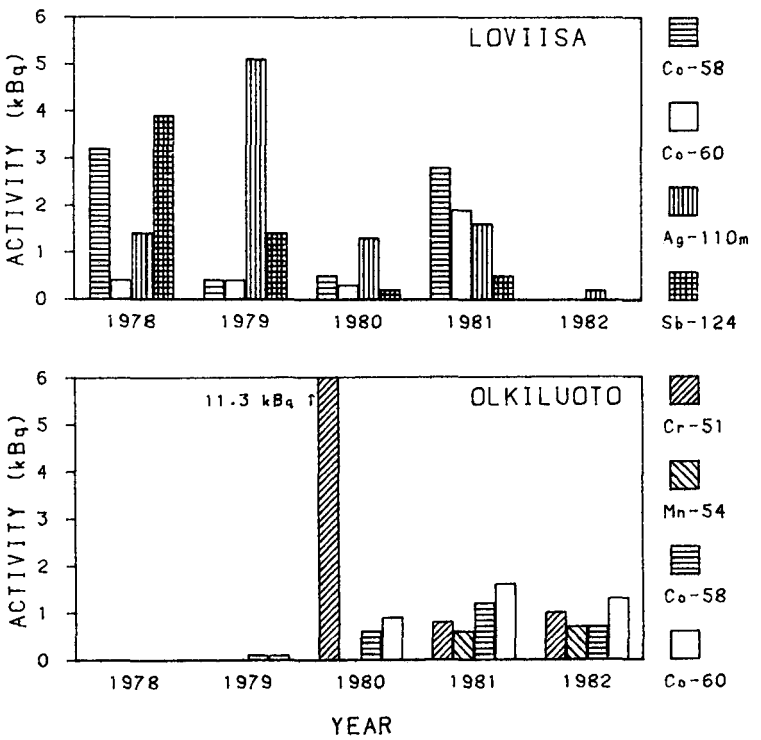

Figure 1. Mean acitivity of internal contamination nuclides detected in workers participating in the mechanical work of opening and closing the reactor during annual maintenance shutdown at Loviisa and Olkiluoto from 1978 to 1982 and at Olkiluoto from 1979 to 1982.

Table I. Mean effective dose equivalent commitments in μSv for different groups of workers at the Loviisa and Olkiluoto nuclear power plants during annual maintenance shutdown (S) and normal operation (O) from 1978 to 1982.

GROUP OF WORKERS		1978 μSv (n)[a]	1979 μSv (n)	1980 μSv (n)	1981 μSv (n)	1982 μSv (n)
LOVIISA:						
Mechanical maintenance	S	110 (36)	130 (29)	220 (16)	130 (30)	200 (5)
	O	10 (7)	110 (7)	40 (18)	74 (1)	6 (52)
Waste handling and decontamination	S	-[b] (0)	490 (5)	70 (2)	210 (3)	110 (1)
	O	-	-	15 (4)	85 (3)	19 (5)
Cleaning	S	34 (7)	150 (6)	160 (4)	110 (6)	-
	O	-	33 (1)	63 (9)	98 (2)	15 (6)
Laundry	S	6 (2)	63 (2)	92 (2)	24 (3)	-
	O	0[c] (1)	9 (2)	13 (3)	7 (2)	1 (3)
Radiation protection	S	34 (1)	85 (5)	31 (9)	24 (5)	16 (12)
	O	5 (8)	7 (8)	18 (5)	15 (9)	1 (4)
Laboratory	S	-	27 (2)	5 (1)	22 (4)	13 (11)
	O	2 (7)	3 (11)	15 (4)	9 (11)	-
OLKILUOTO:						
Mechanical maintenance	S	-	-	47 (18)	77 (22)	61 (28)
	O	-	0 (4)	24 (8)	5 (9)	13 (24)
Waste handling and decontamination	S	-	-	110 (4)	8 (5)	33 (4)
	O	-	0 (1)	30 (4)	9 (3)	48 (6)
Cleaning	S	-	-	0 (2)	-	310 (2)
	O	-	-	-	-	43 (2)
Laundry	S	-	-	5 (2)	28 (2)	36 (1)
	O	-	0 (1)	0 (2)	0 (2)	10 (1)
Radiation protection	S	-	-	15 (8)	39 (9)	37 (5)
	O	-	0 (7)	4 (9)	5 (9)	24 (7)
Laboratory	S	-	-	0 (2)	0 (1)	3 (2)
	O	-	0 (7)	0 (1)	0 (1)	0 (2)

[a] number of workers measured

[b] not determined

[c] < 1 μSv

During the 1982 maintenance shutdown at the Loviisa plant, a test with four workers was made on the influence of the shower and clean clothing on the measurement result. It was demonstrated that after shower only about ten per cent of the activity found in the first measurement was retained in these workers.

For both power plants, the mean effective dose equivalent commitments from 1978 to 1982 are presented in Table I. The highest individual effective dose equivalent commitment found at Loviisa was 1.2 mSv in a worker handling wastes and at Olkiluoto 0.50 mSv in a worker cleaning active areas.

DISCUSSION

Since the individual internal contamination variation is great, it is important that each of the groups to be measured consist of a sufficient number of workers. When the number is below ten it is best to count all of them. According to our results, special attention should be paid to persons working with mechanical maintenance, decontamination, waste handling and cleaning.

Since there were practical difficulties in arranging showers and changing facilities close to our mobile whole-body counter and since the amounts of internal contamination were small, the workers were measured in their own clothes until the end of the year 1982. At any rate, the internal doses are only a small fraction of the external doses, but they have to be determined for follow-up of the radiohygienic working conditions. Although the ALI values are secondary standards they were used in the dose calculations.

REFERENCES

1. Rahola, T. and Suomela, M.: A mobile whole-body counter for measuring internal contamination at nuclear power plants. 1982, Proceedings of the SRP, Symposium, Invernes, pp 305-310.
2. Annals of the ICRP, 1979, Vol. 2 No 3/4, 1980, Vol. 4 No 3/4, 1981, Vol.6 No 2/3, (Pergamon Press, Oxford, New York, Frankfurt).

OCCUPATIONAL AND ENVIRONMENTAL RADIATION EXPOSURES FROM MONAZITE PROCESSING INDUSTRY IN INDIA

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INTRODUCTION

The mineral deposits of west coast of India, in the states of Kerala and Tamil Nadu, are extensively exploited for recovery of monazite and other heavy minerals such as ilmenite, rutile, zircon, etc. The monazite is chemically processed to obtain thorium and rare-earths. In this paper, we present various aspects of radiation exposures in this realm of nuclear fuel cycle operations.

EXPOSURES DURING PHYSICAL SEPARATION OF MINERAL

External

The major monazite recovery plant is located at Manavalakurichi in Tamil Nadu. Studies carried out in the plant have identified 4 categories of workers subjected to radiation exposures by virtue of their working and residing in areas having different radiation background. The external exposures to each category is assessed and data provided in Table-1.

Table-1 Radiation Dose to Different Category of Workers

Exposure category		Exposure mSv/month			% of total	
Work in	Reside in	Occupational	Residential	Total	Occupational	Residential
1. High bkg.	High bkg.	2.05	1.40	3.45	59	41
2. High bkg.	Low bkg.	2.05	0.48	2.53	81	19
3. Low bkg.	High bkg.	0.58	1.40	1.98	29	71
4. Low bkg.	Low bkg.	0.58	0.48	1.06	55	45

S.D. is 20 %

It can be seen from the table that for group 2, contribution of occupational exposure to the total is high whereas for group 3, exposure received at their residence is high. For this study, background in residential areas is considered high when it exceeds more than 1 μGy/hr. For groups 1 and 4, the contributions from two types of exposure are nearly the same.

Persons who are engaged in the final stages of monazite separation (wind tables) are routinely monitored. The collective dose for this operation works out to be 0.3 man Sv/yr. This figure goes up by a factor of 2 when operations such as air table, bagging, bag-stitching and storage are also taken into account. The man mSv t^{-1} of monazite produced in the plant works out to about 0.2.

Internal

Inhalation hazards, especially due to resuspension of fine grained

Table-3: Ra-228 in water and sediments in the river

Location	Ra-228 activity, Bq/l or Bq/Kg			
	Monsoon		Non-monsoon	
	Water	Sediment	Water	Sediment
2-9 Km, upstream	0.03-0.06	48-119	0.11-0.32	41-59
	(0.03)	(52)	(0.07)	(22)
Outfall area	0.15	244	0.76	4315
	(0.04)	(41)	(0.04)	(159)
2-5 Km downstream	0.11-0.15	141-244	0.03-0.65	56-170
	(0.03)	(34)	(0.03)	(34)

S.D. is 40% (maximum)

The fertilizer waste sludge ($CaCO_3$) containing an activity of about 1 Bq/g of Ra-226 is disposed off on ground locally. The activity in the sludge is leachable and it has been estimated that about 90% Ra-226 in the environment is contributed by this source.

Biological uptake of Ra by the edible organisms in the river such as fish, prawns and oyster reveal concentration factors (CF) of 10 and 100 for flesh and calcified tissues respectively. The enhancement of activity was 10 times higher at the industrial zone of the river as compared to the background levels. The observed concentrations are about 10 Bq/kg in flesh and 100 Bq/kg in the bone of a popular species of fish, Etroplus. The levels are, however, not significant from the intake point of view.

On the basis of measurements made on the environmental samples and their intakes, the per capita dose from effluent discharges works out to be only about 0.5 μSv/yr for the population residing within 3 km of the plant. This is considerably lower than the natural background levels (1-2 mSv/yr) existing in these areas.

EVALUATION DES DOSES POUR DES EXPERIMENTS DE FUSION DE GRANDE ECHELLE AVEC MACHINES DU TYPE TOKAMAK

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Les appareils du type Tokamak employés jusq' à présent pour la recherche sur la fusion contrôlée n'ont pas entraîné des risques radiologiques très importants. Les problèmes les plus rémarcables ont étés en effet posés par les rayonnements X durs associés au ralentissement des électrons découplés produits dans la décharge, qui sont toutefois à régarder comme un risque exceptionnel.

Plusieures machines d' une nouvelle géneration sont à présent en cours de project, de contruction ou de première exploitation qui sont conçues pour approcher la limite du breakeven ou même, dans le cas du Jet et faisant les hypothèses les plus optimistes, pour atteindre l' ignition. Ces machines entraînent des risques radiologiques beaucoup plus rémarcables liés au grand nombre de neutrons produits et, par conséquent, à des niveaux d' activation induite assez élevés. De plus, quelques-unes des dites machines (Jet,Tftr) ont étées conçues pour fonctionner soit en deutérium (DD) soit avec des mélanges deutérium-tritium (DT), tandis qu' on va étudier la possibilité d' effectuer un nombre limité de décharges DT aussi pour des autres machines (FTU). Dans ces derniers cas on doit considérer aussi les problèmes liés à la nature radioactive du tritium.

Le tableau 1 montre les caractéristiques les plus importantes de quelques Tokamaks répresentatifs de cette géneration, y compris le "Frascati Tokamak Upgrade" (FTU), dont on va commencer la construction chez le Centre de Frascati[1].

Les niveaux d' irradiation externe par le rayonnement instantané (neutrons thermonucléaires et gammas associés) ont étés limités à des valeur plutôt basses à cause de la présence de quelques maisons à proximité de l' emplacement de la machine. Les épaisseurs adoptées pour l' enceinte de protection en béton sont de 220 cm pour les murs latérals et 150 cm pour le toit. On a estimé que le débit d' équivalent de dose à la distance de 60 m du tore ne dépasse pas quelques dixièmes de mSv/an en faisant des hypothèses sévères sur le rytme de fonctionnement (plasmas DT, 1000 chocs/an, 6.10^{17} neutrons/choc). Les débits de dose dans les milieaux de travail à proximité de la Salle FTU ne dépassent pas quelques mSv/an. Dans ces evaluations on a tenu compte des contributions soit des gammas de capture que des neutrons transmis par effet de ciel. On peut naturallement négliger tout risque d' irradiation externe au dehors de l' enceinte de protection dans le cas du fonctionnement DD.

Tableau 1 - Caractéristiques principales de quelques Tokamaks (R et a: rajons majeur et mineur; B: champ toroïdal; I: courant plasma; Y_{DD} et Y_{DT} nombre de neutrons par décharge en DD ou en DT).

	Organisation	R(m)	a(m)	B(T)	I(MA)	Y_{DD}	Y_{DT}
JET	Euratom Cee	2.95	1.25x2.1	3.5	4.8	$5\ 10^{17}$	$1\ 10^{20}$
TFTR	DOE USA	2.48	0.85	5.2	2.5	$1\ 10^{17}$	$2\ 10^{19}$
FTU	ENEA Italie	0.9	0.3	8.0	1.6	$6\ 10^{16}$	$8\ 10^{18}$ (*)
TS	CEA France	2.15	0.7	4.5	1.7	$2\ 10^{17}$	

(*) possibilité à l' étude

Le niveau de la radioactivité induite par les neutrons de fusion dans les structures de la machine ont étés calculés par un code simplifié qui utilise les fluxes de neutrons thermiques evalués par un model semi-empirique et les fluxes de neutrons rapides tirés d'un calcul monodimensionnel de transport [2]. On a calculé le débit d'équivalent de dose à l' intérieur de la chambre à vide, sur l' axe du plasma, et à l' extérieur du tore à 30 cm du cryostat.

La fig. 1 montre la décroissance du débit d' équivalent de dose en function du temp de rafroidissement pour deux temps de fonctionnement (1 an et 5 ans), soit dans le cas DD que dans celui DT. Pour le premier on a adopté un cycle annuel de fonctionnement qui prévoit 3000 chocs/an, divisés en deux séries chacune de 1500 chocs (30 chocs par jour, 5 jours par semaine pendant 10 semaines) séparées par un arrêt de 16 semaines. Pour le cas DT on a prévu 1000 chocs/an, également divisés en deux séries de 500 chocs (25 chocs/jour, 4 jours/semaine pendant 5 semaines) avec un arrêt de 21 semaines interposé. Il faut remarquer qu' on a considéré pour ces calculs une production moyenne de neutrons inférieure d' un facteur 10 aux valeurs maxima indiquées au tableau 1: le débit de dose à la suite d' une série de chocs de très bonne qualité pourrait donc être majeur d' un facteur 10. Les valeurs relatives à des temps de rafroidissement assez longs sont toutefois toujours correctes.

Le tableau 2 montre, pour les mêmes cas et les mêmes points d'observation, les débits de dose résiduels après un temp de rafroidissement d' un année. La contribution des principaux radionuclides à periode longue est aussi indiquée. On peut noter que, surtout dans le cas DT, le ^{60}Co est très important.

Les niveaux d' irradiation envisagés montrent que, dans le cas DD l' exploitation de la machine FTU peut causer un' irradiation significative du seul personnel chargé d'effectuer des intervention sur la machine immédiatement après des séries prolonguées de chocs de bonne qualité ou pendant des longues périodes. On estime que les doses reçues par ces personnes puissent être limitées au dessous de 8 mSv/an simplement au moyen d' una bonne organisation du travail. Une valeur maximum pour la dose collective peut être situé environ 0.3 homme.Sv/an.

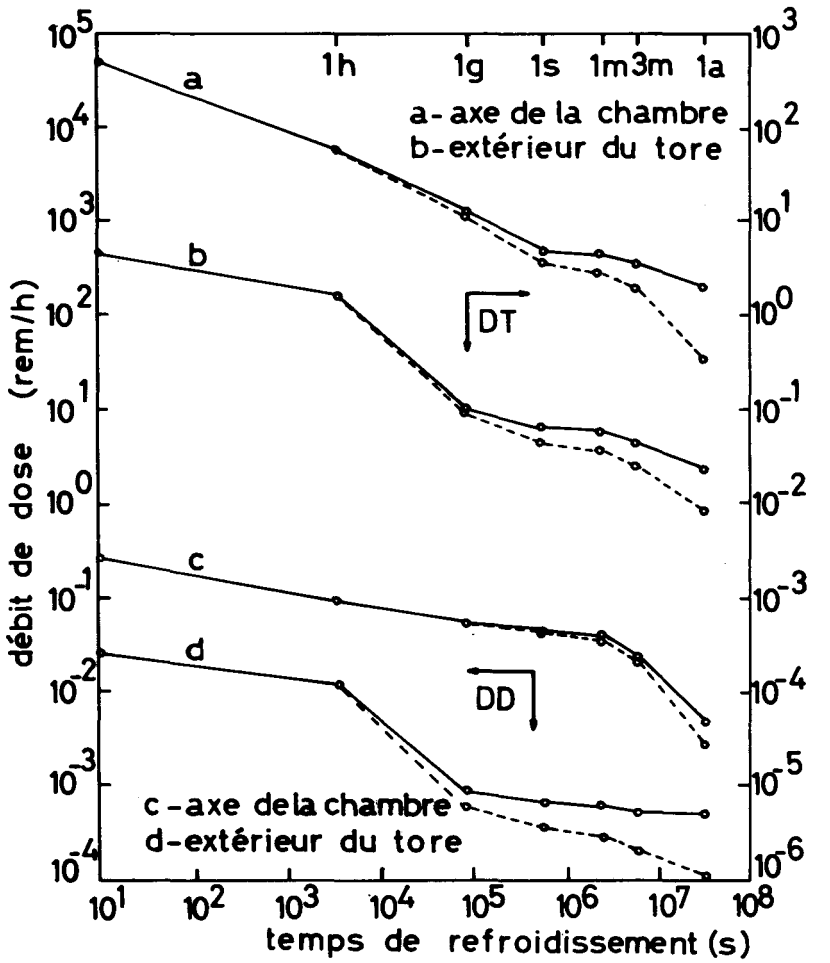

Fig. 1 - Débit d' équivalent de dose calculé à l' intérieur de la chambre à vide et à l' extérieu du tore pour deux temps de fonctionnement: 1 an (trait haché) et 5 ans (trait continu).

Tableau 2 - Débit de dose et contribution des radionuclides les plus importants (temp de rafroidissement: 1 an).

	Temp Fonct. (ans)	Débit de dose (mSv/h)	Contribution (%) ^{60}Co	^{54}Mn	^{58}Co
fonctionnement DD					
Axe de la chambre à vide	1	0.027	7	35	57
" " " " " "	5	0.048	19	37	43
Extérieur du tore	1	0.0011	84	11	5
" " "	5	0.0038	93	5	1.5
fonctionnement DT					
Axe de la chambre à vide	1	6.2	69	16	13
" " " " " "	5	20.	86	8	5
Extérieur du tore	1	0.081	51	35	13
" " "	5	0.23	74	21	5

Pour le fonctionnement DT, si l'on veut réspecter une limite pratique de 10 mSv/an, il faut envisager certainement l' installation d'appareillages pour la manipulation éloignée et il est aussi prévisible un' augmentation du nombre des travailleurs exposés. La dose collective pout tous ces travailleurs est moins que 1 homme.Sv/an.

Quant aux problèmes prôpres du tritium, on estime que les quantités maxima à considérer présentes en forme gaseuse sont: 400 Ci dans la Salle FTU et 6000 Ci dans le local de manipulation. Une étude préliminaire sur les conséquences radiologiques des rejets de tritium compte tenu des caractéristiques météorologiques du site de Frascati a été déjà effectuée[3]. En adoptant une hauteur de la cheminée de 12 m, on trouve des doses qui ne dépassent pas 0.8 uSv à 100 m de distance, pour des rejets continus de 10 Ci/an. Si l' on suppose un rejet accidentel de tous les 400 Ci de tritium présents dans la Salle FTU, on a une dose par inhalation et absorption au travers de la peau d' environ 0.13 mSv à 200 m de distance. Ces niveaux se réduisent 10 fois à 1.5 - 2 km. Il faut aussi rémarquer que les doses d' ingestion sont beaucoup plus faibles, sauf le cas d' une forte pluie pendant le rejet et de la consommation dirécte de l' eau de pluie. Même dans ce cas éxtrême on trouve des doses d' ingestion confrontables avec celles diréctes.

REFERENCES

(1) "Frascati Tokamak Upgrade", Associazione Euratom-ENEA sulla fusione, Rapports 82.49 et 82.59 (1982)

(2) E. Pedretti et C. Di Nicola, Communication privée (1983)

(3) P. Cagnetti et V. Ferrara, Communication privée (1982)

ABSORBED DOSES DUE TO MAMMOGRAPHY IN VARIOUS DUTCH HOSPITALS AND COMPARISON OF MEASUREMENTS PERFORMED WITH TLD AND IONISATION CHAMBER

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Because of the relatively high incidence of mammary cancer in The Netherlands (5000 new cases per 7 million women per year), it is presently being considered to implement screening programs for breast cancer. In this connection, it is of interest to determine the absolute dose and dose distributions in functional mammography installations. It must be considered, however, that in most of the hospitals in The Netherlands mammography is applied only upon medical indication. The mammography procedures in twelve hospitals have been compared with respect to the dose and the dose distribution in an acrylic plastic phantom (10.2. x 10.2 x 4.9 cm^3) simulating the breast. Dose determinations were made at 4.95, 24.5 and 44.05 mm depth with a Baldwin-Farmer ionisation chamber (BF-IC) connected to a Keithley 616 digital electrometer and with thermoluminescent dosimeters (TLD). The TLD were supplied and read out by the Radiological Service TNO, Arnhem, The Netherlands. The BF-IC was especially calibrated for this low energy range by the National Institute for Health Care, Bilthoven, The Netherlands (Hofmeester et al., 1983). The measurements were made under conditions similar to those in routine mammography using the automatic phototimers. The diagnostic image quality at the mammographic units was investigated by the use of a RMI Mammographic Random Phantom (Zuur et al., 1982). A score of "3.6" can be considered as a criterion for a "good" radiograph and a score of "3" as "reasonable". The radiographs of the test phantom were examined independently by four radiologists.

The doses received per mammoradiograph as derived from the ionisation chamber differed greatly among the different hospitals (Table I): between 2 and 21 mGy for the entrance dose, 0.1 and 0.3 mGy for the exit dose and 0.8 and 4 mGy for the mean tissue dose. The mean absorbed dose in the breast per investigation varies from 2 to 9 mGy. Information on the image quality is also given in Table I. The score of image quality should be expected to be positively correlated with the mean absorbed dose, but no correlation was found. This implies that a significant reduction could be achieved at several hospitals without loss of image quality.

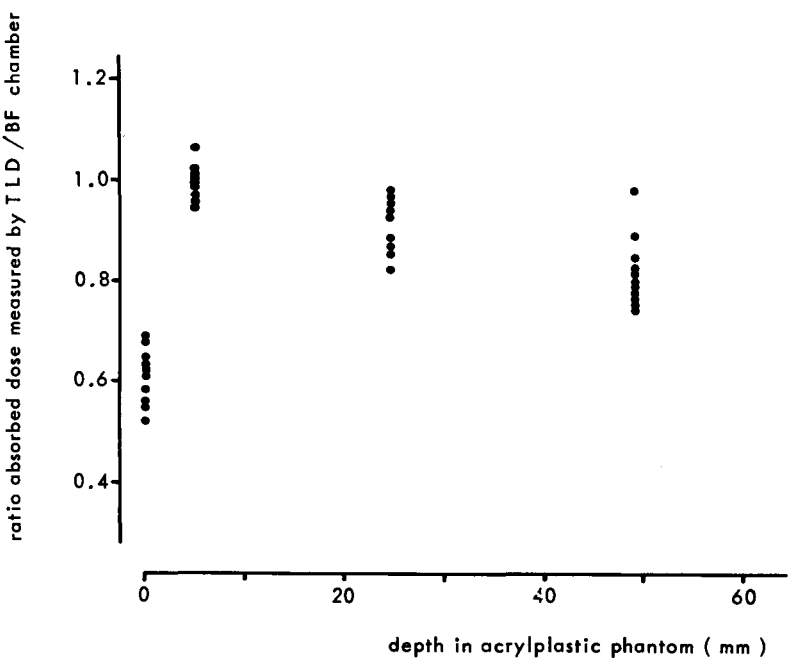
The variations in mean absorbed dose at the same mammographic installation due to the use of a grid and changes in the photographic material are given in Table II. The application of a grid causes an increase in the received dose by a factor of 2 to 3 unless the high voltage is increased. Employing a cassette instead of a vacuum envelope results in an about 30% higher mean absorbed dose. A 15% higher density in the radiographs results in a 25% higher absorbed dose. Measurements performed at the same hospital under different conditions are indicated separately (e.g., 2a and 2b).

TLD measurements were performed at about the same site and at the same time as the BF-IC. A comparison of the two dosimetry methods revealed that the doses derived from TLD were considerably lower than those from the BF-IC (see Figure 1). The mean ratios of the absorbed dose values (TLD/BF-IC) determined with the two dosimetric methods on the surface and at 4.95, 24.5 and 49 mm depth in the phantom were equal to about 0.6, 0.85, 0.80 and 0.75, respectively. The reasons

for these differences are still under investigation. Comparison of the two dosimetry methods at higher kvp values also showed a considerable difference. The discrepancy in the measurements on the surface between the TLD and the BF-IC is of special importance and should be explored in detail because the dosimetry in mammography and in other diagnostic examinations is generally performed with TLD on the skin.

The absorbed dose values determined for the Dutch mammography installations are higher than those obtained by other groups. This might be due to an underestimation of the dose as measured by TLD but also the use of different conversion factors from exposure in air (R) to the mean absorbed dose in the breast (rad). Most of the investigators in mammography dosimetry use a factor of 0.10 rad.R^{-1} (Hammerstein et al., 1979). This value, however, is valid under special circumstances. Our studies on the dose distribution in the phantom showed values of 0.16 to 0.23 rad.R^{-1} for this conversion. It has to be concluded that the factor derived by Hammerstein et al. (1979) should not be rashly used to calculate the mean dose in mammography from the exposure in air or from the entrance dose.

Hammerstein, G.R. et al., Radiology 130 (1979) 485.
Hofmeester, G.H. et al., abstract E2-18. In: Proc. of the Seventh International Congress of Radiation Research. July 3-8, 1983, Amsterdam, the Netherlands.
Zuur, C. et al., abstract 19.26 In: Proc. of the World Congress on Medical Physics and Biomedical Engineering, 13th Int. Conf. on Medical and Biological Engineering, Sept. 5-11, 1982, Hamburg, BRD.



Legend

Ratio of absorbed dose values determined with TLD and ionisation chamber at different positions in the phantom in the participating hospitals.

TABLE I

ABSORBED DOSES PER RADIOGRAPH, MEAN ABSORBED DOSES PER INVESTIGATION AND NUMBER OF WOMEN INVESTIGATED PER YEAR AT DIFFERENT MAMMOGRAPHY INSTALLATIONS

code number hospital	entrance dose (mGy)	mean dose (mGy)	score image quality	mean dose in one breast* (mGy)
1	8.9	1.9	2.8 ± 0.3	3.7
2a	5.9	1.4	2.8 + 0.2	2.8
2b	5.6	1.4	3.0 + 0.4	2.9
3	11.2	2.4	3.2 + 0.6	5.0
4	4.8	1.2	3.1 + 0.3	2.6
5	5.7	1.3	2.5 ± 0.2	2.9
6	21.1	4.2	3.0 ± 0.3	8.8
7a	19.9	4.4	3.8 ± 0.3	9.2
7b	8.6	1.9	2.7 ± 0.3	4.0
8	10.2	2.3	3.0 ± 0.4	5.4
9	8.0	1.8	3.1 ± 0.4	3.7
10	3.6	1.5		3.0
11a	3.4	0.8	3.4 ± 0.4	1.8
11b	4.0	0.9	3.3 ± 0.5	2.2
12	3.9	0.9	3.6 + 0.3	2.0
13	4.9	1.1	3.5 + 0.3	2.5
14a	4.7	1.0	3.6**	2.1
14b	3.5	0.8	3.6**	1.5
14c	2.8	0.6	3.3**	1.3

The dose values have an estimated uncertainty of ± 10%.

* Mean dose in one breast per investigation, including failures and the number of radiographs per investigation.

** Radiographs examined by two radiologists.

TABLE II

VARIATIONS IN RELATIVE MEAN ABSORBED DOSE VALUES UNDER DIFFERENT CONDITIONS FOR THE SAME MAMMOGRAPHY EQUIPMENT

Grid	
without grid	100
with a grid	233
with a grid + higher kVp (30 → 32 kVp)	172
with a grid + higher kVp (30 → 35 kVp)	100
Vacuum envelope/cassette	
vacuum envelope	100
cassette	138
Density	
normal density	100
ca. 15% more density (one step)	125
Film screen combination	
NMB-R	100
3M/alpha∠M	75
3M/-R	100

DOSE DISTRIBUTIONS FROM PANTOMOGRAPHIC AND FULL-MOUTH DENTAL RADIOGRAPHY

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A complete radiological survey of the jaw region is sometimes necessary. The conventional method is a full-mouth intraoral survey consisting of anything between 11 and 27 different films. The same survey using the pantomographic method with a single film is much less time consuming and more reproducible results are practically guaranteed.

The disadvantage of a pantomographic survey is that caries, periodontal, periapical and other types of lesions are not clearly indicated. This shortcoming can be solved by taking two additional bitewing radiographs and any other intraoral radiographs which may be necessary.

The purpose of this investigation was to determine the energy imparted to the jaw region during a pantomographic and other full-mouth examinations. Measuring the absorbed dose at points makes it difficult to compare results with other surveys because of different techniques and exposure values used. By measuring the dose and calculating the average energy imparted to the region enables one to make better comparisons. The survey giving maximum radiological information with the lowest energy imparted, is obviously the technique of choice.

Method:

The following procedures are investigated

1. Pantomographic survey.

2. Pantomographic survey plus two bitewing films at focus skin distances (FSD) of 20 cm and 40 cm.

3. Full-mouth survey consisting of 19 and 13 individual films at 20 cm and 40 cm FSD.

4. Full-mouth survey of 19 films at 10 cm FSD.

5. Combination of intraoral and extraoral techniques consisting of two angled lateral films of the jaw, two occlusal and two bitewing films (Stellenbosch method). For the lateral projection a 13 x 18 cm^2 film and a pair of screens was used.

The pantomographic surveys were done with a GE Panelipse, intraoral films with a GE 1000 using a 7 cm cylindrical cone and all 10 cm FSD films with an Asahi dental unit. The tube voltage of the GE units (2,7 mm Aℓ total filtration) is continuously variable from 50 to 100 kV whilst the tube current can be set at 10 or 15 mA. The Asahi (1,9 mm Aℓ total filtration) is a fixed energy unit of 58 kV and 7 mA.

Table 1. Exposure values.

Survey	FSD	kV	mA.s
Pantomographic		80	240
Bite wing	40	70	6
Bite wing	20	70	3
Stellenbosch	40	70	6
Stellenbosch	20	60	3
Full-mouth	40	68	6
Full-mouth	20	90	1,8

All exposure measurements were done using calibrated lithium fluoride powder as detector and an Alderson Rando tissue equivalent phantom ($\bar{Z}$ = 7,3 ρ = 0,985g/cm^3) simulating the patient. The 2,5 cm thick tissue equivalent slab running through the jaw region was replaced with a similarly contoured wax slab (ρ = 0,93 g/cm^3) in which 77 holes taking the LiF-powder capsules were drilled. Another 30 capsules were arranged on the skin. The dose to the eyes and thyroid was also measured.

The average absorbed dose of the 107 measuring points was determined for each procedure and the energy imparted calculated using the formula

1 joule = 1 gray.kilogram = 100 rad.kilogram

The mean energy and the absorbed dose to the eyes and the thyroid are given in Table 2 and is the average of the experiments.

Table 2. Mean energy and absorbed dose values.

Survey	FSD (cm)	Mean energy mJ	R. Eye mrad	L. Eye mrad	Thyroid mrad
Pantomographic	-	0,925	8 ± 1	8 ± 1,5	6 ± 1
Stellenbosch method	40	1,66	28 ± 3	137 ± 40	13 ± 1
Pan + 2 bite wing	40	2,14	13 ± 2	15 ± 3	12 ± 5
Pan + 2 bite wing	20	2,88	18 ± 5	17 ± 5	12 ± 7
Stellenbosch	20	2,45	46 ± 17	36 ± 4	31 ± 5
Full-mouth 13 films	40	3,57	38 ± 12	35 ± 7,5	20 ± 4,5
Full-mouth 19 films	40	3,76	441 ± 288	506 ± 175	46 ± 6
Full-mouth 13 films	20	5,98	288 ± 75	201 ± 123	54 ± 3
Full-mouth 19 films	20	6,87	577 ± 29	572 ± 25	92 ± 41
Full-mouth 19 films	10	10,3	166 ± 17	132 ± 13	79 ± 44

Discussion

Many publications appeared on dose values of dental radiological procedures. Comparisons are difficult for reasons due to different techniques and exposure parameters used. On the average, our values compare favourably with published data. (Wall, 1979)

Of all the techniques studied the normal pantomographic survey yields the least average energy and dose values. The more films taken and the shorter the FSD the higher the dose and energy imparted. The short FSD techniques tend to use relative low tube voltages which lead to increasing doses.

If more information is required a pantomographic survey with one or two bite wing films at high tube voltage and long FSD will be the method of choice. If a pantomographic unit is not available the Stellenbosch method can be used although the private dentist may have problems in developing the oblique lateral films.

Wall, B.F. et al, 1979, B.J.R., 52, 727-734.

L'EVALUATION DE L'EXPOSITION RADIOLOGIQUE ASSOCIEE AU RADIODEPISTAGE DE MASSE DE LA TUBERCULOSE EN FRANCE

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INTRODUCTION

L'effort particulier développé après la deuxième guerre mondiale pour systématiser le radiodépistage de masse a reposé fondamentalement sur l'hypothèse que les lésions tuberculeuses évoluant assez lentement, il devait donc être possible par des examens radiologiques de masse, répétés assez fréquemment, d'identifier les sujets porteurs de lésions avant que ces dernières aient atteint le stade de la contagiosité. Cependant, la situation ayant depuis lors considérablement évolué, tant sur le plan sanitaire que sur le plan des connaissances concernant l'évolution de la maladie, l'utilité du maintient du radiodépistage systématique semble désormais pouvoir être remise en cause. En effet, de nombreux arguments épidémiologiques et économiques, auxquels peut venir s'ajouter celui du risque radiologique, ont progressivement conduit les responsables des programmes de santé publique à préconiser un abandon du radiodépistage de masse pour le remplacer par un dépistage sélectif pour les groupes dit à "haut risque" /1/.

L'objectif de cette communication est de présenter les résultats d'une étude récente /2/ sur la situation du radiodépistage de masse de la tuberculose en France du point de vue de l'exposition radiologique de la population (*).

L'ACTIVITE DU RADIODEPISTAGE EN 1980.

Malgré l'évolution réglementaire récente, limitation de la radioscopie (période 1960-1970), puis suppression progressive de l'obligation de radiodépistage pour certaines catégories de la population (période 1970-1980), ce sont encore près de 10 millions de personnes qui sont passées en 1980 par le radiodépistage pulmonaire (voir tableau 1). Le taux de couverture de la population est donc d'environ 19 %, mais il s'agit en fait d'un tiers de la population entre 18 et 60 ans.

	Activité en millions d'actes			
	Graphies	Radiophotos	Scopies	Total
Médecine du travail	0.5	2.7	2.2	5.4 (55,0 %)
Dispensaires santé publique	0.2	1.8	0.2	2.2 (22,5 %)
Autres	0.3	1.7	0.2	2.2 (22,5 %)
Total	1.0 (10 %)	6.2 (64 %)	2.6 (26 %)	9.8 (100 %)

Tableau 1 : Répartition des actes de radiodépistage en 1980.

(*) Cette étude a été effectuée dans le cadre du Contrat d'Association C.E.A/EURATOM - SC 002-BIAF-423 F.

Ces dix millions d'actes cachent une réalité assez complexe ou de nombreux organismes sont à la fois prescripteurs et prestataires : chacun travaillant à la fois pour son compte et pour celui des autres. Parmi les 15 institutions impliquées dans le radiodépistage, la médecine du travail et les dispensaires de santé publique qui dépendent du Ministère de la Santé couvrent environ 75 % de l'activité totale. En ce qui concerne la répartition des actes selon le type d'examen, le fait marquant est la rémanence en 1980 d'un taux encore important de radioscopies (près de 25 %) sur le total des actes de radiodépistage.

L'EVALUATION DU RISQUE RADIOLOGIQUE

Du point de vue méthodologique, si une revue de la littérature récente sur la question de l'évaluation des risques associés aux irradiations médicales peut donner l'impression que le sujet est controversé, il apparaît cependant à travers les divers auteurs un certain consensus quant aux procédures d'évaluation, la controverse portant plus sur le choix des indicateurs de risque que sur la méthode d'évaluation proprement dite (voir /3, 4, 5, 6/ par exemple). La méthode finalement retenue pour l'évaluation de l'exposition est celle préconisée par la C.I.P.R. /7/ qui repose sur le concept d'équivalent de dose effectif et qui permet d'évaluer un risque global pour un individu exposé que le corps entier soit ou non irradié de manière uniforme, ce qui est le cas pour les examens radiologiques.

De manière formalisée l'équivalent de dose effectif H est donné par l'expression :

$$H = \sum_{t} W(t)\, H(t)$$

où ; H (t) est l'équivalent de dose à l'organe (t) et W (t) est le facteur de pondération qui traduit la part du risque associé au tissu ou à l'organe irradié (t) par rapport au risque total lorsque le corps entier est irradié de façon uniforme /7/. Connaissant les équivalents de dose aux organes pour chaque type d'examen (i) (radiographie, radiophotographie et radioscopie), il est possible de définir un équivalent de dose effectif moyen par cliché donné par la formule :

$$H_i = \sum_{t} W(t)\, H_i(t).$$

Le passage à la dose collective annuelle s'effectue en prenant en compte l'activité radiologique globale réalisée chaque année, ventilée selon les trois catégories d'actes considérées : le nombre de clichées pulmonaires annuels relatifs à la radiographie et à la radiophotographie et le nombre "d'examens" de radioscopie. La dose collective S associée au dépistage peut donc être formalisée ainsi :

$$S = \sum_{i} H_i\, n_i$$

n_i étant le nombre de clichés annuels pour un examen de type i.

Les équivalents de dose aux organes

Afin de déterminer les équivalents de dose aux organes H (t) une série de mesures dosimétriques sur un fantôme anthropomorphe Alderson Rando ont été réalisées pour les trois techniques d'examen pratiquées en radiodépistage. Ces mesures ont été effectuées sur des installations radiogènes actuellement en fonctionnement dans des centres de radiodépistage et jugées représentatives du parc.

L'utilisation du Rando a permis pour chaque type d'examen d'évaluer la distribution de la dose au niveau des différentes tranches considérées, chacune d'entre elles pouvant représenter soit entièrement un organe donné (cas de la thyroïde) ou une sous-partie (cas du poumon). Les dosimètres employés ont été des dosimètres thermoluminescents en borate de lithium activés au manganèse conditionné sous forme de pastilles très maniables et d'utilisation facile. Ont été ainsi évaluées les doses à la thyroïde, aux gonades (ovaires et testicules), aux poumons et aux seins, pour les trois techniques d'irradiation. En ce qui concerne l'évaluation de l'équivalent de dose à la moëlle osseuse, une mesure directe pour être correcte, aurait supposé la mise en place dans le fantôme, d'une à plusieurs centaines de dosimètres /12, 13, 14/. Compte tenu des moyens limités dont nous disposions pour la lecture des dosimètres, il a été décidé de recourir à une analyse de la littérature pour compléter nos propres mesures /8, 9, 10, 11/.

Le tableau 2 présente les résultats pour les trois types d'examens. Pour la radioscopie les valeurs portées sur ce tableau correspondent à un temps d'exposition moyen de 20 secondes confirmé par plusieurs praticiens. La contribution de la dose gonade a été évaluée en moyennant les doses ovaires et testicules.

Type d'examen	Equivalents de dose moyens aux organes par cliché (mSv)				
	Poumon	Thyroïde	Gonades	Sein	Moëlle[1]
Radiographie	0,23	0.09	0.008	0.11	0.07
Radiophotographie	0.66	0.21	0.002	0.15	0.40
Radioscopie	4.77	0.60	0.02	0.45	2.00

1 : Valeurs moyennes tirées de la littérature

Tableau 2 : Résultats dosimétriques

Les équivalents de dose effectifs collectifs

Le tableau 3 présente les équivalents de dose effectif individuels moyens par acte, les équivalents de dose effectifs collectifs annuels associé à chaque type d'examen. La contribution de l'équivalent de dose au poumon est prépondérante et représente environ 50 % de la dose totale. Les radioscopies constituent de loin l'examen le plus irradiant : six fois plus que la radiophotographie et quinze fois plus que la radiographie. Du point de vue de l'exposition collective il est intéressant de noter que les actes de radioscopies qui ne représentent que 25 % de l'activité totale du radiodépistage contribuent pour environ 70 % de la dose collective totale.

Type d'examen	Equivalent de dose effectif individuel moyen (mSv/acte)	Equivalent de dose effectif collectif (HommeSv/an)
Radiographie	0,06	60
Radiophotographie	0,16	990
Radioscopie	0,90	2 340

Tableau 3 : Exposition individuelle et collective

CONCLUSION

Bien qu'en déclin (l'activité a presque diminuée de moitié entre 1975 et 1980), le radiodépistage systématique touche encore un tiers de la population active française. L'exposition collective associée étant loin d'être négligeable puisqu'elle conduit à une dose "per capit" annuelle de l'ordre de 0,65 mSv, et compte tenu d'une efficacité devenue médiocre (2 000 à 4 000 nouveaux cas dépistés grâce aux 10 millions d'actes effectués en 1980, soit environ 25 % du total des nouveaux cas), il apparaît souhaitable de réexaminer si le maintien en sa forme de cette activité reste justifié.

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REFERENCES

/ 1/ TOMAN K., "Lutte anti-tuberculeuse : la radiographie de masse", Chronique OMS 30, p. 51-57 (1976).

/ 2/ C. LEFAURE, J. LOCHARD. Contribution à l'étude de l'irradiation médicale : le cas du radiodépistage systématique en France. Rapport C.E.P.N. n° 58 (1982).

/ 3/ COHEN L.B., "Proposals on use of the BEIR III Report in Environmental Assessment", Health Physics.

/ 4/ WALL B.F., SRIMPTON P.C., "Deliberations on a suitable quantity and technique for assessing the somatic risk for radiodiagnostic radiology", in Patient Exposure to Radiation in Medical X-ray diagnostics, Commission of the European Communities EUR 7438 EN, Luxemburg (1981), p. 413-425.

/ 5/ LAWS P.S., ROSENSTEIN M., "A somatic dose index for diagnosis radiology", Health Phys. 35 (1978), p. 629-642.

/ 6/ PERSSON B., "Review of various weighted dose concepts used for estimation of risks from medical X-ray diagnosis", in Patient Exposure to Radiation in Medical X-ray Diagnosis, Commission of the European Communities EUR 7438 EN, Luxemburg (1981), p. 355-373.

/ 7/ ICRP, Publication n° 26 - Recommandations of the International Commission on Radiological Protection, Oxford, Pergamen Press, (1978).

/ 8/ GREGG E.C., "Risk-benefit considerations in chest radiography", Public Health Service, F.D.A., Bureau of Radiological Health (BRH), Optimization of chest radiotherapy Madison April 30 - May 2 1979 (HHS Publication (FDA) 80-8124) Rockville BRH 1980, 166-171.

/ 9/ HASHIZUME T., KATO Y., KUMAMOTO Y, et al., "Population mean marrow dose and leukemia significant dose from diagnostic medical X-ray examinations in Japon, 1969. Health Physics 23, p. 845-853, 1972.

/10/ PORETTI G., GARAVAGLIA G., IONESCO F., MINI R., OTT P., FENZ J., "The determination of bone marrow dose resulting from diagnostic X-ray examinations", in Patient exposure to radiation in medical X-ray diagnosis, Possibilities for dose reduction. Proceedings of a meeting held in Munich-Neuherberg 27-30 april 1981, Rapport EUR-7438 EN, Commission of the European Communities 1981.

/11/ SHLEIEN B., TUCKER T., JOHNSON D.W., "The mean active bone marrow dose to the adult population of the United States from diagnostic radiology, FDA 77-8013, 1977.

THE RADIOLOGICAL IMPACT ON THE RHINE-MEUSE REGION FROM NORMAL OPERATION OF NUCLEAR FACILITIES

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The aim of this study /1/ was to assess the future radiological impact on the population in the Rhine-Meuse Region (RMR) attributable to radioactive discharges from nuclear facilities under normal operating conditions. This has been done by calculating dose rates to the period around the year 2000 on the basis of currently available plans for facilities and sites and the forecast development of nuclear technology in the coming decades.

Around the year 2000, on the basis of the currently available plans for facilities and sites, nuclear electricity generating capacity in the RMR will amount to about 55 GWe. The locations of the various sites can be seen from Fig. 1

The calculations were based on the release rates of 17 fission and activation products regarded responsible for the greater part of the total radiological exposure. For operation facilities the measured discharge rates have been adopted, while for facilities yet to be commissioned, and this is the majority of the 55 GWe, realistic anticipated values were used.

The dose rates are calculated with regard to the most important exposure pathways; namely,

for gaseous discharge
- external exposure from activity in the stack plume and from activity deposited on the ground
- internal exposure from activity inhaled and from activity ingested with food

for liquid discharge
- external exposure from activity in the water body and from activity deposited on the ground of inundation areas
- internal exposure from activity ingested with drinking water and food.

These calculations were backed by data on the environmental conditions of the

region (meteorology and hydrology) and on the civilizing structure of the region (population, agriculture, drinking water supplies, and fresh water fisheries).

The investigations show that the maximum individual dose rates for the period around the year 2000, calculated on the basis of the assumptions made, are of the order of magnitude of 1 mrem/a for most organs. For the thyroid, dose rates of up to about 27 mrem/a may occur via the gaseous effluent pathway. The average individual dose rates to the individual organs are in the region of 0.01 to 0.1 mrem/a. The average whole body dose rate - derived from the dose rates for the individual organs weighted on the base of relative organ weight - works out at 0.035 mrem/a. The average effective dose rate - derived from the dose rates for the individual organs weighted on the basis of the respective organ-dependent risk-coefficients - using the ICRP-recommended weighting factors - works out at 0.034 mrem/a (Table I).

Analysis of the contributions of the various nuclides and exposure pathways to the doses thus calculated shows in the case of the gaseous effluent exposure pathways that the ingestion pathway is the main contributor to the overall dose to every organ apart from the skin. For the whole body, the contribution is some 88%, with the C-14 isotope accounting for 66%, H-3 for 14% and Sr-90 for 6%. In the case of the skin, the Kr-85 isotope makes a particularly significant contribution to the overall dose, as do I-129 and I-131 in the case of the thyroid and the isotope Sr-90 in the case of bone.

As far as impact via the liquid effluent exposure pathways is concerned, the caesium isotopes Cs-134 and Cs-137, I-131 and Sr-90 are the main contributors. In the case of external exposure due to activity deposited on the ground, the contributions of Co-58 and Co-60 also assume relatively great significance.

A comparison of dose rates attributable to other sources in terms of whole-body exposure shows that the radiation exposure due to nuclear installations in the Rhine-Meuse Region in 2000 contributes approx. 0.02% to the total whole-body dose. In the case of persons exposed mathematically to maximum dose, the portion is approx. 0.5% (Table II)

/1/ A. Bayer
The Radiological Exposure of the Population in the Rhine-Meuse Region
Report of the Comm. of the European Communities V/2475/81EN (1982)

Table I: Maximum and Average Individual Dose Rates Within the Rhine-Meuse Region

	Dose Rates (mrem/a)	
	Maximum	Average
Gaseous Releases:		
Whole Body	1.2	0.031
Bones	0.94 (Child)	0.026
Skin	1.6 (Child)	0.082
Thyroid	27 (Infant)	0.13
Liquid Releases:		
(Drinking Water Only)	0.03	
Whole Body	0.03 (Child)	0.001
Bones	0.13 (Child)	0.004
Thyroid	0.06 (Infant)	0.001

Table II: Comparison of Whole-Body Exposures

Type of Exposure	Dose Rate (mrem/a)	(%)
Exposure to natural radiation in the Rhine-Meuse Region	75 - 260	approx. 63
Exposure to radiation (average value) attributable to medical diagnosis and treatment (1975)	50 - 80	approx. 34
Exposure to radiation attributable to nuclear weapons test (1975)	1 - 8	approx. 3
Exposure to radiation (whole body) attributable to nuclear installations in the Rhine-Meuse Region (2000)	0.035 (average) 1.2 (maximum)	0.02 0.1
Exposure to radiation (whole body) attributable to global nuclear technology (2000)	0.01	0.01
Total	approx. 175	100

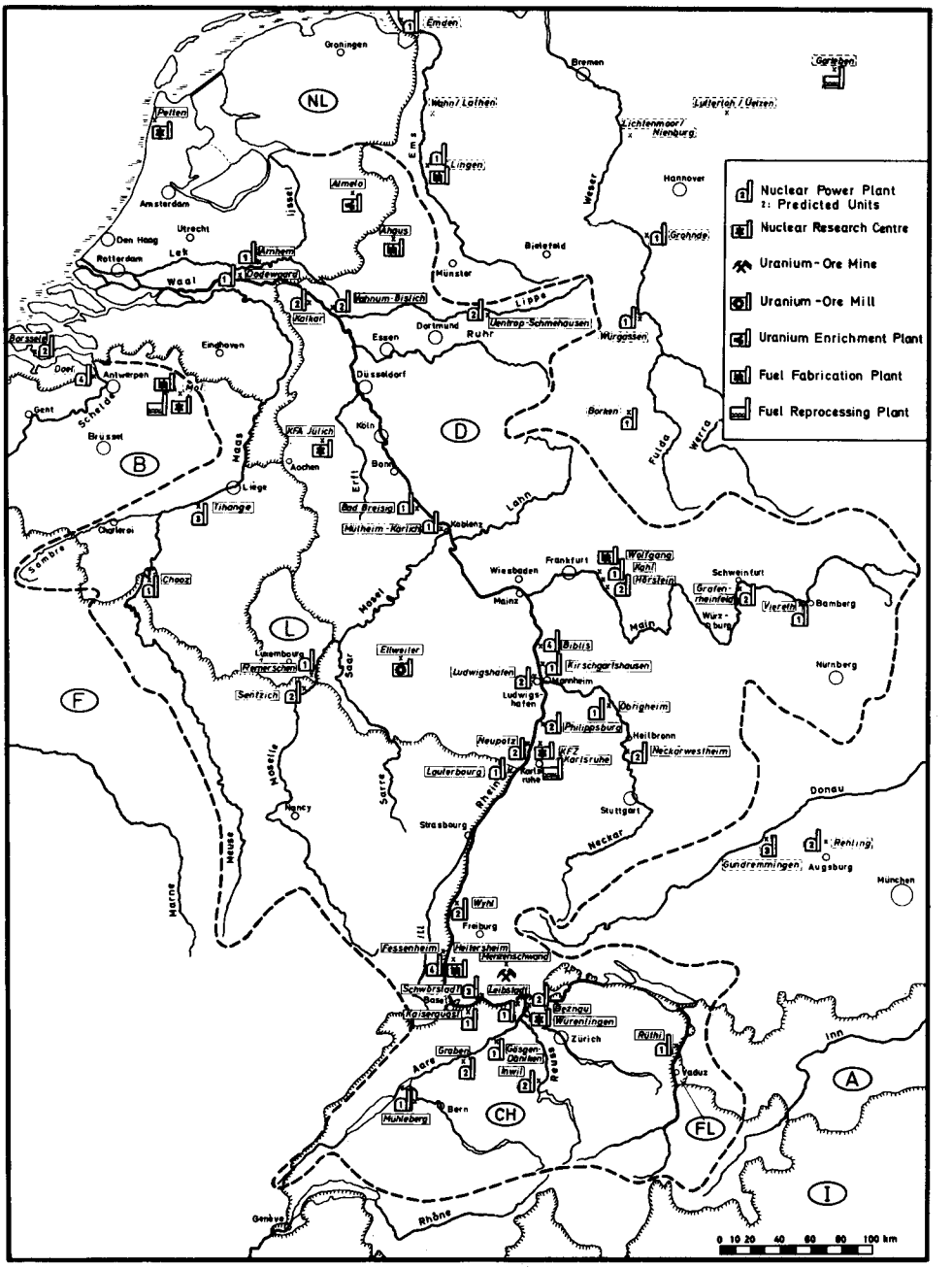

Fig. 1: Sites of Nuclear Facilities within the Rhine-Meuse Region

METHODE D'EVALUATION DES CONSEQUENCES SANITAIRES DES REJETS D'EFFLUENTS LIQUIDES DE L'USINE DE RETRAITEMENT DE LA HAGUE

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Dans le cadre du cycle du combustible nucléaire, l'usine de retraitement a pour mission de traiter les éléments combustibles irradiés au sein des centrales électrogènes afin d'en récupérer l'uranium et le plutonium qui seront réutilisés dans les centrales nucléaires classiques et celles dites surgénératrices. Ce type d'installation permet également d'isoler et de concentrer, pour les stocker, les autres radionucléides.

En ce qui concerne les effluents liquides, qui sont les seuls à nous intéresser dans le cadre du présent travail, ils seront réduits à un volume minimal par un recyclage des réactifs minéraux et organiques mis en jeux dans les différentes étapes du traitement.

Les rejets en milieu aquatique ne concernent que les effluents radioactifs de faibles activités dont seules les fractions surnageantes clarifiées issues des opérations de décantation et de filtration seront autorisées à être libérées dans l'environnement, après mesures des niveaux de radioactivité des différents radionucléides. Les principaux produits de fission concernés par les rejets et qui peuvent contribuer de façon significative à la dose collective sont le ^{137}Cs, ^{134}Cs, ^{106}Ru, ^{90}Sr auxquels il faut adjoindre ^{144}Ce, ^{95}Zr, ^{95}Nb, ^{125}Sb, ^{60}Co et les actinides comme le ^{239}Pu, ^{238}Pu et ^{240}Pu [1].

Les doses admissibles totales fixées par la CIPR sont le point de départ de l'évaluation des taux maximum admissibles de rejets de radionucléides en un lieu donné. Ces quantités sont établies en fonction de l'état physique et chimique des déchets au moment de leur émission, des processus d'advection et de diffusion turbulente au lieu et instant d'évacuation, des échanges entre les matières en solution ou en suspension dans l'eau de mer et les sédiments, des facteurs et des voies de transfert du milieu marin à l'homme : importance des ressources alimentaires maritimes et consommation locale de ces produits, temps de contact possible avec le sable des plages ou le sédiment des lieux de pêche.

Avant tout rejet, une étude d'impact sur l'environnement a été réalisée afin de déterminer les composantes naturelles à surveiller [2].

On présente ici à titre d'exemple, les modalités de rejets, les résultats de mesures et de dilution des effluents associés aux mesures effectuées sur un bioindicateur et ce, pour 2 radioéléments caractéristiques des rejets : le ^{106}Ru et le ^{60}Co [2].

MODALITES DES REJETS

La modélisation des conséquences sanitaires a conduit à optimiser les caractéristiques des rejets dans le but d'en minimiser les effets sur l'environnement et les populations humaines. Ainsi, à partir des mesures de courants marins et de suivis de traceurs chimiques comme la rhodamine, et physiques comme des flotteurs lestés, un modèle physique de la courantologie locale a été validé, ce qui a

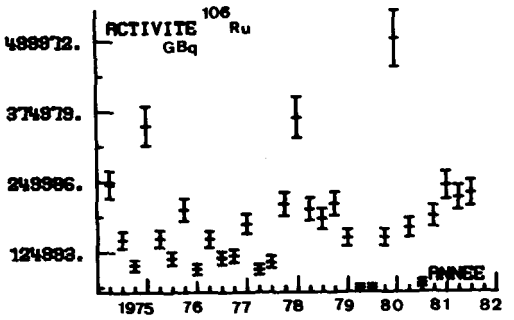


fig1a effluents liquides

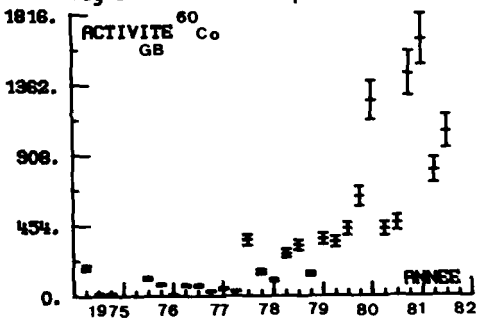


fig1b effluents liquides

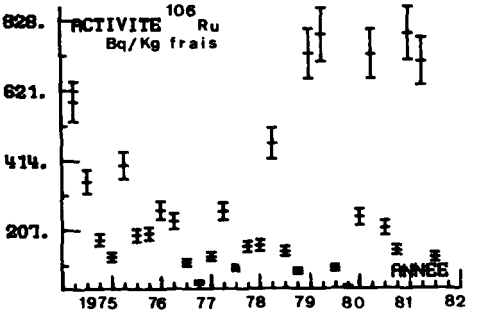


fig1c fucus serratus à equigrain

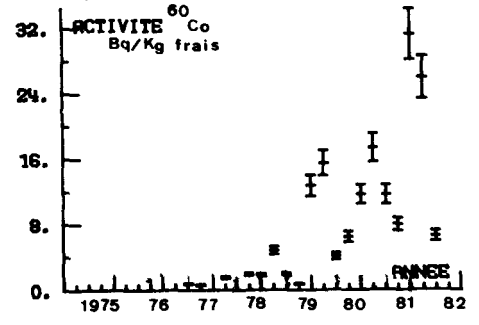


fig1d fucus serratus à siouville

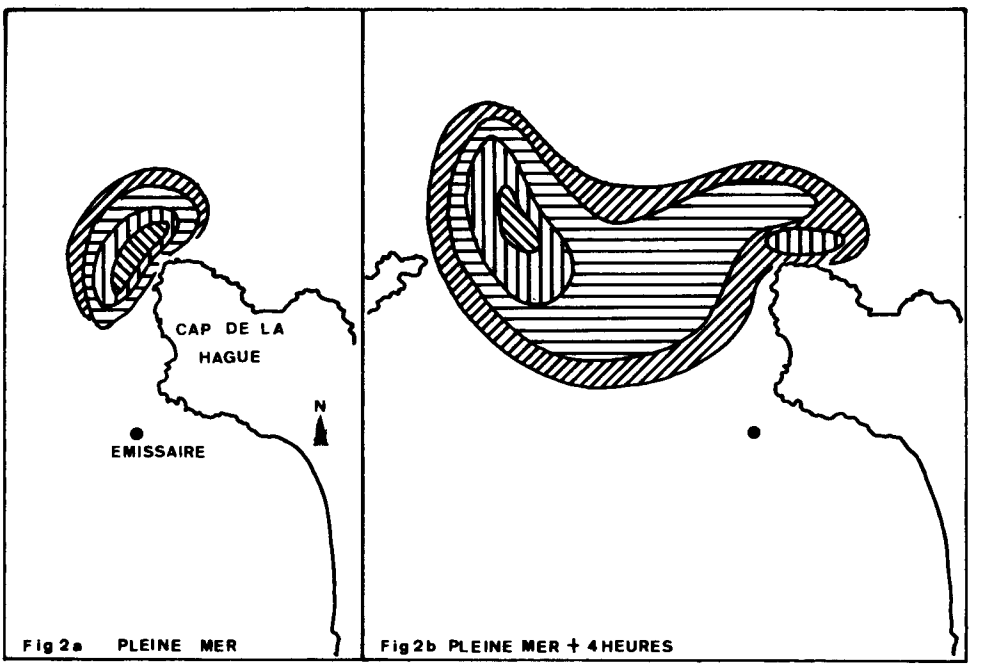
permis d'établir le lieu et le créneau horaire de rejet favorisant au mieux la dilution et la dispersion des radionucléides [4 et 5]. Le Cap de La Hague a été retenu pour sa forte capacité de dilution liée à de très forts courants de convection, de 3 heures avant la pleine mer à 1 heure après la pleine mer.

RELATIONS TEMPORELLES ENTRE LES REJETS ET LES MESURES DE SURVEILLANCE

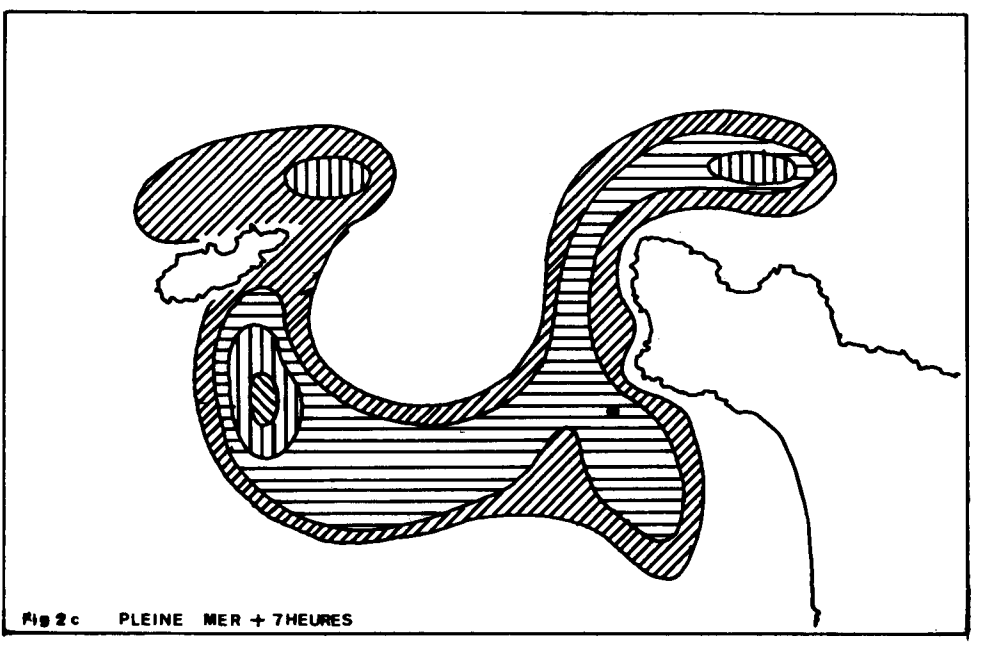
Les changements de combustibles traités et de modes de traitement des effluents ont conduit à des évolutions temporelles de rejets caractéristiques de chacun des radionucléides libérés dans l'environnement. Ainsi, le ^{106}Ruthenium est un élément rejeté en quantité variable pendant les 6 dernières années (figure 1a) par contre les rejets de ^{60}Cobalt s'accroissent depuis 1978 (figure 1b). Comme le montrent les figures 1c et 1d, ces fluctuations se retrouvent au sein des différents bioindicateurs échantillonnés à proximité de l'émissaire de rejets. L'étude statistique de l'ensemble des mesures permet alors d'établir le temps et le taux d'intégration des radionucléides entre les effluents et les bioindicateurs.

SUIVI SPATIO-TEMPOREL DU TRANSFERT DES RADIONUCLEIDES

La modélisation de la courantologie locale permet à l'heure actuelle de suivre avec plus de détail les évolutions temporelles et spatiales des masses d'eaux contaminées. Des champs de courants horaires ont été établis à partir d'interpolations filtrées de mesures eulériennes de courants, réalisées en 20 stations de la zone délimitée par la carte de la figure 2. La dilution des radionucléides à proximité de l'émissaire a été établie par la résolution spatio-temporelle de l'équation de diffusion-advection par une méthode de MONTE-CARLO. On résout cette équation dans un milieu soumis aux courants précédemment déterminés.



10^{-10} à 10^{-9} 10^{-9} à 10^{-8} 10^{-8} à 10^{-7} 10^{-7} à 10^{-6} les isoconcentrations sont exprimées en masse. l^{-3}



Évolution temporelle des concentrations pour un rejet massique unitaire

Les figures 2a, 2b et 2c, représentent la répartition des radionucléides solubles à différentes heures de la marée, obtenue par un rejet ponctuel et instantané effectué à partir de l'émissaire à 3 heures avant la pleine mer.

CONCLUSION

La modélisation validée par des mesures de surveillance, la connaissance des ressources maritimes exploitées loalement [6] et des régimes alimentaires associés à l'éthologie des populations normandes permet d'aboutir à l'estimation des conséquences sanitaires prévisibles sur les différents groupes de populations intéressés par les rejets marins. Compte tenu des termes de la modélisation utilisée jusqu'à ce jour, le calcul de l'exposition totale, (interne et externe) qui serait due à des rejets liquides maximaux en principaux radionucléides, est de l'ordre du millième de la limite de dose pour l'organisme entier et ce pour la population la plus sensible, c'est-à-dire celle constituée par les pêcheurs, consommant exclusivement un produit de pêche local.

BIBLIOGRAPHIE

[1] D. ROBEAU, D. CALMET, C. MADELMONT, J.C. NENOT Limites et conséquences sanitaires des rejets d'effluents radioactifs liquides AIEA, CN 43. Conférence Internationale sur la gestion des déchets radioactifs, SEATTLE, 16-20 mai 1983 (sous presse).

[2] Colloque de radioécologie marine, CHERBOURG, 22-25 avril 1964.

[3] Notes Techniques, Surveillance, Rapports COGEMA, 1975-1982.

[4] Commissariat à l'Energie Atomique, rejets en mer des effluents liquides du du Centre de LA HAGUE, 1965, p. 40.

[5] Université de GRENOBLE, Laboratoire de Mécanique des Fluides. Dispersion dans la Manche d'un effluent rejeté au Cap de LA HAGUE. Etude sur modèle réduit, 1963, p. 50.

[6] Commissariat à l'Energie Atomique, évaluation des conséquences sanitaires du rejet en mer des effluents radioactifs du Centre de LA HAGUE. Résultats de l'enquête sur l'alimentation des populations du Nord-Ouest du Cotentin, 1965, p. 20.

AN ASSESSMENT OF THE POPULATION DOSE DUE TO RADON-222 FROM MINE TAILINGS ON THE WITWATERSRAND

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1. Introduction

The contribution of radon as a source of radiation exposure to man is well established. At higher concentrations, such as is experienced in underground mines, epidemiological evidence of radiogenic lung cancer has been found and a quantitative life-time risk of 1,5 to 4,5 x 10^{-4} cases per Working Level Month (WLM) has been calculated [1]. Natural concentrations of radon in the atmosphere normally vary around 200 pCi.m^{-3} but could be an order of magnitude or more higher indoors, depending on factors such as ventilation and the type of building materials used. The effective dose equivalent to the average global population from inhalation of radon daughter products is 80 mrem.a^{-1} [2] with approximately 90 % of this due to indoor exposure and 10 % from outdoor exposure. The radon exposure pathway accounts for almost 40 % of the total dose from natural radiation sources to the population.

Enhanced radon releases from disposed mining wastes result in an additional radiation exposure. The population of 3,8 million presently living in and around Johannesburg, situated on the Witwatersrand, the centre of the South African gold/uranium mining industry, is exposed to these accumulated wastes, the product of a century of mining activities.

In this study the radium content of individual tailings dams on the Witwatersrand is determined and resultant radon emanation calculated. The dispersion of radon from this source is predicted by means of an atmospheric dispersion model using average meteorological conditions, and is verified by measurement. With knowledge of the population distribution, individual and collective doses are calculated.

2. Distribution of Mine Tailings

Since the discovery of gold on the Central Witwatersrand in 1886, 19 million kg of gold has been produced from 2,4 x 10^{9} t of ore [3]. Although uranium occurs as a byproduct of gold it has only been exploited since 1952 with approximately 30 000 t of U_3O_8 produced to date from the Central Witwatersrand area. Known reserves of uranium in tailings amount to 70 000 t while a further 50 000 t are estimated to be present [4].

The total radium associated with these tailings amounts to 50 000 Ci and is contained in 250 tailings dams with a total area of 6 475 ha. These dams are spread over a distance of 80 kms from the town of Springs in the east, to Randfontein in the west, with the metropolis of Johannesburg roughly in the centre of this east-west strip.

Uranium, and therefore radium concentrations, increase from east (10 pCi Ra.g^{-1}) to west (>100 pCi Ra.g^{-1}) with an average value of 20 pCi.g^{-1}, which is low compared to uranium mines in other parts of the world. However, factors such as the population of 3,8 million people living within 20 km of the strip, the large total quantity of radium and the extensive period over which this situation has lasted, result in a significant contribution of mine tailings to the collective dose exposure of the surrounding population.

3. Distribution of Radium

Results available from an airborne radiometric survey of an extensive part of the mining area conducted by the South African Geological Survey during 1976 provided a useful data base of radiometric anomalies which were used to determine the average radium concentration of individual tailings dams. The data from the airborne survey was calibrated by judicious sampling of a few selected tailings dams for which the radium content was determined. By applying corrections for the geometrical shape of the dam and the intersection with the flight line, a calibration curve of radiometric values versus radium concentration was obtained.

Tailings dams not showing an anomaly in the airborne survey were allocated a value of 20 $pCi.g^{-1}$. In areas not covered by the radiometric survey, major tailings dams were surveyed on foot with scintillation detectors, and representative samples were collected and combined for radium analyses.

4. Radon Emanation

The emanation of radon from different soil types has been studied in great detail by various authors both theoretically and experimentally [5,6,7] and it is well established that external factors such as barometric pressure, precipitation, wind and atmospheric stability conditions affect the escape of radon to the atmosphere. Diffusion appears to be dominant in deeper layers and can be described theoretically, but in shallow layers close to the surface convective processes, caused by external influences, modify theoretical predictions. In an extensive review, Wilkening [8] proposes a mean value of 0,42 $pCi.m^{-2}.s^{-1}$ for Radon flux from natural soils with a mean radium concentration of 1 $pCi.g^{-1}$.

Momeni et al [5] give the following specific radon fluxes for different tailings materials - acid tailings 0,63, alkaline tailings 0,30, loose sediments rich in clay 0,37, sandy soil 0,18, mixture of clays and heavy loams 0,28, and abandoned vitreous tailings 1,6. They suggest 1 as an average value for radiological assessment purposes.

The radon flux from the surface of different tailings dams on the Central Witwatersrand was measured experimentally and the results normalised to a unit radium concentration (Table 1). Applying a specific emanation rate of 0,4 $pCi.m^{-2}.s^{-1}$ to the 250 tailings dams, an annual release of 16 000 Ci. ^{222}Rn. results. More than 50 % of this total originates from 27 tailings dams and 90 % from 120 tailings dams.

5. Atmospheric Dispersion Modelling

The atmospheric dispersion model [9] was adapted for use with multiple sources and the radon distributions from numerous tailings dams were integrated for grid points at 1 km intervals to a distance of 20 km from each source. An additional source of radon was the ventilation air from shafts at active mines. This was calculated from a knowledge of down-draught air into the mines, by assuming ambient underground radon levels of 0,3 WL, and amounted to 3 000 $Ci.a^{-1}$. All shafts at a particular mine were considered as a single source and the distribution of radon for these sources were integrated with that of the tailings dams. Additional sources, such as releases from abandoned shafts, sand dumps and uranium mills, were not considered.

The individual areas of the tailings dams varied between 1 and 450 ha and their average height was taken as 5 m. For the purpose of the model, releases were considered to be 5 m above ground level and resultant concentrations at ground level were calculated.

The area under consideration is at an altitude of between 1500 and 1700 meters above mean sea level and is traversed from east to west by a principal watershed. The major topography is a series of parallel longitudinal valleys, orientated east-west, separated by rocky ridges. Over the tailings dams, generally situated on the southern slope of the watershed, a southward drainage of air is experienced under stable atmospheric conditions.

Meteorological data from 2 stations within a 50 km radius of the area, one to the east (Jan Smuts Airport) and one to the west (National Nuclear Research Centre, Pelindaba) were found to have comparable wind patterns and stability conditions. Using a modified Star method, [10] data from Jan Smuts Airport were used to provide correlations of wind speed, direction and Pasquill stability category.

Model predictions were verified at 12 measuring points around the Crown Mines site over a period of 1 month using site-specific source and meteorological data. Radon concentrations were measured with passive radon monitors and satisfactory comparison with predicted values was obtained.

6. Radiation Exposures

From the 1 km grid-pattern of radon concentrations, predicted by the model, average radon concentrations for each designated suburb in the region were calculated. Using population census data from 1980 it was possible to determine collective exposures in pCi-man.m^{-3}.

Exposure-dose relationships have been calculated by various groups. McDowell-Boyer *et al* [11] proposes a value of 4 rem.WLM^{-1} to bronchial epithelium of members of the public. The weighting factor (W_{TB}) to calculate the effective dose equivalent equals 0,06 resulting in a value of 0,24 rem.WLM^{-1} .The ICRP [1] recommends a value of 1 rem.WLM^{-1} for the effective dose equivalent to workers. Varying environmental conditions and behaviour patterns will result in a lower value for the population.

However, UNSCEAR [12] proposes values of 0,55 rem.WLM^{-1} and 1,1 rem WLM^{-1} for members of the public, for in-door and out-door conditions respectively. These values correspond to 0,28 and 0,55 mrem.a^{-1} per pCi.m^{-3}. A value of 0,37 for an average occupancy distribution of 16 hours indoors and 8 hours outdoors is used in this report. Assuming an equilibrium factor of 0,5 between radon and its short-lived progeny the collective exposures in pCi.man.m^{-3} were converted to man-rem.a (Table 2).

7. Conclusions

The annual release of radon from the tailings dams in the Witwatersrand area amounts to 19 000 Ci with peak concentration of 3 000 pCi.m^{-3} being predicted in the Randfontein area, which could result in an annual effective dose equivalent of 500 mrem to individuals. However, the average annual dose for the entire area is predicted as 70 mrem which is of the same order as that due to the radon contribution from natural sources. The annual collective effective dose equivalent to the population of the Witwatersrand due to enhanced radon releases from mine tailings is calculated as 2,6 x 10^{5} man-rem.

7. References

1. ICRP Pub 32 (1981) vol 6(1).
2. UNSCEAR (1982) Report. Annex B. U.N. New York.
3. Chamber of Mines (1982). Mining statistics, Johannesburg, South Africa.
4. Dept. of Geology, NUCOR - Private communication.
5. MOMENI, M H; Kisieleski, W E; Tyler, S; Zeelen, A; Yuan, Y; Roberts, C J (1979). Proceedings of Health Physics Society Simposium - Ed Watson, J E, Williamsburg, Va p 307-328.
6. TRAVIS, C C; Watson, A P; McDowell-Boyer, L M; Cotter, S J; Randolph, M L; Fields, D E (1979). NUREG/CR 0573, ORNL/NUREG-55.
7. KRANER, H W; Schroeder, G L; Evans, R D (1964) p 190. The Natural Radiation Environment II. University of Chicago Press, Chicago.
8. WILKENING, M H; Clements, W E; Stanley, D (1975) p 717. The Natural Radiation Environment II. University of Chicago Press, Chicago.
9. MOORE, R E et al, (1979). Oak Ridge Nat. Lab Tennessee ORNL 5532.
10. BOTHA, P (1983). Private communication.
11. McDOWELL-BOYER, L M; Watson, A P; Travis, C C. (1979). NUREG/CR 0574, ORNL/NUREG-56.
12. UNSCEAR (1982) Report. Annex D. U.N. New York.

TABLE 1

Measured Radon Emanation from Tailings dams.

Tailings dam	Radium concentration pCi/g	Emanation $pCi.m^{-2}.s^{-1}$	Specific emanation $pCi.m^{-2}.s^{-1}/pCi.g^{-1}$
1L22	8	2,9	0,4
1L24	15	5,2	0,3
1L26	9	3,9	0,4
3L32	9	2,7	0,3
3L44	6	3,8	0,6
4L4	6	3,1	0,6
4L33	18	3,0	0,2
		AVERAGE	0,4

TABLE 2

Collective exposure to and annual effective doses from Radon.

Area	Population	Exposure $pCi\text{-}man.m^{-3}$	Effective dose $man\text{-}rem.a^{-1}$
West Witwatersrand	317 880	$215x10^6$	$0,4x10^5$
Johannesburg	2 296 300	$846x10^6$	$1,6x10^5$
East Witwatersrand	1 183 960	$333x10^6$	$0,6x10^5$
TOTAL	3 798 140	$1\ 394x10^6$	$2,6x10^5$

HOW TO INTERPRET MEASUREMENTS OF INTERNAL CONTAMINATION

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SCOPE OF THIS PAPER

To judge measurements of internal contamination has been so far reserved to some scientifically oriented specialists. However, this must not remain necessarily so. Some basic understanding and some simple formulas should allow it to the non expert, to compare the measurements with the regulations. To provide this basic understanding and the very simple formulas needed for some important cases is the scope of this paper.

This paper must limit itself to some interesting simple cases. Nuclides with long effective half-life are not treated here. We deal only with cases, where the measured activity after a single intake becomes insignificant after one year.

CONCEPT FOR THE CALCULATIONS

The basis of the calculations are the dose limitation of ICRP-26 and the data given in ICRP-30. Data on urinary excretion from ICRP-10 are also used [1].
The scope of the calculations is to determine the probable committed dose equivalent or uptake and to compare it to the annual limit given by ICRP-30. We propose to abandon the idea of calculating the worst case, i.e. the assumption that any discovered contamination took place on the day after collection of the immediately preceeding bioassay sample (ICRP-10).

If we drop the assumption or information on the time of intake the calculations gain unexpectedly simplicity because we can drop some other assumptions derived from metabolic models! Of course, for a single intake, the uncertainty due to the lack of knoweldege about the time of intake is considerable, however, for recurrent intakes our calculation converges to the correct result. The remaining error depends mainly on the length of the intervals between the measurements, relative to the effective half-life of the nuclide.

SOME SIMPLE EXAMPLES

Simple cases are characterized by the possibility to measure the activity in the most important source organ. The isotopes of iodine are good examples. If we track the iodine in the thyroid with a calibrated NaI-crystal at reasonably short intervals, we can easily calculate the number of nuclear transformations in the thyroid (nnt_m) from these measurements:

$$nnt_m = \bar{A} \times T_e \qquad (1)$$

$\bar{A}$ is the mean of the measured activities (Bq), T_e the period (s) for which we would like to calculate the dose from the measurements. Fig. 1 shows, how the area representing the real number of nuclear transformations is approximated by the area under the measured activities.

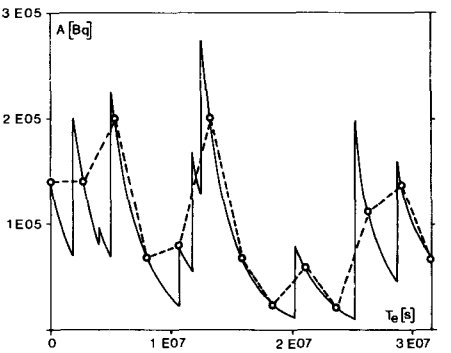

Fig. 1: Approximation of the number of nuclear transformations by the area under the measured activities

Now we have to add some dosimetry from ICRP-30. We would like to compare a measured value nnt_m to a value nnt_a corresponding to the annual limit. From there we calculate the result $I_{\%}$, the percentage intake to the annual limit of intake ALI.

$$I_{\%} = 100\% \times nnt_m / nnt_a \qquad (2)$$

The value of nnt_a results from the multiplication of the ALI by the number of nuclear transformations in the thyroid per unit intake (nnt). From the supplement to part 1 of ICRP-30 we calculate for nnt_a 2.3E12 and 3.0E11 for I-125 and I-131 respectively. Hence we obtain for a period T_e of one year the following simple formulas ($\bar{A}$ in Bq):

$$\underline{\text{I-125}} \quad I_{\%} = \frac{100\% \ \bar{A} \times 3600 \times 24 \times 365}{2.3E12} = \bar{A} \times 1.4E\text{-}03\ \% \qquad (3)$$

$$\underline{\text{I-131}} \quad I_{\%} = \frac{100\% \ \bar{A} \times 3600 \times 24 \times 365}{3.0E11} = \bar{A} \times 1.0E\text{-}02\ \% \qquad (4)$$

The results for ingestion (oral) or inhalation are identical in this case (except for rounding errors). The dosimetric calculation involves only one assumption for the weight of the thyroid. E.g. the assumed effective half-life cancels out in the multiplication of nnt by ALI.

A similar and important case is tritiated water. The concentration of tritium in urine (C_u) is an excellent measure of the internal irradiation. From the models used in ICRP-30 we obtain for 42 L in the body.

$$nnt_m = 42 \times \bar{C}_u \times T_e \qquad (5)$$

For nnt_a we find from nnt and ALI 3.5E15. Again for a period T_e of one year we obtain (C_u in Bq/L).

$$\underline{\text{HTO}} \quad I_{\%} = \frac{100\% \times \bar{C}_u \times 42 \times 3600 \times 24 \times 365}{3.5E15} = \bar{C}_u \times 3.8E\text{-}05\ \% \qquad (6)$$

Recently, Johnson [2] proposed a similar formula with 10% higher figures in order to include the irradiation from non-exchangeable tritium. He also pointed out, that the dose is estimated accurately and independently of individual differences in body water retention, if urine samples are obtained frequently enough.

A DIFFICULT EXAMPLE

To analyze samples of urine at intervals of e.g. 1 month is a popular method of routine survey. If urinary excretion follows an exponential law with only one time constant then the concentration C_u or daily excretion A_u must be proportional to the activity in some important source organ and we can assess $I_{\%}$ as outlined in the previous chapter. If the excretion function is more complicated, we have to use another way of calculation. Let us take P-32 as an example.

Exept for inhalation class W, which will not be treated here, the irradiation of the red marrow is the dominating factor. On the other hand we do not get any measurable signal in the urine from the P-32 in trabecular bone, the source organ for this irradiation. Therefore we have to rely very much on the metabolic model. The best we can do is to estimate the total activity in urine (tau_m) excreted during the period T_e on the basis of regular samples of the daily excretion (A_u). From the mean $\bar{A}_u$ we obtain

$$tau_m = \bar{A}_u \times T_e. \qquad (7)$$

Again we have to compare this to a value tau_a corresponding to the annual limit. We calculate this from ICRP-30 and ICRP-10 for oral ingestion:

$$tau_a = 0.8 \times 2.4E07 \times 0.9 \int_0^{\infty} (0.2e^{-t/0.7} + 0.05e^{-t/2.5} + 0.013e^{-t/11.7})\,dt \qquad (8)$$

Unfortunately the first term with a half-life of half a day would disturb the measurements. But we can eliminate most of this fast decaying component by the prescription that urine samples for dosimetric purposes should be taken regularly at Sunday evening or Monday morning before work. In this case we systematically loose the first two days, therefore the lower boundary of the integral in (8) should start at 2 days. If we evaluate the integral and if we put T_e in (7) equal to one year, we obtain ($\bar{A}_u$ in Bq/day).

$$\underline{\text{P-32}} \quad I_{\%} = \frac{100\% \times \bar{A}_u \times 365}{1.7E07 \times 0.19} = \bar{A}_u \times 1.1E\text{-}02\ \% \qquad (9)$$

For inhalation (class D) of P-32 the first two terms in (8) change: Instead of 0.8x2.4E07 Bq we derive from the lung model (47.6 % + 0.8x15.4%) x 3.4E07 Bq and obtain practically the same result (9). Inhalation class W is more complicated and not so important for P-32. With formula (9) one would underestimate the dose by about a factor of 2.

CALCULATION FOR KNOWN TIME OF INCORPORATION

If the time t elapsed between the intake and the time of measurement is known, this information can be used to increase the precision of the evaluation of the committed dose. We suggest to use this procedure only for serious cases and not for routine evaluation of small doses.

As soon as a measurement A(t), $C_u(t)$ or $A_u(t)$ is available, we would like to compare it to what we expect from the models. E.g. for P-32 the latter would be the non integrated form of (8). Given the urinary excretion $A_u(t)$ in Bq/day for the day t after intake, we calculate

$$\underline{\text{P-32}} \qquad I_{\%} = \frac{100\% \times A_u(t)}{1.7E07 \times (0.2e^{-t/0.7} + 0.05e^{-t/2.5} + 0.013e^{-t/11.7})} \qquad (10)$$

In the simple case of iodine and HTO we propose to work out the daily number of nuclear transformations measured versus expected (from the intake of 1 ALI). The integral of the expected number must be nnt_a, for the rest the function looks like the retention function. We obtain for $C_u(t)$ in Bq/L and A(t) in Bq

$$\underline{\text{HTO}} \qquad I_{\%} = \frac{100\% \times C_u(t) \times 42 \times 3600 \times 24}{3.5E15 \times e^{-t/14.4} / 14.4} \qquad (11)$$

$$\underline{\text{I-125}} \qquad I_{\%} = \frac{100\% \times A(t) \times 3600 \times 24}{2.3E12 \times e^{-t/57.7} / 57.7} \qquad (12)$$

$$\underline{\text{I-131}} \qquad I_{\%} = \frac{100\% \times A(t) \times 3600 \times 24}{3.0E11 \times e^{-t/10.9} / 10.9} \qquad (13)$$

If several measurements in the period from t_1 to t_2 are available we suggest to insert in the nominator of the equations (10) to (13) a graphical or numerical solution of the area $A_u(t)$, $C_u(t)$ or A(t) times $(t_2 - t_1)$ and in the denominator the corresponding integration from t_1 to t_2. The equations (3), (4), (6) and (9) are nothing else than such an integration for one year.

REFERENCES

[1] Annals of the ICRP, Pergamon Press
[2] Johnson J.R. (1982),The Estimation of the Effective Dose Equivalent from Tritiated Water Exposures Using Tritium Concentrations in Urine. Rad. Prot. Dosimetry 2, 245-247.

ANALYSIS OF EXTERNAL OCCUPATIONAL EXPOSURE TO NUCLEAR RADIATION - CLOR /POLAND/ 1979 - 1981

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METHODS

Dose measurements were made with individual photometric dosemeters [1]. Each dosemeter has a two-sided system of plastic filters /surface density of 50 mg/cm^2 and 300 mg/cm^2/, copper filters /thickness of 0.05 mm, 0.5 mm and 1.5 mm/ and zink-lead filters /thickness of 0.6 mm Sb and 0.3 mm Pb/. The Kodak Radiation Monitoring Film used has emulsion layers of different sensitivity, which permits dose equivalent measurements ranging from $5 \cdot 10^{-4}$ Sv to 10 Sv. Dose film badges are worn for a period of one month or 3 months depending on the exposure level to the persons working with ionizing radiation.

DATA PROCESSING

To make an analysis of the occupational exposure risk the results of annual dose equivalents were grouped in five ranges: below 0.005; 0.005 - 0.015; 0.015 - 0.05; 0.05 - 0.12 and above 0.12 Sv [2].

Cases of the annual dose equivalent values exceeding 0.05 Sv and those of exceeding the quarterly maximum permissible levels /i.e. 0.013 Sv for women in the reproductive age and 0.03 Sv for other workers/ were analyzed on an individual basis taking into consideration the age and sex of the person, type of radiation, kind of work with ionizing radiation.

All the establishments were classified into research, industrial, medical and others and the persons were divided into the following age groups: 18 - 30, 31 - 40, 41 - 50 and above 50 years.

RESULTS OF MONITORING

The number of persons occupationally exposed to the ionizing radiation and covered by individual dose monitoring by the Central

Laboratory for Radiological Protection was 7414, 7275 and 7455 in 1979, 1980 and 1981, respectively.

Persons employed in research establishments constituted over 40% of all persons monitored /47.8%, 43.9% and 43.3% in 1979, 1980 and 1981, resp./. The second largest group was that of persons employed in industry: 24.7%, 25.3% and 24.6% in 1979, 1980 and 1981, resp./. Medical institutions employed 18.5%, 22.2% and 22.8% in 1979, 1980 and 1981 of all the working force monitored. The remaining persons were those working in other branches of economy.

The results of annual individual dose measurements between 1979 and 1981 are shown in Table 1.

Table 1. Distribution of annual dose equivalents for persons working in establishments using ionizing radiation /1979 - 1981/. Values are given in per cent.

Type of establishment	Year	Dose equivalents range				
		<0.005 Sv	0.005 - -0.015 Sv	0.015 - -0.05 Sv	0.05- -0.12 Sv [x]	>0.12 Sv [x]
Research	1979	98.06	1.46	0.42	0.06 /2/	-
	1980	97.94	1.59	0.47	-	-
	1981	98.27	1.24	0.46	0.03 /1/	-
Industrial	1979	84.78	8.78	5.56	0.65 /12/	0.22 /4/
	1980	84.23	9.24	5.38	1.09 /20/	0.05 /1/
	1981	84.93	8.65	5.60	0.65 /12/	0.16 /3/
Medical	1979	89.27	9.49	1.24	-	-
	1980	90.89	8.49	0.62	-	-
	1981	92.24	6.88	0.82	0.06 /1/	-
Others	1979	98.49	1.21	0.30	-	-
	1980	97.92	1.28	0.64	0.16 /1/	-
	1981	98.40	1.31	0.29	-	-
All groups of establishments	1979	93.19	4.73	1.83	0.19 /14/	0.05 /4/
	1980	92.91	5.03	1.76	0.29 /21/	0.01 /1/
	1981	93.62	4.36	1.80	0.19 /14/	0.04 /3/

[x] In brackets are indicated numbers of persons whose dose equivalents exceeded 0.05 Sv.

About 16% of all the persons monitored in industry received annual dose equivalent higher than 0.005 Sv, while the percentage of those employed at research and other establishments was about 2% and the percentages for those working in medical institutions were 11% in 1979 and 8% in 1981 for the same dose equivalent value.

Annual dose equivalents above 0.12 Sv were found only in persons working in industrial defectoscopy. In 1979, 1980 and 1981 four, one and three cases were registered of values above 0.12 Sv /i.e. 0.22%, 0.05% and 0.16% of the total number of persons in this group/ respectively.

The numbers of cases where the quarterly maximum permissible dose was exceeded in persons working with gammagraphic units were 20, 27 and 17 for 1979, 1980 and 1981, respectively.

It is, therefore, clear that the highest exposure levels were obtained in persons employed in defectoscopy. The mean annual dose equivalents calculated for this group were twice as large as the mean values for the whole group monitored /Table 2/.

Table 2. Mean values for annual dose equivalents received by persons employed at various establishments /1980 - 1981/.

Type of establishment	Mean values of annual dose equivalents /in Sv/	
	1980	1981
Research	$0.139 \cdot 10^{-2}$	$0.133 \cdot 10^{-2}$
Industrial	$0.441 \cdot 10^{-2}$	$0.452 \cdot 10^{-2}$
Medical	$0.203 \cdot 10^{-2}$	$0.192 \cdot 10^{-2}$
Others	$0.134 \cdot 10^{-2}$	$0.103 \cdot 10^{-2}$
All groups of establishments	$0.229 \cdot 10^{-2}$	$0.223 \cdot 10^{-2}$

The mean values of annual dose equivalents were calculated from

$$\bar{D} = \frac{D_c}{N}$$

where D_c is the annual collective dose equivalent calculated by summing the values for annual dose equivalents received by individual persons in the group, and N is the number of persons in the group.

The annual collective dose equivalent for the whole group monitored was 16.6 man Sv /1980 and 1981/, when the persons employed in industry it was 8.1 man Sv in 1980 and 8.3 man Sv for 1981.

LITERATURE

1. Jasiak J., Musiałowicz T., Wysopolski J.: Personnel film monitoring in Poland in the years 1963-1965 /in Polish/, Report CLOR-58/D, Warsaw 1966.

2. Mazgajska E., Jasiak J., Wysopolski J., Musiałowicz T.: Analysis and evaluation of the professional exposure to ionizing radiation in Poland in the years 1976-1978 /in Polish/, Probl.Techn.Med., 1980, XI, 1, 37-45.

OCCUPATIONAL RADIATION EXPOSURE AT NUCLEAR POWER PLANTS IN JAPAN

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INTRODUCTION

A shortage of energy resources has been a worldwide concern since the onset of the so-called oil crisis. Especially in Japan, almost all energy resources are dependent on overseas suppliers, there is an urgent need for new sources of energy to substitute for the limited supplies of oil. Under these circumstances, nuclear power, among several new energy resources under development, has been expected to be one of the most reliable energy resources in Japan from both technological and economical standpoints.

The first nuclear power plant in Japan, the Tokai Atomic Power Station, started commercial operation in 1966. In the subsequent years, nuclear power developed rapidly despite the severe controversy surrounding the construction of nuclear power plants. At present, there are 24 nuclear power reactors in commercial operation (1 GCR, 12 BWR and 11 PWR as of October 1983), and the total capacity of 171,77 MW(e) ranks third in the world, following the United States and France. Nuclear power provides about 20% of the total electric demand in Japan.

As the number of nuclear power plants increased, the number of workers exposed to radiation in their jobs, as well as the total occupational exposure at nuclear power plants, has increased markedly. In this paper, statistical summaries of occupational exposures at nuclear power plants in Japan are presented, and some topics relevant to occupational exposures at nuclear power plants are discussed.

SOURCE OF DATA AND METHODS

In Japan, each nuclear-power-reactor licensee submits statistical summary reports of personnel monitoring information recorded during the fiscal year (April-March) in accordance with the requirement of the Nuclear Reactors Regulatory Law. These statistical summaries of occupational exposure at each nuclear power plant are compiled and published annually by the Resources and Energy Agency, Ministry of Trade and Industry.

In nuclear power plants in Japan, onsite workers are classified as "plant personnel" or "external personnel". The plant personnel are employed permanently in the nuclear power plant, while the external personnel work in the plant temporarily during routine or special maintenance of the reactor. Usually, three kinds of personal monitoring devices, such as film badges, TLDs and personal alarm meters are used together by radiation workers. The dose measured by film badges is designated as the official record of extenal exposure. In routine monitoring, film badges are developed once a month, and the minimum detectable limit of conventional film badges used in the nuclear power plants is around 10 mR.

The data presented here are based mainly on these official reports, with some rearrangement and additional graphical presentaions of the original data for the year 1970 through 1982.

RESULTS AND DISCUSSION

The annual collective doses and the number of workers at nuclear power plants have increased rapidly with the increase of nuclear power capacity(Fig. 1). The proportion of external personnel to the total workers in both the number of radiation workers and in the collective dose has been gradually increasing. In 1982, 86% of the total number of workers exposed and 94% of the total collective dose were occupied by external personnel.

The annual individual doses have averaged less than one-tenth of the annual dose limit for occupational personnel(0.05 Sv/yr) and no workers exceeded the 0.05-Sv limit since the start of nuclear power plants. However, it must be noted that some workers(e.g. external personnel) work in several different plants within a given year and are counted several times when the total for all plants is arrived at in the annual statistics.

One of the principle concerns of radiation protection at nuclear power plants is controlling the exposures of these transient workers. To ensure adequate recording of prior exposure of these transient workers, it is neccessary to establish a system capable of registering radiation dose of workers throughout the country. In this regard, Radiation Dose Resistration Center has started in 1977.

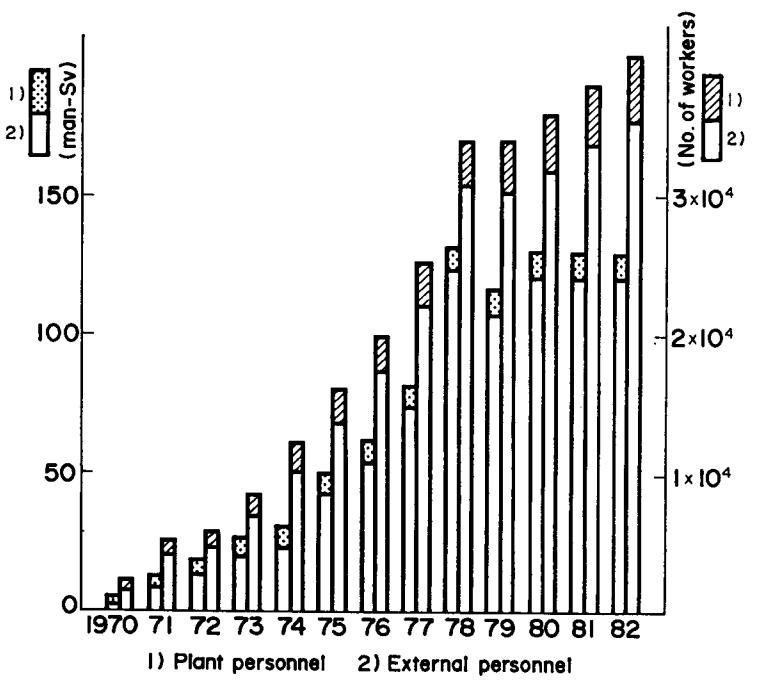

Fig. 1. Summation of annual collective dose and number of radiation workers at nuclear power plants in Japan.

The annual collective dose per reactor has fluctuated year by year and gradually increased as shown in Fig. 2. This trend is more remarkable in boiling-water reactors(BWRs) than in pressurized-water reactors(PWRs). The fluctuations indicate that special maintenance has been carried out at many nuclear power plants in the given years. The general rise reflects the increase in collective dose at each plant with plant age.

Many reasons will be proposed to explain the difference in average collective doses between BWRs and PWRs. One of the most likely reasons is the difference in design: BWRs have radiation areas in many locations, including the turbine building, while the radiation areas of PWRs are mainly within the reactor building. Because of this difference in design, more work is performed in radiation areas at BWRs than at PWRs, especially during reactor maintenance. This difference between BWRs and PWRs is the trend also observed in the nuclear power reactors operated in the United States(NUREG-0713,1981).

Most occupational exposure at nuclear power plants has been incurred during maintenance and repair rather than routine operation. The most of the collective dose(80% in 1980) is recieved by maintenance workers performing routine and special maintenance activities.

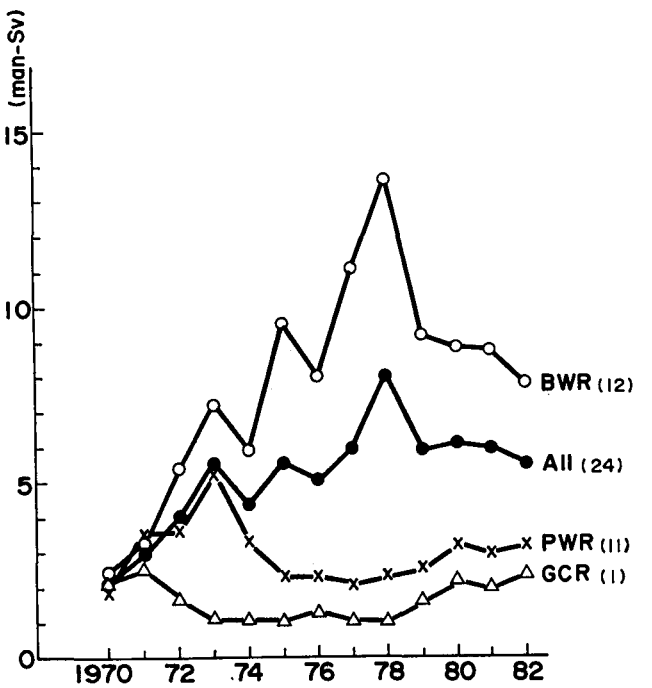

Fig. 2. Average annual collective dose per reactor by reactor types. (Numbers in parentheses indicate number of reactors in commercial operation at the end of 1982)

The annual collective dose of an individual plant varies widely, depending on the maintenance activities performed at the plant in a given year. The longer the reactor is shut down for the maintenance, a larger number of external personnel are engaged in these maintenance jobs on short-term assignment and the greater the occupational exposure will be. The annual collective dose per unit of electric output is almost in inverse proportion to the annual plant capacity factor(Fig. 3). The average capacity factor of BWRs was especially low in 1975 and 1977, corresponding to high collective doses. This indicates, there were major repairs to reactor components at many BWRs operating in Japan in these years, which caused unexpectedly high collective dose accumulations.

The basic goal of radiation protection is to keep exposure as low as reasonably achievable(ALARA). The ALARA principle must be applied not only to individual doses but also to collective population doses. How to achieve ALARA not only in individual doses but in collective doses is one of the most urgent tasks in radiation protection at nuclear power plants.

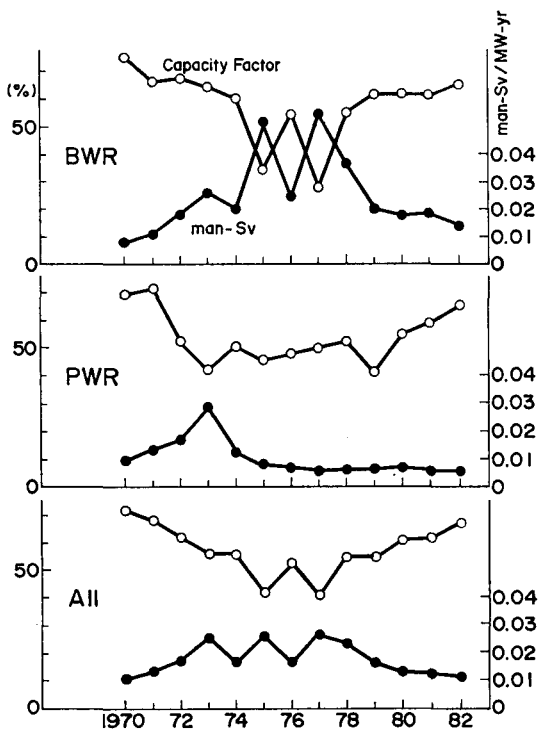

Fig. 3. Annual plant capacity factor and collective dose per unit of electric output, average of BWRs, PWRs and all plants (including a GCR).

$$\text{Annual plant capacity factor}(\%) = \frac{\text{annual output}(MWyr)}{\text{plant capacity}(MW)} \times 100$$

ANALYSE STATISTIQUE DES DONNEES D'IRRADIATION DES TRAVAILLEURS DU C.E.A.

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INTRODUCTION

Le système de surveillance individuelle des travailleurs du CEA exposés à une irradiation externe est basé sur des dosimètres photographiques qui sont portés un mois. Chaque travailleur possède donc douze relevés dosimétriques mensuels par an. Le premier objectif de la surveillance est de veiller à ce que la dose reçue reste, chaque mois, inférieure à la valeur autorisée ; le second est de pouvoir évaluer la dose individuelle dans le cadre de la vie professionnelle ; le dernier est de suivre dans le temps l'exposition de groupes de travailleurs, information indispensable à une bonne gestion du système de Radioprotection. Les trois notions, dose mensuelle, dose annuelle et dose sur la vie professionnelle ou dose-vie, sont nécessaires pour ces objectifs. Cette étude se limite aux problèmes posés par les doses annuelles et les doses-vie.

LE FICHIER DES DOSES INDIVIDUELLES ANNUELLES

La donnée de base est constituée par la dose individuelle annuelle, cumul des relevés dosimétriques mensuels. Dans ce cumul, tout relevé mensuel inférieur au seuil de sensibilité du dosimètre est mis à zéro. Cette pratique est totalement justifiée pour les très nombreux travailleurs qui doivent être surveillés mais ne sont jamais exposés. Pour les autres, il a été montré que la sousestimation résultant des doses mensuelles considérées comme nulles pouvait être évaluée au moyen de divers procédés présentés dans [1, 2, 3]. Pour une activité et une année données, la distribution des doses individuelles annuelles fournit aux gestionnaires d'installations le profil dosimétrique de l'activité. Classiquement l'état de la protection s'apprécie chaque année au moyen des deux indicateurs usuels que sont la dose individuelle moyenne et la dose collective. Suivis sur plusieurs années consécutives, ils permettent de saisir l'évolution de l'exposition au sein de l'activité. Quand il est nécessaire d'aller plus loin, la démarche usuelle est la modélisation de la distribution des doses annuelles. L'UNSCEAR [4] a proposé le modèle log-normal, controversé car trop souvent inadéquat et remis en question par certains auteurs [5] ; la variété des situations est telle, qu'a priori de nombreux modèles sont possibles et que toute tentative de modèle unique est vouée à l'échec.

DOSE-VIE

Pour le travailleur que l'on surveille il ne s'agit là que du cumul de ses doses annuelles durant sa vie professionnelle. Pour le responsable de radioprotection, par contre, la dose-vie du travailleur exerçant son activité au sein d'un groupe donné et dans un certain contexte pose un problème d'estimation. Pour une vie de travail qui se déroulerait au sein du même groupe, c'est-à-dire sur une certaine période de temps, la pratique usuelle consiste à sommer les moyennes des doses individuelles annuelles relatives à la période considérée. Cette pratique est parfaitement justifiée si le groupe de travailleurs est homogène quant aux postes de travail et quant à l'âge des individus le composant. Les phénomènes de spécialisation dans un poste, de vieillissement des individus, de glissement d'un poste à d'autres seront mieux perçus si l'on suit durant la période considérée une cohorte de travailleurs. C'est à partir d'un fichier où chaque individu apporte son histoire dosimétrique qu'il est possible de saisir de tels phénomènes.

Dans une étude sur des mineurs d'uranium, il a été montré [6] que la spécia-

lisation dans le poste est un facteur dont il faut tenir compte : la forte liaison existant entre les doses successives et ce généralement pour chaque mineur rend délicat tout amalgame des doses au sein d'un groupe considéré pourtant comme relativement homogène. Le comportement d'une cohorte et celui du groupe dont elle est issue est étudié dans les trois autres exemples suivants :

1. RP : Réacteur de puissance à eau lourde ; postes autres que les Services Généraux ; période 1971-1981 ; cohorte de 78 travailleurs ;
2. EC : Fabrication d'éléments combustibles au Plutonium ; période 1971-1978 ; cohorte de 72 travailleurs ;
3. TU : Fabrication de sources de transuraniens à usage industriel ; période 1974-1981 ; cohorte de 25 travailleurs.

Le tableau ci-dessous récapitule les doses annuelles par individu $\overline{D}$ (en mSv) dans chacun des groupes et dans la cohorte qui lui est associée en fonction de l'année calendaire.

Année	RP			EC			TU		
	Effectif total	$\overline{D}$	$\overline{D}$ coh.	Effectif total	$\overline{D}$	$\overline{D}$ coh.	Effectif total	$\overline{D}$	$\overline{D}$ coh.
1971	269	.4	.9	157	2.6	3.3			
1972	164	2.2	3.	128	4.1	5.			
1973	169	2.6	3.2	110	2.6	3			
1974	138	2.2	3.2	110	3.6	4.2	51	15.6	18.6
1975	129	2.3	3.	126	2.9	3.7	51	13.4	16.5
1976	132	2.7	3.	114	2.3	2.6	53	13.4	16.6
1977	156	4.9	4.7	111	2.	2.	54	15.8	16.8
1978	109	4.3	4.9	162	1.7	2.	58	12.	13.9
1979	153	2.5	3.1				64	8.	9.9
1980	201	3.8	4.2				63	6.8	7.8
1981	216	5.	5.5				63	7.7	9.

Chaque année, la dose individuelle moyenne dans la cohorte présente toujours un léger excès par rapport à celle du groupe pris dans son ensemble. Est-il possible d'étendre ce constat à tout groupe ? Non, car il existe des situations où les permanents sont en moyenne moins exposés que l'individu moyen du groupe et d'autres situations où la moyenne de la dose annuelle est la même que l'on soit dans la cohorte ou dans le groupe.

Il est aussi nécessaire d'étudier la distribution des doses individuelles au cours de la période observée. Au sein de la cohorte, il est classique de le faire au moyen des distributions annuelles des doses. Une autre information vient du calcul des coefficients de corrélation linéaire entre les doses reçues au cours d'une année et celles reçues au cours des années ultérieures. Enfin, il est utile de voir quelle est la distribution des doses individuelles reçues en moyenne par an sur la période, cette dernière notion n'étant pour chaque individu que la moyenne sur cette période de ses doses annuelles, soit encore à un facteur près le cumul de ses doses.

Le tableau suivant donne les estimations des coefficients de corrélation linéaire entre la dose D_i reçue pendant l'année i et la dose D_{i+1} reçue pendant l'année i+1, puis entre D_i et D_{i+2} etc... Ces liaisons temporelles sont donc relatives au nombre d'années d'écart entre les diverses doses.

Ecart (ans)	1	2	3	4	5	6	7	8	9	10
Cohorte RP	.77	.71	.70	.71	.72	.65	.68	.73	.69	.61
Cohorte EC	.75	.61	.46	.30	.23	.19	.13			
Cohorte TU	.73	.64	.62	.55	.58	.55	.46			

Une forte corrélation statistique apparaît entre les doses annuelles successives de chaque travailleur, liaison qui s'explique essentiellement par la spécialisation dans le travail. Ceci signifie que les cohortes étudiées et donc les groupes dont elles sont issues, présentent des tâches diversifiées au sein d'une même activité. On trouve dans [7] des calculs du même type appliqués à d'autres cohortes. Ces liaisons ont pour conséquence immédiate que la distribution des doses individuelles reçues en moyenne par an ne peut pas être correctement modélisée avec la démarche suivante :

- d'abord, calculer la moyenne et l'écart type de toutes les doses annuelles de la période ;
- puis retenir le modèle gaussien dont les paramètres sont la moyenne calculée et l'écart type divisé par la racine du nombre d'années observées.

Cette démarche suppose implicitement que les doses sont chaque année distribuées aléatoirement au sein du groupe de travailleurs, ce qui est rarement le cas.

CONCLUSION

Pour un groupe de travailleurs, les deux indicateurs statistiques que sont la moyenne des doses individuelles annuelles et la dose collective sont insuffisants pour traduire la situation dosimétrique. La distribution des doses annuelles fournit bien toute l'information mais il n'existe pas un modèle probabiliste suffisamment large pour permettre une comparaison aisée entre les distributions d'une année sur l'autre ou de groupe à groupe. En ce qui concerne la dose-vie, son estimation par une simple valeur moyenne n'est pas satisfaisante en regard de la diversité des situations rencontrées.

REFERENCES

[1] JAMMET H., MECHALI D., NENOT J.C., BRENOT J.
Distribution des doses individuelles et interprétation statistique.
Colloque international sur l'application du système de limitation des doses, AIEA-SM-258, Madrid, 1981.

[2] DARBY S.C., KENDALL G.M., GREENSLADE E.
Patterns of dose incurred by workers on the NRPB' dose record keeping service, JRSP, vol. 2, n° 4, 1982.

[3] GILBERT R.O., KINNISON R.R.
Statistical methods for estimating the mean and variance from radionuclide data sets containing negative unreported or less-than values
Health Physics, vol. 40, pp. 377-390, 1981.

[4] UNSCEAR
Sources and effects of ionizing radiation
United Nations, New-York, 1977.

[5] KENDALL G.M., DARBY S.C., GREENSLADE E.
Patterns of dose incurred by workers on the NRPB'dose record keeping service. Annual doses. JRSP, vol. 2, n° 3, 1982.

[6] BRENOT J., PIECHOWSKI J., TIRMARCHE M.
Statistical analysis of results on miners'exposure
Congress "Radiation hazards in mining", Golden, Colorado, 1981.

[7] ILARI O., JOHNSTON P.
Study on the distribution of individual doses to workers
Internal Report.

OCCUPATIONAL RADIATION EXPOSURE IN NUCLEAR MEDICINE[+)]

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The use of radioactive material in nuclear medicine is associated with radiation exposure of patients, staff and part of the population. The radiation protection regulations of the Federal Republic of Germany guarantee the protection of the individual as well as the general public from damage caused by the radiation from radionuclides. Accordingly, the user is required to ensure that the radiation exposure of the persons involved be kept as low as is reasonably achievable. A compromise is often necessary in order to fulfil this requirement, when taking into account the different nuclear medical application of radionuclides in therapy, diagnostic procedures and research. The knowledge of the extent of radiation exposure and its main sources is certainly necessary in order a reasonable reducation.

In a large hospital (University Hospital, Homburg (Saar), 2ooo beds) the use of radionuclides was determined with the aim of a balance of the radionuclide flow through the clinic and the resulting radiation exposure for the persons involved.

Methods

Determination of external occupational exposure: this is based on the result of individual monitoring by film dosimeters. In addition on this the dose values were taken from ionization chambers worn as monitors and the result of dose measurements obtained with a portable scintillation dosimeter at the place of work.

Internal exposure: this was calculated for staff after the results of internal contamination monitoring using the MIRD concept(1) (2).

Area survey of environment and place of work: we analysed wipe tests taken from surfaces, and air and sewage samples with regard to γ-rays. Samples also containing ß emitting nuclides were examined in a low level liquid scintillation counter.

Results: Table 1, 2, 3 and 4

Surface contamination in working areas was seen in a high percentage. It results from radioiodines ^{51}Cr, ^{59}Fe, ^{99m}Tc, ^{113m}In and ^{198}Au (3).

[+)] with Support of the Bundesminister des Innern of the Federal Republic of Germany

Radio-nuclide	Acquired x1o^{1o}Bq/a	Decay %	Westeair %	Sewage %	Refuse %
^{123}J D	14	65	o.o3	2o	15
^{124}J D	o.o67	5o	o.o4	35	15
^{125}J D	o.16	3	2	5o	45
^{131}J T	115	9o	o.o5	9	o.2
^{131}J D	3	3o	o.o3	6o	1o
^{75}Se	o.1	27	1.o	69	3
^{198}Au	o.16	6o	-	38	2
^{51}Cr	1.9	3o	-	68	2
^{59}Fe	o.1	2o	-	78	2
^{99}Mo	93	99.99	-	-	o.oo1
^{99m}Tc	19o	65	-	3o	5
^{32}P	o.o19	9o	-	8	2
^{89}Sr	o.o3	25	-	73	2
^{14}C	o.o1	o	?	65	35
^{113}Sn	o.16	99.99	-	-	o.oo1
^{113m}In	15	9o	-	8	2
^{3}H	o.o56	o	o.5	55	44.5

Tab. 1: Balance of acquisition and disposal of radioactive material at the University hospital Homburg (Saar)
?: Loss by evaporation established, but quantitative results at present not available

Discussion

Most of the aquired activity (3.7 x 1o^{12}Bq) is lost by physical decay or leaves the hospital with sewage (Tab. 1). Little is seen in refuse and air.

Occupation radiation exposure results mainly from physical decay inside the hospital and from radioactivity in the air which leads to incorporation of radioisotopes from iodine and tritium (Tab. 2). ^{99m}Tc, ^{59}Fe, ^{58}Co and ^{22}Na were not really incorporated but found on skin and clothes.

The application on the patient himself leads to an internal irradiation of the organism, the distribution of which is almost homogenous and partly more inhomogenous. The information supplied by giving the mean dose is not very helpful, as, among other things, the frequency of any particular method of examination would have to be taken into account. The external radiation orginating from other patients after application of the radio pharmaceutical is in comparison to the internal negligibly low.

Number of persons	percentage with detectable incorporation	Incorporated nuclide	Interval (Bq)	Activity Average value (Bq)	processed activity ($x10^8$ Bq/a)
36o	15	^{131}J	4o-185ooo	3ooo	33
	1	^{123}J	4o-148ooo	15oo	3.7
	4	^{99m}Tc	4o-148ooo	185oo	11o
	2	^{59}Fe, ^{58}Co, ^{22}Na	4o- 148oo	11oo	o.oo4
4o	95	^{125}J	4o- 185oo	3ooo	o.15
447	7o	^{3}H	4o- 185oo	66oo	o.o2

Tab. 2: Results of the incorporation monitoring

Place	Nuclide	Average value (mBq/cm^3)	Interval (mBq/cm^3)	Percentage of workplaces with a detectable contamination with this nuclide
Lab 1	^{51}Cr	63o	4 - 63oo	48 %
	^{59}Fe	1o	- +	4 %
	^{99m}Tc	12o	1 - 37ooo	36 %
	^{113m}In	125	1.5- 37o	12 %
	^{123}J	65	o.7-1184ooo	56 %
	^{124}J	33	5 - 1oo	24 %
	^{125}J	115	o.7- 4o	32 %
	^{131}J	3oo	o.4- 48oo	76 %
	^{198}Au	6	-	4 %
Lab 2	^{125}J	78	o.7- 66o	94 %
Lab 3	^{51}Cr	5	4.8- 6	1oo %
	^{59}Fe	2	o.7- .3	75 %
	^{125}J	9	-	25 %

Tab. 3: Surface contamination at different working places
+: Single values

The external exposure of the staff originates from radionuclide sources (storage containers, injections) in the laboratory and examination rooms, as well as the activity applied to the patient. In the area of diagnostic an average dose value of o.8 to 1.2 mGy/a. Intracavitary therapy with ^{226}Ra has to be considered separately. A small group of coworkers are immediately engaged with the posing and removal of the socalled ^{226}Ra needle. These persons receive in the Homburg Hospital the highest occupational radiation exposure of about 12 mGy/a; in individual cases the dose limit of 5o mGy/a is reached.

Group of persons	Type of exposure	Type of dose determination	Average Dose ($x1o^{-5}$Gy/a)	Interval Dose ($x1o^{-5}$Gy/a)
Patient	internal	MIRD Concept	-	o.7 -1ooooo
	external	Scintillation dosimeter	3	1 - 1o
Staff	internal	Whole body counter and MIRD	2	n.n.x - 5oo
	external	Film badge D	8o	n.n. - 15oo
	external	Film badge T (^{226}Ra)	1ooo	n.n. - 5ooo
	external	Scintillation dosimeter D	14o	n.n. - 15oo
	external	Contamination of room surfaces	o.oo6	n.n. - 6
Parts of the population	internal	General calculation basis BMI 1979	-	$5x1o^{-6}$ - 5o

n.n.x: not detectable D: diagnostic T: therapy

Tab. 4: Radiation exposure by the use of radioactive isotopes in nuclear medicine

In comparison, the external irradiation by contamined room surface contributes little to the total burden (Tab. 3), especially if one considers that the given values have to be assumed to be maximum values.

Finally, parts of the population can be included in the circle of persons exposed to radiation through nuclear medical use of radioisotopes, in addition to the naturally radiation background. The "General Basis for Calculation of the Bundesminister for Internal Affairs for the Radiation Exposure through Radioactive Drainage in Waste Air or in Survace Waters" supply rules to assess the expected radiation exposure of individuals in the most unfavourable cases.

This leads to an overestimation of the real exposure. The maximum value of the internal presented must therefore be considered as the maximum possible radiation dose. This is below the permitted limit for parts of the population.

References

(1) LOEVINGER, R., M. BERMAN
A schema for absorbed dose calculation for biologically distributed radionuclides. MIRD-Pamphlet 1, J.Nucl.Med.,9.Suppl. 1 (1968), revised 1976

(2) ROEDLER, H.D., A. KAUL
Radiation absorbed dose from medically administered radiopharmaceuticals. In: Biomedical Dosimetry, IAEA-Publ.STI/PUB/4o1, Vienna (1975), 665

(3) GLÖBEL, B., K.D. KELLER, G. KELLER, H. LEHNEN, C. ANDRES
Strahlenexposition durch die nuklearmedizinische Verwendung radioaktiver Stoffe. Nuc-Compact 12, 6 (1981), 242-248

RADIATION INCIDENT INVOLVING AN X-RAY FLUORESCENCE SPECTROMETER

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An incident will be described in which a scientific officer received a high absorbed dose, from an x-ray fluorescence analyser, to three fingers of one hand resulting in injury to two of them. The absorbed dose to the long finger was 213 Gy, to the index finger it was 134 Gy and to the thumb 17 Gy. The two fingers were relatively uniformly exposed while the thumb was exposed over a small area. The two fingers suffered permanent injury while the thumb appears to have recovered.

The paper will cover the incident, its investigation and the dosimetry used to assess doses received. The remedial action taken to prevent a repeat of the incident will also be given.

This particular incident came about on the one hand as a result of an operator trying out a new procedure without having first carried out a safety check, and on the other hand, as a result of safety requirements recommended by the manufacturer not having been adopted.

The main conclusions drawn from the incident are:

1) x-ray fluorescence equipment with all its in built safety features is not automatically safe;

2) new or novel procedures should be carefully investigated before being tried out; and

3) as has previously been found, fluorescence analysers can deliver high radiation doses in a very short time.

PROBABILISTIC ASSESSMENT OF ORGAN DOSES AFTER ACCIDENTAL RELEASES FROM NUCLEAR POWER PLANTS

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1. Introduction

Results of risk assessments for facilities of the nuclear fuel cycle are so far nearly exclusively presented in the form of frequency distributions and expectation values of the number of fatalities. These final results are less appropriate when conclusions should be drawn with respect to the planning of emergency actions and medical countermeasures including the evaluation of criteria for these actions. To judge the necessity, effectiveness and feasibility of protective measures and medical care, information about the radiation dose levels to be expected after accidental releases of activity as well as the number of persons exposed should be available.

For this reason a modified version /1/ of the accident offsite consequence model UFOMOD /2/ of the German Risk Study /3/ was developed to calculate the 50 a doses after accidental releases to the organs bone marrow, bone surface, lung, thyroid, gonads and to the whole body. Additionally, acute bone marrow doses (up to 30 days) and 1-year lung doses both relevant for early fatalities (EF) can be assessed. Analysing recalculations were performed employing the release categories established in the German Risk Study (GRS).

2. Results

2.1 Individual Doses

Activity releases result in space-dependent concentrations of radionuclides in the air and on the ground. The variability of the possible meteorological conditions during the dispersion process determines the possible concentrations and thereby the variation in the doses to be expected. Moreover, protective actions considered in the accident consequence model depend on dose levels. This causes further variation in the actual doses to individuals. With the results of accident consequence calculations based on the 4 x 115 different weather sequences of the GRS, frequency distributions of the above-mentioned organ doses were built for each mesh of the radial grid (4 meshes per decade). From these distributions, characteristic quantities like mean values and percentiles can be derived. As an example, Fig. 1 shows the distance dependent 95%-fractiles of the acute bone marrow doses calculated under the assumption of releases due to the categories FK1 to FK8. They can be interpreted as doses to individuals at any azimuthal position, which are not exceeded in 95% of all accident consequence situations. In particular, only 5% of all situations lead to radiation doses greater than 100 rad (threshold dose for EF) at distances beyond about 2 km from the site.

2.2 Population Doses

Dependent on wind direction and site, different population distributions are affected by the dispersed radioactivity. For this

reason, calculations based on the 19 sites considered in the GRS were performed to assess the spectrum of the possible numbers of persons exposed. The results were linked in probability distributions of the number of persons within a given dose interval for each radial grid mesh, release assumed. These 3-dimensional probability distributions for each distance interval, release category and organ are not easy to review, interprete and present completely. A reasonable reduction of information is necessary which has regard to the special application. So e.g. the mean and the maximum number of persons within a given dose range or a given distance could be sufficient information. Typical results are exemplarily shown in Fig. 2 and 3 for the release category FK2 of the GRS.

In Fig. 2 the number of persons with acute bone marrow doses above the dose levels 50 rad, 100 rad (threshold dose for EF) and 500 rad (LD-50 for EF) are given in dependence of the distance from the site. The curves show, that such doses occur at distances up to about 20 km. The total numbers of persons affected with doses of the different ranges are given in Table 1, showing the number of EF calculated with UFOMOD/B3, too. These numbers as well as the corresponding distributions allow to analyse the collective early fatality risk and to make conclusions on the effectiveness and the necessity of administrative and medical countermeasures.

Fig. 3 shows the same distribution type for the 50 a-thyroid doses, whereby the dose ranges are 15 rad (dose limit established in §28(3) of the German Radiation Protection Regulation), 100 rad and 1000 rad (begin of a non-linear dose-risk-relationship for thyroid /4/). Thyroid doses above 15 rad occur at all distances up to 540 km, where the site specific calculations terminate /2,3/. Even the number of persons with high dose values is rather large and non-linear dose risk curves can influence the predictions of the number of late thyroid cancer fatalities shown in Table 1, too.

3. Conclusions

The detailed graphical and numerical presentation of risk assessment results on the basis of radiation doses may provide more insight into the nature of risk from nuclear facilities. This may be helpful in decision-making processes as well as in the discussion of acceptance problems.

/1/ J. Ehrhardt, U. Zöller, "Darstellungsformen für individuelle und kollektive Strahlendosen nach unfallbedingten Aktivitätsfreisetzungen", Report KfK-3562 (1983)

/2/ A. Bayer et al., "The German Risk Study: Accident Consequence Model and Results of the Study", Nucl.Techn. 59, 1982, p.20-50

/3/ "Deutsche Risikostudie Kernkraftwerke", Verlag TÜV, Rheinland, Köln, Hauptband (1979), Fachband 8 (1981)

/4/ E. Oberhausen, "Die Dosis/Wirkungs-Beziehung bei der Strahlenexposition", GRS-Fachgespräch 1982, Report GRS-52 (1983)

	Number of Early Fatalities	Number of Persons with Acute Bone Marrow Doses		
		D>50rad	D>100rad	D>500rad
$\bar{P}$	29,8	1 551	559	25,2
	Number of Late Fatalities by Thyroid Cancer	Number of Persons with Radiation Doses to Thyroid (50a)		
		D>15rad	D>100rad	D>1000rad
$\bar{P}$	1233	3 096 000	243 000	12 780

Tab. 1: Number of Early and Late Fatalities in Correlation with the Number of Persons Exposed to the Corresponding Organ Doses (Release Category FK2)

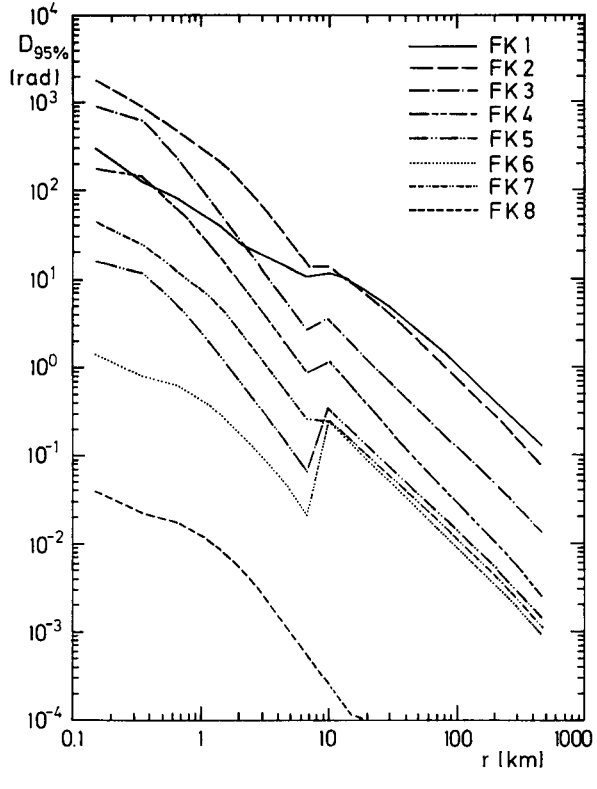

Fig. 1: 95%-Fractiles of the Individual Acute Bone Marrow Dose Distributions

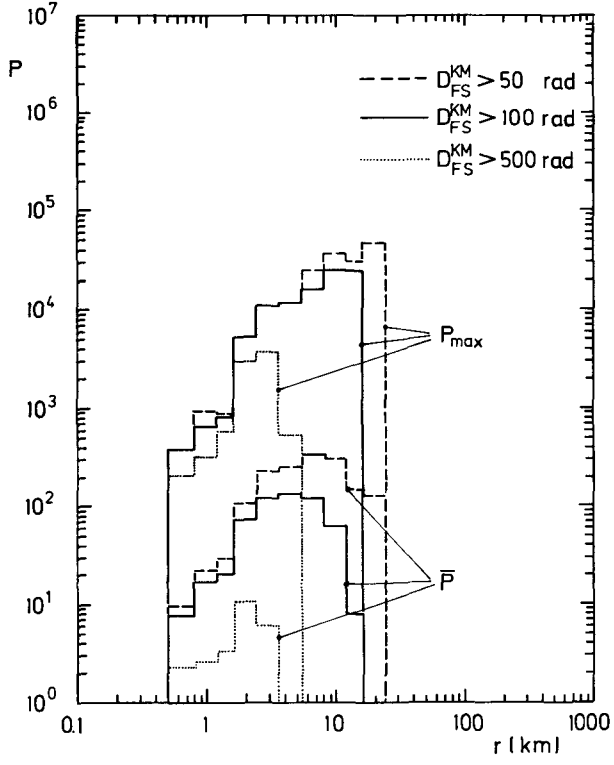

Fig. 2: Mean and Maximum Number of Persons with Acute Bone Marrow Doses for Release Category FK2

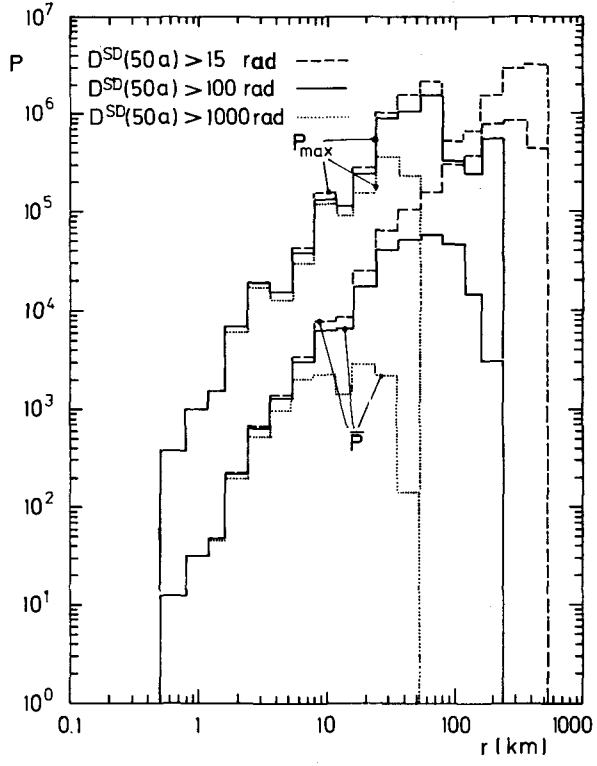

Fig. 3: Mean and Maximum Number of Persons with 50 a-Doses to Thyroid for Release Category FK2

BESTIMMUNG UND BEWERTUNG VON STÖRFALLFOLGEN NACH KURZZEITIGEN AKTIVITÄTSFREISETZUNGEN AUS KERNKRAFTWERKEN: EINE NEUERE METHODE

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EINFÜHRUNG UND ZIELSETZUNG

In der Bundesrepublik Deutschland (BRD) darf, insbesondere aufgrund gesetzlicher Anforderungen, die Genehmigung zum Bau und Betrieb von Kernreaktoren nur dann erteilt werden, wenn eine dem Stand von Wissenschaft und Technik entsprechende Vorsorge gegen Störfälle und ihre möglichen Folgen gegeben ist. Als Instrumentarium zum Nachweis dieser Vorsorge bedient man sich im Genehmigungsverfahren für nukleare Anlagen u.a. entsprechender Modellvorstellungen, mit deren Hilfe die möglichen Störfallfolgen deterministisch bestimmt werden.

Unter einer deterministischen Störfallfolgenabschätzung soll hier die Verfahrensweise verstanden werden, bei der zur Bestimmung der potentiellen Auswirkungen ein ganz bestimmter Geschehensablauf (Szenario) mit entsprechenden Parametern - häufig solche konservativer Art - zugrunde gelegt wird.

Diese Verfahrensweise ist verschiedentlich kritisiert worden, und zwar aus unterschiedlichen Gründen. So bleibt vor allem die Variabilität von Modellparametern sowie deren mögliche Interdependenzen außer Betracht. Weiterhin besteht die Gefahr,daß bei einer ausschließlichen oder partiellen Verwendung von "konservativen" Parameterwerten die möglichen Störfallfolgen weit überschätzt werden und so zu falschen Schlußfolgerungen, z.B. bezüglich eventueller Gegemaßnahmen, führen können.

Überlegungen, diesen Kritikpunkten in geeigneter Form Rechnung zu tragen, haben dazu geführt, die Anwendbarkeit statistischer Methoden zur Störfallfolgenanalyse eingehender zu untersuchen sowie Möglichkeiten zur Bewertung von Störfallfolgen aufzuzeigen. Als Anwendungsbeispiel wird hierzu ein Ingestionsdosismodell zur Beschreibung des Aktivitätstransportes über Nahrungsketten nach kurzzeitigen Aktivitätsfreisetzungen und der daraus resultierende Strahlenexposition der Bevölkerung gewählt.

METHODIK

Im Gegensatz zur deterministischen Verfahrensweise, bei der jeder Prozess eines Störfallfolgenmodells durch einen festen Parameter repräsentiert wird, sind bei der statistischen Vorgehensweise die Modellparameter stochastische Variablen mit bekannten Verteilungsfunktionen. Mathematisch gesehen ist somit die Bestimmung der Häufigkeitsverteilung der Ergebnisgröße, z.B. der organspezifischen Äquivalentdosis in der Bevölkerung, Ziel der Untersuchung. Die statistische Betrachtungsweise liefert somit nicht nur eine Aussage über das Ausmaß potentieller Störfallfolgen, sondern auch über die Häufigkeit, mit der ein bestimmtes Störfallausmaß auftreten kann.

In der Statistik sind verschiedene prinzipiell anwendbare Verfahrensweisen verfügbar, die für solche Problemstellungen

herangezogen werden können. Als ein relativ einfaches und flexibles numerisches Lösungsverfahren hat sich vor allem die Monte-Carlo Simulationsmethode erwiesen, die auch in dieser Untersuchung Anwendung findet.

INGESTIONSDOSISMODELL

Das der folgenden Untersuchung zugrunde liegende Ingestionsdosismodell zur Bestimmung der alimentären Inkorporation radioaktiver Stoffe und der daraus resultierenden Strahlenexposition entspricht, von wenigen Ausnahmen abgesehen, der Verfahrensweise, die in [1] für eine Beurteilung von Auslegungsstörfällen in Kernkraftwerken vorgeschlagen worden ist.

Danach ist bei der Durchführung des Vorsorgenachweises eine Aktivitätsinkorporation über verschiedene Nahrungsmittelpfade, nämlich den Konsum von Milch, Fleisch und pflanzlichen Produkten zu unterstellen. Die Beschreibung der Aktivitätsausbreitung in der Atmosphäre erfolgt nach dem GAUSS´schen Fahnenmodell [2] bzw. im Falle von windschwachen Wetterlagen und Windstillen nach dem Le QUINIO´schen Puff-Modell [3].

Folgende ausgewählte, wichtige Einflußgrößen des Ingestionsdosismodells wurden hinsichtlich ihrer statistischen Eigenschaften eingehender untersucht: (1) die Ausbreitungs (χ)- und Ablagerungsfaktoren (F,W) zur Bestimmung der Kontamination des Bodens und Bewuchses, (2) der Transferfaktor Boden-Pflanze und (3) die Ernährungsgewohnheiten von Erwachsenen. Für die übrigen Parameter des Ingestionsdosismodells, z.B. die Dosisfaktoren und bzgl. des Nuklidverhaltens im Boden, wurden hingegen vereinfachend die festen Referenzwerte - also keine Verteilungen - nach [1] zugrunde gelegt.

ERGEBNISSE

Tab. 1 enthält eine Zusammenstellung ausgewählter statistischer Kenngrößen der berechneten Dosisverteilung. Die Werte geben die Ingestionsdosis pro freigesetzter Aktivitätsmenge D/Q an einem festen Aufpunkt in der Umgebung des Emittenten an. Zum Vergleich sind auch die nach der deterministischen Berechnungsvorschrift [1] ermittelten $(D/Q)_{SSK}$-Werte angegeben.

Die Streubreite der durch einmalige, kurzzeitige Aktivitätsfreisetzungen bedingten potentiellen Ingestionsdosen ist nach Tab. 1 erheblich und beträgt aufgrund der Variabilität der im Ingestionsdosismodell involvierten Prozesse, Verhaltensweisen etc. etwa 2-4 Größenordnungen. Sie wird dominiert durch die zum Freisetzungszeitpunkt (zufällig) vorherrschenden meteorologischen Bedingungen sowie die individuell unterschiedlichen Ernährungsgewohnheiten. Weiterhin ist bemerkenswert, daß die nach der statistischen Methode berechneten D/Q-Werte, basierend auf der Simulation von 10000 Emissionssituationen, die $(D/Q)_{SSK}$-Werte in keinem Fall überschreiten.

BEWERTUNGEN VON KURZZEITIGEN AKTIVITÄTSFREISETZUNGEN

Als quantitative Beurteilungsgrundlage zur Festlegung von störfallbedingten Emissionsrichtwerten können folgende Verfahrensweisen herangezogen werden: 1. eine sich am Risikobegriff orientierende Methode und 2. eine perzentilenorientierte Bewertung.

Bewertungsgröße des risikoorientierten Konzeptes ist in Anlehnung an den klassischen Risikobegriff das Produkt aus Störfall- oder Schadensausmaß und Eintrittshäufigkeit. Betrachtet man z.B. die normierte Strahlenexposition D/Q als Schadensmaß, so erhält man über die Beziehung

$$D_G = Q \int_0^\infty w(D/Q)\ (D/Q)\ d(D/Q) = Q\ (\overline{D/Q})$$

eine unmittelbare Verknüpfung zur Festlegung eines Emissionsrichtwertes (Q) mittels eines Dosisgrenzwertes (D_G) und einer statistisch abgeleiteten Risikomaßzahl ($\overline{D/Q}$) (vgl. Tab. 1).

Tab. 1: Statistische Kenngrößen der berechneten Ingestionsdosishäufigkeitsverteilung D/Q (pSv/Bq) an einem festen Aufpunkt (Freisetzungshöhe H = 100 m, Abstand vom Emittenten 500 m)

	J 131/Schilddrüse		Sr 90/Knochen	
Emissionsdauer T_E(h)	1	24	1	24
$\overline{D/Q}$	0,0012	0,0010	0,062	0,041
CV(%)	552	192	1133	371
$(D/Q)_{99}$	0,027	0,009	1,1	0,68
$(D/Q)_{99,9}$	0,086	0,016	10	1,8
$(D/Q)_{max}$	0,14	0,027	26	3,5
$(D/Q)_{SSK}$	0,30	0,24	34	14

$\overline{D/Q}$	Erwartungswert der Verteilung
CV	Variabilitätskoeffizient
$(D/Q)_n$	n-Perzentile der Verteilung
$(D/Q)_{max}$	Maximaler (D/Q)-Wert bei 10000 Simulationen

Bei der Zugrundelegung des perzentilenorientierten Konzeptes, bei dem äußerst selten auftretende Geschehensabläufe als praktisch bedeutungslos angesehen werden, erfolgt die Festlegung von Emissionsrichtwerten nach der Beziehung

$$D_G = Q \cdot (D/Q)_n$$

wobei $(D/Q)_n$ der Wert der normierten Dosisverteilung (oder eines anderen Schadensmaßes) ist, die n-Prozent aller Ereignisabläufe umfaßt.

Tab. 2: Anwendung der verschiedenen Konzepte zur Festlegung von störfallbedingten Emissionsrichtwerten in GBq

	J 131/Schilddrüse		Sr 90/Knochen	
Emissionsdauer T_E(h)	1	24	1	24
	Risikoorientiertes Konzept (D_G= 15 mSv)			
	—— 13800 ——		—— 296 ——	
(100-n)	Perzentilenorientiertes Konzept (D_G = 50 mSv)			
0,1 %	555	3070	4,8	27
1 %	1850	5440	44	74
5 %	13200	10800	-	229
	Deterministische Vergleichswerte (D_G = 50 mSv)			
	166	210	1,5	3,3

In Tab. 2 sind als Anwendungsbeispiel die nach den beiden vorgenannten Bewertungskonzepten bestimmten Emissionsrichtwerte (Q) angegeben. Dabei wurden verschiedene Dosisgrenzwerte bzw. Überschreitungshäufigkeiten (100-n)% des Grenzwertes zugrunde gelegt. Ein Vergleich mit den nach dem deterministischen Verfahren bestimmten Emissionsrichtwerten läßt erkennen, daß letztere nach den vorliegenden Modellrechnungen zu deutlich geringeren Abgabewerten führen. Ihre Anwendung stellt daher möglicherweise eine zu restriktive Beurteilungsgrundlage für störfallbedingte Freisetzungen dar.

SCHLUSSBEMERKUNGEN

Als Ergebnis zeigt sich, daß eine statistische Störfallfolgenanalyse sowohl von der Methodik als auch vom Aufwand her durchführbar ist. Beschränkungen ergeben sich möglicherweise in solchen Fällen, in denen der für eine statistische Untersuchung erforderliche Informationsstand aus der Literatur nicht unmittelbar bereitgestellt werden kann. Weiterhin ist der statistische Ansatz geeignet, die Beurteilungsgrundlage zur Bewertung von störfallbedingten (oder sonstigen kurzzeitigen) Radionuklidfreisetzungen zu objektivieren und damit auch den bisherigen kritischen Vorbehalten entgegenzuwirken.

LITERATURVERZEICHNIS

1. Empfehlungen der Reaktorsicherheits- und Strahlenschutzkommission: Leitlinie zur Beurteilung der Auslegung von Kernkraftwerken mit DWR gegen Störfälle (Entwurf), Geschäftsstelle der Reaktorsicherheitskommission, Köln 1982
2. Geiß, H., Nester, K., Thomas, P. und Vogt, K.J.: In der Bundesrepublik Deutschland experimentell ermittelte Ausbreitungsparameter für 100 m Emissionshöhe, JÜL-1707 (KfK-3095), Febr. 1981
3. Doury, A., Gerard, R. und Picol, M.: Abaques d´evaluation directe des transferts atmospheriques d´effluents gazeux (Deutsche Übersetzung), Rapport DSN No. 84 (Rev.1), Mars 1977

DOSE ASSESSMENT IN INDUSTRIAL RADIOGRAPHY INCIDENTS WHICH CAUSE BURNS

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1. Introduction

Two accidents which resulted in finger burns to industrial radiographers who touched source capsules are described and illustrated and the methods used to assess the radiation doses to the workers are discussed. Exposure rates measured near ^{192}iridium capsules are presented. The importance of careful investigation and follow-up is stressed and the psychological impact on the workers is explored. The lessons learned and recommendations for dealing with such incidents are given.

2. Background

Alberta, in common with other oil and gas producing areas, supports a very active gamma radiography industry. Although industrial radiographers in Alberta have a higher average radiation dose than any other radiation group,[1] the number of mishaps resulting in somatic injury is small. Canada has had three such cases in 24 years and the dosimetric aspects of two of these are explored. It would appear to be a simple matter to assign a dose to a worker where a radiation monitor has been worn, but this rarely proves to be the case when accidents occur. It is, however, important to assess the doses carefully, to determine whether prompt medical treatment and legal action are required, and to satisfy the worker's curiosity as to the eventual outcome.

3. Case I

(i) Accident Details:

An untrained worker released a pigtail iridium capsule of 44 Ci of ^{192}Ir from its shield and with the fingers of his left hand in direct contact with the capsule re-inserted it into the container. Five days later his fingers became tender and discoloured and after a further 4/5 days blisters appeared. He was examined after three weeks when the first pictures were taken. The areas of severest damage were dry, surrounded by superficial tissue which was sloughing off, and these zones were circumscribed by areas of reddened skin with sharply defined edges. The injuries appeared to heal after six months but after 18 months the fingers became acutely sensitive to heat and cold, demonstrating ulceration and telangiectasis. At 22 months amputation was considered by the worker but he was pursuaded to see a plastic surgeon who removed tissue from the painful areas and performed grafts at 29 months. The last photograph shows the worst of the fingers at 40 months. Recent contact with this case showed that his fingers were still satisfactory, apart from susceptibility to heat and cold. There had been no loss of flexion in the fingers and the hair growth on the dorsal surfaces of the fingers was normal.

(ii) Dose Assessment:

It was difficult for the worker to recall the duration of his contact with the source, but by repeating the operation using a dummy source it was shown to take

from 3 to 5 seconds. This was used in several methods to assess the finger dose. By using the specific gamma ray constant for ^{192}Ir the dose was estimated to be between 5.5 and 9.2 grays. An estimate using the absorbed fraction of gamma energy deposited in a finite volume of tissue gave a dose of 5.1 to 8.5 Gy. Extrapolating the monitor result of 14 mGy gave a dose between 15.7 and 26.2 Gy. Some source contact measurements were made and these indicated a dose between 9.7 and 16.2 Gy, and by using the N.C.R.P. contact exposure rates[2] from 15.6 to 26.1 Gy was obtained. In addition the finger injuries were compared with previous cases and this showed the dose to be at least 15.0 Gy up to possibly 40.0 Gy.[3,4,5] Details of these figures are found in table I. The finger dose appeared to lie somewhere between 5 and 40 Gy. For this first case an assessment from a radiotherapist was not obtained.

METHOD	ABSORBED DOSE (grays)
Specific gamma-ray constant	5.5 - 9.2
Absorbed fraction (to 1 g)	5.1 - 8.2
From film monitor	15.7 - 26.2
From measure exposure rates	9.7 - 16.2
N.C.R.P. exposure rates	15.7 - 26.1
Comparison to other cases	>15.0 - 40.0

TABLE I: CASE I FINGER DOSE

Table II gives exposure rates in the vicinity of ^{192}Ir radiography capsules measured using TLDs.

LOCATION	EXPOSURE RATE R/s Ci
Source contact (side)*	7.5
Source contact (end)*	5.2
Guide tube contact **	1.0
1 metre from guide tube	0.4

* 0.5 mm steel wall
** 3.9 mm steel wall

TABLE II: EXPOSURE RATES MEASURED NEAR ^{192}IRIDIUM RADIOGRAPHY CAPSULES

4. Case II

(i) Accident Details:

An untrained worker ejected a pigtail source from its shield to the exposed position where it remained for about two hours while he examined 30 pipeline welds. Between welds he carried the shield cradled in his arms and during the setting up for each weld handled the source guide with his fingers about 5 mm from a capsule of 35 Ci of iridium-192.

His fingers became sore after ten days. Photographs were taken after 30, 33 and 38 days. At three months complete healing was reported, but at 15 months the finger with the highest dose was sore and dry, and the following month the end was amputated.

(ii) Dose Assessment:

Accident simulation revealed how the employee had manipulated and moved the equipment. Although the total involvement with the equipment was from 2 to 3 hours, it was not possible to estimate the time that the fingers were close to the source. The dose to the worker's TLD monitor, worn in the hip pocket, was 0.93 Gy. A number of measurements were made in the vicinity of the source and equipment and the exposure rate at the point where the fingers were closest to the source was 36 roentgens per second. This information would have been useful if times had been known.

Shortly after the finger injuries this man was examined by a radiotherapist who estimated that the index fingers had absorbed 20 and more than 50 Gy, respectively, and his thumbs had absorbed 10.0 Gy.

METHOD	ABSORBED DOSE (grays)
Radiotherapist's opinion	Thumbs 10 Lt. index 20 Rt. index >50

TABLE III: CASE II FINGER DOSE

5. Comments

The various methods used to estimate finger dose in these two incidents show that where times of involvement are well known, the dose can be fairly realistically estimated. Where times are not known, one of the best methods of dose assessment is by experienced radiotherapist's assessment from the damage observed.

The legal outcome of both incidents was similar. In keeping with Canadian practice, a decision was made by physicians of two separate radiation protection agencies that the victims would be prohibited for life from working as atomic radiation workers. For the second case they also made a strong recommendation that the worker protect his fingers from cold and physical injury. Unfortunately he ignored this advice and worked outside in winter at a rough labouring job. It was thought that this aggravated the injury to his finger and that the amputation might have been avoided. After the first case regulations requiring the licensing of industrial radiographers based on proof of medical fitness and radiation safety knowledge were put into effect in Alberta.

Copies of photographs and reports were supplied to the workers, their physicians, and the local compensation board and both workers gave written permission for their photographs to be used for training purposes.

Although the incidents themselves were comparable, it is of interest to contrast the effect they had on the workers. Unfortunately some of the examinations of both took place at a cancer clinic, and this in itself had a marked impact on the men. The first worker was initially very apprehensive about the radiation damage he could see on his fingers, but was reassured after he learned that the worst outcome might be loss of his hand. He then became concerned that other workers would get into the same difficulties. He suggested that the mishap could be used to train others to work safely. The second case also was very fearful, and has remained so. He has become isolated and a recluse, thinking he is not quite acceptable to others. Attempts to contact him recently have been unsuccessful, but

a relative reports that he is very depressed. The first case has been glad of the ongoing contact. The experience confirms that of others:[6] that injured radiographers are not interested in becoming scientific curios, but do seek an honest and simple appraisal of the possible consequences of their injuries.

6. Lessons

There are some important lessons which have been learned from dealing with these accidents:

1. Appropriate medical supervision of burned fingers, and personal care by the individual will ensure optimum outcome.
2. The initial dose to the injured tissue should be accurate enough to enable sound medical and legal decisions to be made. However, the opinion of an experienced radiotherapist should be sought rather than spending a great deal of time on mathematical calculations.
3. A medical photographer such as is generally employed by a large hospital should be used to obtain a photographic record of the injuries.
4. An attempt should be made to obtain written permission from the worker to use photographs of injuries for training purposes.
5. Although most of the industrial radiography performed in Alberta makes use of free source capsules which are driven from and returned to their shields by air pressure, the most serious overexposures to workers have resulted from the use of pigtail sources.

7. Conclusion

Various methods were used to estimate finger dose in these two incidents. It was initially thought that the calculation of a finger dose, to the first worker, would be a simple matter. However, the range of doses, resulting from the various methods used, made prognosis difficult.

In the second case, where only a total time of involvement was known, detailed dose calculations were not possible. However, the opinion of a well-experience radiotherapist to assign specific doses to the worker's fingers, was thought to be more useful, and less time consuming.

8. References

1. Occupational Radiation Exposures in Canada - 1980, 82-EDH-79; Health and Welfare Canada, Ottawa, 1981.
2. National Council on Radiation Protection & Measurements, Protection against radiation from brachytherapy sources, NCRP Report 34, Washington 1972.
3. Maxfield, W.S. and Porter, G.H., Accidental radiation exposure from ^{192}iridium camera, in Handling of Radiation Accidents, IAEA Symp. Proc. STI/PUB/229, Vienna, 1969.
4. Saenger, E.L., Medical Aspects of Radiation Accidents, USAEC, Washington D.C., 1963.
5. Blatz, H., ed., Radiation Hygiene Handbook, McGraw-Hill, New York, 1959.
6. Bunin, S.M., Psychological aspects of acute radiation accidents, in Handling of Radiation Accidents, IAEA Symp. Proc. STI/PUB/229, Vienna, 1969.

DOSE PROJECTION CONSIDERATIONS FOR EMERGENCY CONDITIONS AT NUCLEAR POWER PLANTS

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In recent years, the objectives and scope of dose projection considerations for emergency conditions at nuclear power reactor facilities have changed dramatically. Previously, dose projection procedures were a major basis for categorizing reactor emergencies.

Presently, the condition of the major reactor safety systems has become the major basis for categorizing the degree of seriousness of a reactor accident. Operating organizations have implemented emergency action levels which allow classification of a wider range of accident sequence types. The primary role of offsite dosimetry is to determine the extent of required protective actions or to support an analysis that no action is required. Preplanned offsite protective actions may obviate the need to complete dosimetry calculations prior to recommending actions to offsite government officials.

Nevertheless, regulatory guidance for offsite dosimetry under accident conditions stresses meteorological forecasts, rapid data collection and analysis, and explicit representation of terrain effects. Current guidance also emphasizes calculations to 16 or 80 km distances, consideration of ingestion doses pathways, and a capability to simulate short-term variations in wind trajectory, wind speed, and diffusion class.

There are some technical difficulties associated with these advances. Lapse rate measurements are recognized to be potentially inaccurate for some daytime conditions if the data averaging period is shorter than one hour. The accuracy and precision of atmospheric dispersion estimates are, in general, degraded as the time interval of for the analysis is reduced.

Emphasis on rapid release scenarios requires consideration of short-lived fission products as well as competence by all shift staffs in completing dosimetry procedures. Emergency dosimetry analysts must also be able to consider the expected duration of releases and atmospheric conditions in order to estimate projected dose integrals. The time required to attain protective response dose limits must be compared to the time required to notify, warn, or evacuate members of the general public.

These changes of emphasis are reflected in two dosimetry computer codes that have recently been developed for the U.S. Nuclear Regulatory Commission (NRC): IRDAM (NUREG/CR-3012) and MESORAD (PNL-4753). The former was designed as a rapid, versatile, user-friendly model that will operate with preliminary source term estimates or refined data, as the situation demands. IRDAM requires minimal user training and contains extensive programmed default values of important parameters. IRDAM also

supports the NRC's need for rapid emergency deployment to reactor sites. The program is implemented on a portable microcomputer that is also programmed to emulate a terminal of the more sophisticated MESORAD system.

MESORAD is, in essence, a variant of the MESOI puff-adversion dispersion code that has been enhanced extensively by addition of dosimetry subroutines. MESORAD normally requires a meteorologist and a dosimetrist for operation. An associated source term code (TACT) requires a nuclear engineer, in addition. The computer environment, dual Data General MV-6000 processors, requires a system manager. MESORAD is intended for use in a national or regional emergency operations centers, or a well established site team, by a cadre of trained analysts.

Source Term Estimates

IRDAM provides source term estimates by allowing the user to input isotopic release rates, gross release rates, or to specify a containment leakage problem. The containment leakage options provide estimates of source term according to the equilibrium core inventory, estimated or default leakage rates, and default core release fractions for fuel melt, void space activity, and coolant concentration. The program includes a typical relationship between the containment area radiation monitor and airborne radioactivity concentration for the major types of containments.

MESORAD has no source term capabilities and requires a separate code, TACT, for source term calculations. TACT contains default core release fractions for coolant, void space, and fuel matrix activity. In addition, it allows the analyst to model transfer rates, filtration efficiencies, plateout rates, source quantity, concentration and dose rates for a network of user-defined modes. Isotopic release rates are stored in a data base for later reference or as input to MESORAD. The source term capabilities of TACT greatly exceed those of IRDAM. However, both codes allow estimates of source terms based on descriptions of reactor safety system failures and therefore support the need to calculate projected releases and doses. These codes also assist the emergency response analyst in modifying preplanned protective action recommendations according to actual conditions of time of release, filtration effectiveness, meteorological conditions, etc. The major difference between these models is that IRDAM is a Class A model, as defined in NUREG-0654, while TACT/MESORAD is a Class B model. These distinctions are further discussed in NUREG/CR-3011.

Atmospheric Dispersion

Enhanced dispersion analysis is absent from IRDAM because the need for implementation on a microcomputer precludes inclusion of extensive site specific data bases and because meteorological forecasts are assumed to be unavailable in the short time period where IRDAM would be effective. In contrast, MESORAD accommodates files of observed and forecast meteorological parameters for 30 measurement locations for periods as short as 15 minutes. Site specific data bases contain topographic information. A subroutine simulates the effect of topography on the wind filed vectors. MESORAD requires the user to explicitly specify the stability classification.

Several alternative parameterizations of the diffusion coefficients are contained in the code. IRDAM classifies the dispersion conditions according to lapse rate, horizontal variance in the wind vector, or the user's specification.

MESORAD calculates dispersion data by simulating the transport and diffusion of individuals puffs of released material and summing the current and cumulative contributions of each puff during each 15 minute time step. IRDAM contains tables of Xu/Q at fixed distances (0.5, 1, 2, 3, 8, 20 km). These tables are reasonably accurate if the meteorological conditions are constant during the period of the release and downwind transport. Distances closer than 0.5 km are not included because IRDAM does not model building wake effects. The data necessary for building wake correction is not presumed to be available immediately following an accident and the inclusion of wake effects increases the necessary training level of the dosimetry analyst.

Also building wake corrections are not significant at distances of 0.5 km and beyond.

Dosimetry

MESORAD provides the user with a plethora of output options: relative concentration, daughter concentration, depleted plume, thyroid dose, lung dose, total whole body dose, external plume dose, deposition dose, inhalation dose, (ICRP 26/30 method), and semi infinite cloud (SIC). All options include current dose rate or cumulative dose and tabular or graphic output. There is a total 51 output options. IRDAM calculates thyroid and whole body dose rates and projected doses.

External irradiation from plumes of radioactive material is modeled in MESORAD by averaging the plume concentration over finite volume elements and summing the contribution from each element. The accuracy of this method compares favorably with the rigorous point kernel integration method and the amount of calculational processing is significantly lower. This approach allows real-time analysis of short-term variations in source term or meteorological conditions while maintaining the accuracy of a finite plume simulation model. MESORAD performs finite plume calculations for elevated or ground level releases, unless the user specifices the SIC approximation. The SIC method is not necessarily conservative when a puff advection dispersion model calculates concentrations because the matrix elements that are specified as locations for dosimetry calculations do not, in general, lie along the centerlines of each puff's trajectory.

IRDAM, by comparison, calculates plume exposure from ground level releases according to the SIC model. The code's documentation includes a table of finite plume correction factors for a range of photon energies and stability dosses. Matrices of this type are difficult to program into a computer code because the effective energy of the released photons is a function of the emitter-receptor geometry and because the photon energy frequency distribution changes with time as short-lived radionuclides decay. The effort required to properly implement a finite plume correction is equivalent to the effort expended in developing the finite volume element method in MESORAD.

In the case of elevated releases, IRDAM incorporates a unique algorithm which is sufficiently simple and rapid to permit elevated plume calculations on a microcomputer. This algorithm approximates a Gaussian plume with two concentric cylinders. The radii are equal to σ_y and $2\sigma_y$, respectively. The concentrations of the center cylinder and the outer annulus are approximately Gaussian. The photon energy distributions of the released radionuclides are collapsed to three energy groups that are weighted according to the attention of 50m of air. The results are generally within a factor of two of codes such as WRAITH, which contain rigorous point kernel integration routines.

Dispersion estimates are limited to an accuracy of a factor of two or three by the stochastic nature of atmospheric turbulence. Hence, the limitations of IRDAM's simple dosimetry approximations do not constrain the ability of the code to estimate dose projections under accident conditions.

MESORAD accommodates rapid changes in composition of released plumes by estimating physical decay, daughter radionuclide ingrowth, deposition, washout, and plume depletion. The code calculates the concentration profile and resulting radiation doses are for each radionuclide in each puff.

IRDAM monitors the isotopic composition of the plume for elevated releases or when the exact mixture is specific as an input. Otherwise, the release is modeled to consist exclusively of a mixture of Xe-133 and I-131 when the age of the released material is greater than 24 hours. For more rapid releases, a time-dependent correction factor modifies the dose conversion factors of Xe-133 and I-131 to values that represent the appropriate mixtures of noble gases and iodines.

Both models convert calculated concentrations of radionuclides to environmental dose equivalent or dose commitment uptake rates. The U.S. Environmental Protection Agency, however, has issued radiological protective action guidance (PAG) in the form of projected dose integrals. MESORAD calculates projected doses by carrying the simulation forward in time.

IRDAM calculates the product of the dose rate and the release duration at each distance. The default duration is 8 hours, which is a reasonable approximation of the 99th percentile of the wind direction persistence distribution.

The amount of time required to achieve an integrated dose equivalent equal to a PAG at a given location is an important factor which influences protective action decision making. Using MESORAD, the emergency response analyst may project doses using forecast meteorology and releases, observing the time at which PAGs or other administrative limits are attained at locations of interest. IRDAM produces a table of transport times which, in conjunction with the dose rate data, allows the analyst to calculate the amout of time corresponding to the accumulation of the PAG or administrative limit.

ENVIRONMENTAL AND HEALTH CONSEQUENCES OF AN ACCIDENTAL ATMOSPHERIC RELEASE OF RADIOACTIVE MATERIAL.

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INTRODUCTION

Malgré leur faible probabilité, il est nécessaire de connaître les conséquences dans l'espace et dans le temps des rejets accidentels de substances toxiques. Cette connaissance permet d'évaluer les risques associés au transport de ces produits et l'efficacité des mesures destinées à réduire ces effets directs ou indirects sur l'homme : évacuation, décontamination. Il existe déjà de nombreux modèles généralement informatisés d'évaluation des conséquences d'un rejet atmosphérique accidentel : c'est le cas en particulier des modèles adaptés au cas d'un site d'installation nucléaire.

Le présent modèle est plus spécialement adapté aux problèmes posés par les accidents de transport. Ceux-ci se caractérisent par le fait que ni la densité de population, ni la nature et les risques du produit transportés ne sont connus d'avance. En conséquence, ces données font partie des entrées du modèle, ainsi que le choix des isodoses qui peuvent être pertinentes, et un bloc "dosimétrie" a été introduit dans le modèle pour relier dose individuelle et concentration intégrée relative au produit relâché. La souplesse et la facilité d'emploi ont été privilégiées dans ce contexte où la précision des résultats est, de toute façon, faible. (Aux problèmes généraux de la modélisation du transfert atmosphérique s'ajoute le fait que les paramètres en sont mal connus dans le cas d'accidents de transport). C'est pourquoi le modèle est conversationnel et écrit en A.P.L.

Le programme réalisé calcule les concentrations intégrées, les équivalents de doses inhalés individuels atteints en un point quelconque du passage de la bouffée. Il fournit la répartition géographique de la dose collective ainsi que le dépot éventuel. Il trace les courbes isoquantes des doses et des dépôts à une échelle cartographique. Il permet l'estimation des effets aigus ou retardés dus à l'inhalation. La mise au point de cet outil est par ailleurs faite avec en vue une première application pratique au cas du transport d'oxyde de plutonium /1/. D'où les références aux résultats de cette application dans l'exposé.

I - PRESENTATION DU MODELE

Le schéma suivant présente l'enchaînement des étapes de calcul des conséquences sanitaires d'un relâchement atmosphérique accidentel, mettant ainsi en évidence les paramètres d'entrée et les quantités calculées à chaque étape /1/.

a) Le "modèle dosimétrique" choisi tient compte de la composition isotopique du mélange. Il utilise les résultats du modèle de la C.I.P.R. /2/et donne les équivalents de dose reçus après absorption du produit. Les effets à long terme se déduisent de la dose collective grâce à la valeur recommandée par la C.I.P.R. (publication 26) : $1.65\ 10^{-2}\ Sv^{-1}$, sous l'hypothèse fondamentale de linéarité de la loi dose-effet. Par contre les effets aigus sont à seuil et s'observent dès que la dose individuelle a dépassé une certaine valeur correspondant à une dose engagée importante (de l'ordre de 50 Sv dans le cas du Pu d'après /3/).

b) Le "modèle de dispersion" adopté pour l'évaluation des concentrations atmosphériques est du type panache bigaussien classique donnant les concentrations intégrées sur le temps au niveau du sol. Sont également introduits dans ce modèle les réflexions sur le sol et la

couche d'inversion de température. Il prend également en compte la hauteur du rejet et les phénomènes de dépôt sec et humide /4/. Le calcul des surfaces est réalisé pour chaque niveau de dose choisi ce qui permet ensuite d'évaluer les doses collectives ou les quantités déposées au sol le pas de cette intégration correspond au découpage retenu pour les niveaux de doses. Il est à noter que ces surfaces peuvent être calculées pour des hauteurs de rejets quelconques et que le pas de cette intégration correspond au découpage retenu pour les niveaux de dose.

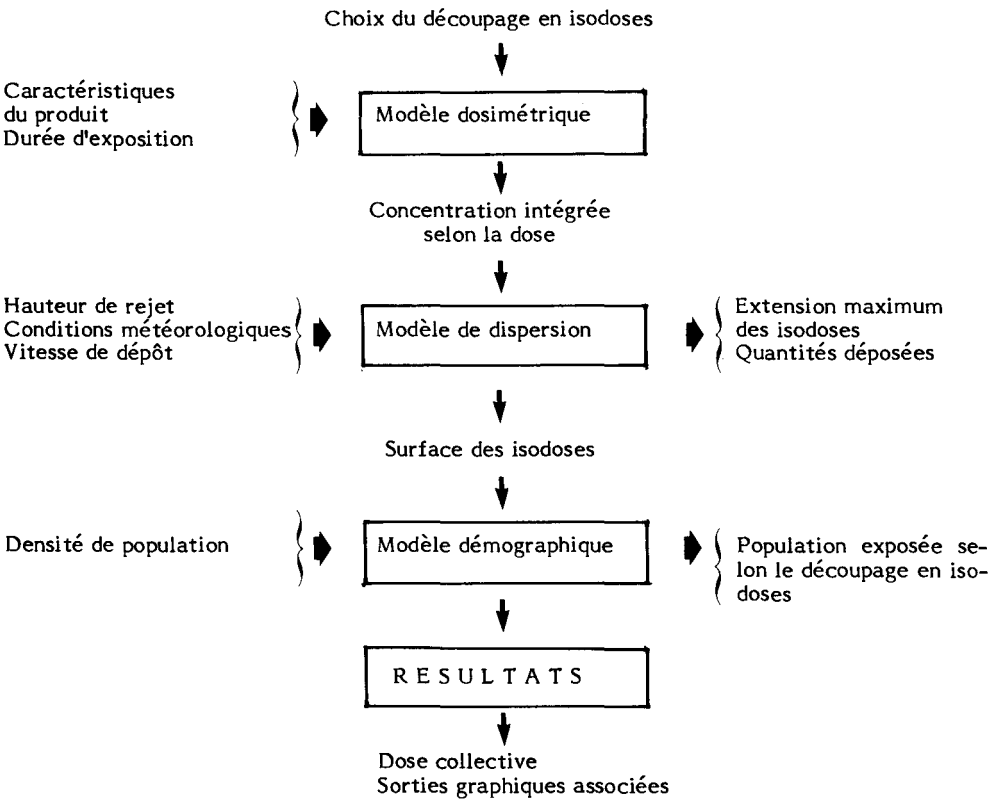

Figure 1 : Les étapes du modèle général de calcul des conséquences d'un rejet atmosphérique.

c) Le "modèle démographique" reprend une typologie des densités de population le long d'un itinéraire déjà effectué /5/ et qui a permis de retenir 3 types de densité suivant le lieu. Sont ainsi différenciées les zones urbaine, suburbaine, rurale dont les densités suivantes peuvent être considérées comme respectivement représentatives : 6 800, 700, 50 (hab/km^2) .

II - EXEMPLE DE RESULTATS

Une première application du modèle a été effectuée dans /1/. Les tableaux qui suivent présentent quelques résultats pour des situations accidentelles entraînant la mise en suspension de 1 à 100 g d'oxyde de plutonium en zone urbaine. On présente ici deux résultats correspondant à un rejet effectué à hauteur nulle dans deux conditions météorologiques. La première est une diffusion faible avec un vent de 1 m/s (DF1, H = 0), la deuxième est une

diffusion normale avec un vent de 1 m/s (DN1, H = 0). Il est à noter qu'une hauteur de rejet différente (10 m) conduit pratiquement au même résultat dans le cas de la seconde condition météorologique. Le tableau 1 indique pour des niveaux de dose donnés les surfaces et les quantités déposées correspondantes (vitesse de dépôt de 5 mm/s).

[mSv]	FRACTION DEPOSEE DF 1,H=0 (*)	DN1,H=0	SURFACE CONTAMINEE [m^2] DF1,H=0	DN1,H=0
H ≥ 500	0.18	0.00	2.9E2	0.0E0
H ≥ 100	0.27	0.18	2.1E4	3.0E3
H ≥ 50	0.31	0.21	4.7E4	6.8E3
H ≥ 10	0.44	0.34	3.1E5	4.3E4
H ≥ 5	0.5	0.39	6.7E5	9.5E4
H ≥ 1	0.64	0.52	3.7E6	6.1E5
H ≥ 0.5	0.70	0.58	7.2E6	1.4E6
H ≥ 0.1	0.81	0.75	3.2E7	9.1E6

(*) DN1 = diffusion normale, vent de 1 m/s ;
DF = diffusion faible

Tableau 1 : Quantité déposée et surface affectée à la suite d'un rejet de 1 g PuO_2.

	Choix du domaine de calcul	Conditions de rejet (H = 0) (*)	Population exposée	Nombre d'effets sanitaires
Effets sans seuil	≥ 0.1 mSv	1g (*)	220 000	1.8
	≥ 0.1 mSv	1 g (**)	62 000	0.3
	≥ 10 mSv	100 g (*)	220 000	180.0
	≥ 10 mSv	100 g (**)	62 000	30.0
Effets aigus	≥ 50 Sv	1 g (*)	0	0
	≥ 50 Sv	1 g (**)	0	0
	≥ 50 Sv	100 g (*)	20	20
	≥ 50 Sv	100 g (**)	0	0
Limite annuelle travailleur	≥ 0.05 Sv	1 g (*)	320	0.5
	≥ 0.05 Sv	1 g (**)	50	0.05
	≥ 0.05 Sv	100 g (*)	50 000	150
	≥ 0.05 Sv	100 g (**)	9 000	22

(*) Diffusion faible, vent 1 m/s
(**) Diffusion normale, vent de 1 m/s

Tableau 2 : Effets sanitaires d'un rejet accidentel

CONCLUSION

Le modèle présenté ici comporte outre le type de résultats présentés ci-dessus, la possibilité de sortir à chaque instant de l'histoire de l'accident l'histogramme des doses individuelles engagées (c'est-à-dire à l'intérieur d'un rayon donné autour du lieu de l'accident), ainsi que le dessin des isodoses à une échelle cartographique. Ces paramètres sont importants pour décider de la conduite à tenir (modalités d'une évacuation éventuelle ou d'un confinement). Tout ou partie de ces possibilités sont obtenues grâce à la mise en oeuvre d'un programme interactif qui facilite la modification des paramètres d'entrée aussi bien pour juger de la sensibilité des résultats aux différentes hypothèses que pour évaluer l'efficacité de mesures éventuelles destinées à réduire les conséquences de l'accident. La grande souplesse du langage choisi (APL) qui permet en particulier d'extraire aisément des résultats intermédiaires en cours d'exécution, est un atout précieux pour faire face à l'hétérogénéité des situations envisageables pour les accidents de transport. De fait l'application du modèle a été aisément étendue à certains cas d'accidents susceptibles d'entraîner le relâchement d'autres substances toxiques dans l'atmosphère.

REFERENCES

/1/ F. RANCILLAC, Evaluation des risques sanitaires associés aux accidents de transport de plutonium. Rapport C.E.P.N. n° 66, mars 1983.

/2/ C.I.P.R., Limits for intakes of radionuclides by workers, Rapport du Comité 2 de la Commission Internationale de Protection Radiologique, I.C.R.P. 30, Pergamon Press, Oxford, 1980.

/3/ M. METIVIER, Devenir biologique et toxicité du plutonium, in P. Galle Editeur, "Toxiques nucléaires", Masson, Paris, 1982.

/4/ CEA-NRPB, Méthodologie pour l'évaluation des conséquences radiologiques des rejets d'effluents radioactifs en fonctionnement normal. Rapport CEA-NRPB, Commission des communautés européennes, juillet 1979.

SOURCES OF ERROR IN THE DETERMINATION OF THE RADIATION EXPOSURE OF A POPULATION DUE TO DIAGNOSTIC X-RAY EXAMINATIONS

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In the industrialized countries the great majority of the exposure of the population to "artificial" radiation stems from the use of X-ray in medical diagnosis. Because of the presence of scattered radiation during examinations,practically every organ is subjected to a certain radiation dose. Therefore, every X-ray examination potentially involves a genetic and a somatic risk. The first one can be assessed by determining the corresponding doses to the gonads. The estimation of the somatic burden is more difficult. The yardstick usually adopted is the mean dose in the active bone marrow. By means of sophisticated phantom measurements or by means of Monte Carlo calculations, it has now become possible to state the relevant dosage figures involved in many different types of X-ray examination (s. (1) as an example). Using these figures any physician considering a particular examination can form an idea of the corresponding radiation burden.

In order to judge the radiological activity of a country from the point of view of radiation protection it is necessary to deduce useful reference values. Beside the dosage figures mentioned above these have also to take into account the frequency of examination. Already in 1958 (2) concepts of "average annual genetically significant dose" (GSD) and "average annual bone marrow dose" (BMD) were introduced. Both values have the character of an index. A comparision of index values with earlier values, or with those in other countries, enables sometime important radiation-protective conclusions to be drawn.

In 1978 the Department of Medical Radiation Physics of the University of Berne established these indices for Switzerland for the second time. The results are given in Table 1

	1971 (mrad)	1978
GSD	19	23
BMD	63	63

Table 1 : Average annual genetically significant gonadal doses and average annual bone marrow doses for the Swiss population in 1971 and 1978

Since these values are basic figures to many discussions on radiation protection it seemed to us important to check their reliability. This depends on the accuracy of the measurements of the doses to the gonads and to the bone marrow and on the reliability of the estimated number of examinations per year.

Determination of the Resultant Gonadal and Bone Marrow Doses per Examinations.

For the phantom measurements mentioned above a sophisticated measuring technique employing TL detectors was developed. An accuracy of within $\pm$ 4% was achieved. Therefore the real difficulty was not the measurements itself, but laid in the correct interpretation.

For 17 types of radiological examinations a "standard man" phantom was irradiated with X-ray unit parameter (KV, SSD, film size) corresponding to approximately the mean value to the parameter reported to us by some 200 swiss doctors. The bone marrow dose (BMD, in mrad) per skeletal region was derived by an often laborious method from the data obtained with 55 TLD placed in the different bones. The doses derived apply, as the "adult phantom", to the irradiation parameters set in our X-ray unit. A mean bone marrow dose per examination for the parameters to which the X-ray units are actually set by the swiss doctors was determined using a computer program.

The gonadal doses, which are derived in effect from dose measurement at one point, are much more subject to variation due to anatomical position or equipment settings than the corresponding bone marrow doses. The average location of the female gonads is not well-known. It is therefore not always clear whether these lay within the radiation field in case of an examination. The resulting gonadal doses can vary thereon by orders of magnitude. For this reason we always measured the ovaries dose in the three different locations well-known from reference works.

The Statistical Determination of the Frequency of Examination

As early as 1971 for the first enquiry, a large number of doctors were questioned about their X-ray work. A detailed analysis of this data showed that it would be possible in any future enquiry, without significantly impairing accuracy, to greatly reduce the number of examination types investigated and - due to a suitable division of the doctors interviewed into various sampling strata - also the number of enquiries themselves. Letters were sent out beforehand to about 200 doctors asking them to describe their examination procedures with particular emphasis on the field setting as well as on their equipment parameters. If one is prepared to accept a statistical uncertainty of 10 %, the time expenditure involved in such an enquiry can be kept down to a reasonable amount.

Summary and Conclusions

Although it was not possible to measure directly many influential factors, their effect on the variations in the resulting average annual GSDs and BMDs can be estimated.

	GSD	BMD
Accuracy of measurement	± 4%	± 2%
Simulation of the field and equipment setting	± 12%	± 8%
Localisation of the organs	± 15%	± 5%

Estimated average variation in the determination of gonadal and bone marrow doses per examination: as already mentioned, the above estimations show that the determination of the gonadal doses is liable to greater error than that of the BMDs. On the other hand, the highest extrapolation error for both calculations is about 10%.

When calculating the average annual GSDs and BMDs the average must be taken of all examination types of medical practice: the relevant resulting variation lies between ± 10% and ± 20%. It can be concluded from this that the established doses per examination can be a satisfactory reference aid for medical practitioners. If it is intended to compare figures between regions or countries, the question of standardising the measurement procedures arises. The effort and expense involved to date in various countries in determining the relevant values is probably out of proportion to their meaningfulness; for this reason we would like to conclude by making an urgent request to those interested that they exert their influence in such a way that propositions on this topic be prepared in the near future.

Bibliography

(1) G. Poretti, R. Mini, G. Garavaglia, F. Ionesco, P. Ott und J. Feuz - Die Bestimmung der Knochenmark- bzw. der Gonadendosis infolge von diagnostischen Röntgenuntersuchungen. Report 1983 Inselspital, Radiuminstitut, CH-3010 Bern (Switzerland).

(2) Report of the United Nations Scientific Committee on Effects of Atomic Radiation, New York, 1958 and 1962.

EVALUATION OF THE GENETICALLY SIGNIFICANT DOSE DUE TO DIAGNOSTIC RADIOLOGY PROCEDURES IN FRANCE

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INTRODUCTION

As in many other developed countries, medical radiology in France is the second source of population exposure to ionizing radiation, following the natural background contribution. Furthermore the doses received by patients from routine x-ray examinations in France has not been assessed on anything approaching a national scale for the past 25 years.

The last evaluation of the mean individual gonadal dose had in fact been performed by Reboul and al. in 1957 in Bordeaux /1/. Since then, considerable changes in both, the practice of medical radiology and the technology in use in France, have occured, and the necessity of an updated Genetically Significant Dose (GSD) evaluation became evident.

Thus the CEPN (Centre d'étude sur l'Evaluation de la Protection dans le domaine Nucléaire) decided to perform in 1982 a new national survey of radiological activity to establish the annual GSD from medical x-ray procedures in France.

THE SURVEY OF THE RADIODIAGNOSTIC ACTIVITY

Information on the frequency of diagnostic x-ray examinations was obtained by questionnaires sent to 157 publics hospitals and 280 private clinics and surgeries spread throughout the country. These medical centers were selected using a stratified random sampling method.

The questionnaires asked for detailed informations concerning every x-ray examinations carried out at these medical centers (Hospitals, clinics, surgeries) during a specified week, such as the sex and age of the patients, the number of films used, the size of films, the fluoroscopic screening time, the tube voltage and the mAs used, the type of projection...

The response rate to the questionnaire was rather high : the average value is 70 % for both public and private medical centers. Eventually, 13 000 examinations were recorded for the whole country during the survey week.

The total annual number of x-ray examinations was estimated by multiplying the weekly values of radiological activity by seasonnal and other extrapolations coefficients : this procedure of assessment leads to a figure of 45,3 million x-ray examinations in France in 1982. This corresponds to 836 examinations per thousand of inhabitants (excluding mass chest screening, dental examinations, scanning procedures).

As it can be seen from the table 1 below, this rate is rather high when comparing with some earlier surveys in other industrialised countries.

Country	Year	Examinations per thousand
Japon	1979	1 013
West Germany	1978	864
France	1982	836
U.S.A.	1980	742
Sweden	1977	494
Great Britain	1977	408

Table 1 : Frequency of x-ray examinations per thousand head of inhabitants for industrialised countries /2/.

THE GONADAL DOSE MEASUREMENTS

Aiming at the assessment of the gonadal dose associated with every x-ray examination, a set of dosimetrics measurements have been performed, using thermoluminescent dosimeters (lithium borate chips), on an anthropomorphic RANDO phantom. These measurements took into account the following parameters :

- the size of each film used for a given x-ray examination,
- the projection related to each film,
- the anatomical focus point of x-ray beam,
- the fluoroscopic screening time.

The collective gonadal dose equivalent associated with 45,3 million examinations in France in 1982 reaches 28,600 man-Sievert : this means an individual mean gonadal dose equivalent of 0.53 mSv per inhabitant /3/.

When comparing this last value with that evaluated by Reboul in 1957, we observe an increase of 64 %.

Computed mean gonadal doses, averaged over all age groups and sex, are shown for each type of examination in figure 1 below.

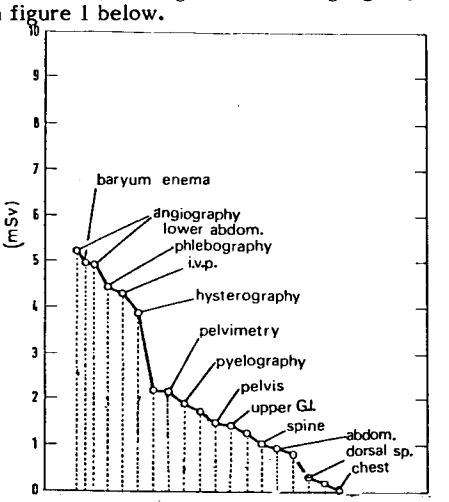

Fig. 1 : Mean individual gonadal doses per examination

As might be expected, the angiography of lower abdomen, the hysterography and the IVP appear to be the most important examinations in terms of gonadal exposure ; furthermore for these examinations it seems quite difficult to use gonadal shielding.

THE GSD

The previous results concerning the frequency and gonadal dose for each examination types have been combined with child expectancy of the patients to obtain the GSD. The total GSD to the population of France from all diagnostic examinations is estimated to be 290 μSv (290 m-rem).

Considering the French GSD value, it appears clearly that it is a mean value in comparison with other industrialised countries (see Fig. 2).

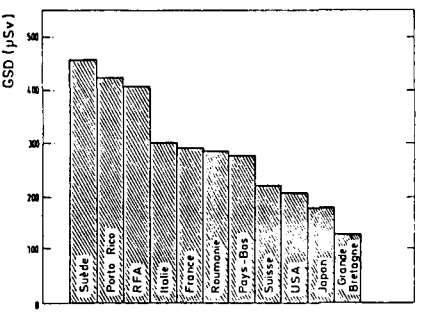

Fig. 2 : Genetically significant doses in different countries

The following figure shows to which extent the various types of examination contribute to the total GSD.

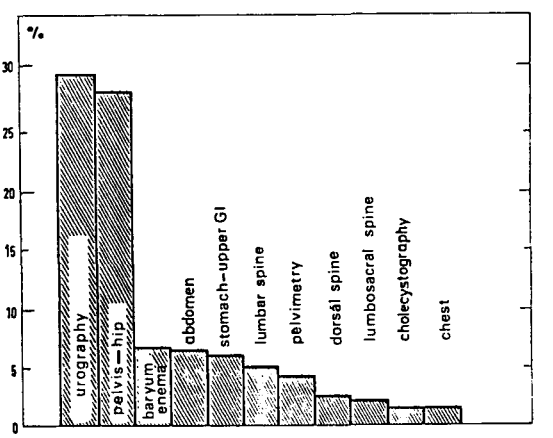

Fig. 3 : Relative contribution to GSD of different x-ray examinations (France 1982)

One can note that IVP and pelvis examination account for more than 50 % of the total GSD. The last figure 4 shows the break-down of the total GSD for each age group by sex.

It is important to stress that more than 50 % of the GSD is due to the exposure of the less than 30 years old female population ; otherwise the irradiation of the less than 1 year old children represents 16 % of the total GSD.

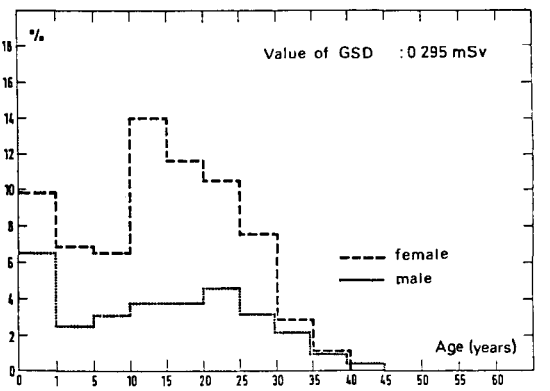

Fig. 4 : Contribution to GSD different age groups

While the GSD from diagnostic radiology amounts to only 14 % of that from natural background, it is however the largest man-made contribution to the collective GSD received by the population of France : the contribution of the other types of sources (radiotherapy and nuclear medecine, fallout, occupational exposure) does not exceed more than 3 % of the total GSD.

It is possible to make a very approximate estimate of the number of cases of serious hereditary ill health that will result from this level of population gonadal exposure. ICRP publication 26 /4/ recommends a value of 10^{-2} Sv^{-1} for the probability of serious defects appearing in the first two generations following irradiations of either parents with an additionnal equal probability for all succeeding generations. Applying the ICRP risk factor to a GSD of 290 μSv and the number of children born in France per year, the estimate is of 120 cases of serious hereditary ill health appearing in the population each year at genetic equilibrium if the current annual GSD is maintained.

REFERENCES

/1/ REBOUL J., DELORME G., TAVERNIER J. et al, "Doses gonades résultant de l'utilisation des radiations ionisantes en France", Ann. Radiol. 1959, 2, 1960, 3.

/2/ UNSCEAR, "Ionizing radiation sources and biological effects", 1982.

/3/ BENEDITTINI M., FAGNANI F., LEFAURE C., LUCCIONI C., MACCIA C., "Evaluation de la dose collective et de la dose génétiquement significative liées au radiodiagnostic en France en 1982", Rapport n° 68, CEPN 1983.

/4/ ICRP, "Recommandations of the International Commission on Radiological Protection", ICRP, Publication n° 26, Oxford, Pergamon Press, 1978.

KÖNNEN DIE DERZEIT ÜBLICHEN RADIOÖKOLOGISCHEN REFERENZKONZEPTE IHREN BEITRAG ZUM SCHUTZ DER BEVÖLKERUNG LEISTEN?

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Der Schutzbeitrag radioökologischer Referenzkonzepte, wie sie derzeit im Genehmigungsverfahren für kerntechnische Anlagen angewendet werden, ist nach wie vor umstritten. Die einen behaupten, ihr überzogener Pessimismus pervertiere die Realität und führe zu unangemessenen Auflagen hinsichtlich der Rückhaltung radioaktiver Stoffe. Die anderen bestreiten ihren Beitrag zum Strahlenschutz der Bevölkerung und argumentieren, ihre Anwendung im Genehmigungsverfahren führe zu einer systematischen Unterschätzung der realen individuellen Strahlenexposition der kritisch exponierten Personengruppe.
Welche Auffassung ist nun richtig?

Es sei gestattet, auf diese Frage zunächst mit einem Gedankenexperiment zu antworten: Man stelle sich vor, die jährliche Strahlenexposition aller in einem Umkreis von 10 km Radius um die vorhandenen Standorte kerntechnischer Anlagen lebenden Personen sei meßbar. Dann ergäbe sich aufgrund der Vielfalt der die Strahlenexposition beeinflussenden Größen ein Spektrum von Individualdosen (H), etwa derart, wie es in Abb. 1 dargestellt ist.

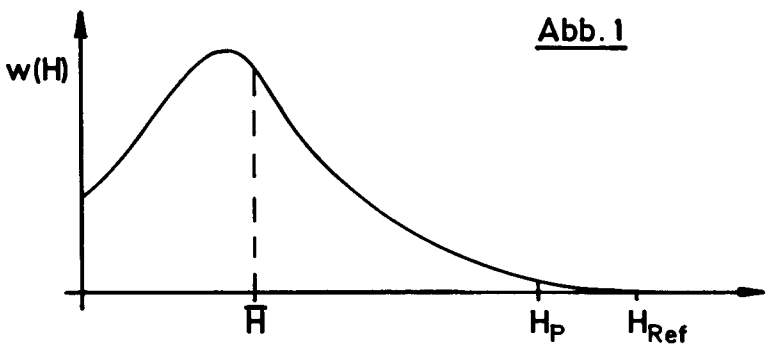

Abb. 1

Bei Kenntnis obiger Dosishäufigkeitsverteilung (w(H)) ließe sich die Frage nach dem Schutzbeitrag radioökologischer Referenzverfahren eindeutig beantworten, weil lediglich zu prüfen wäre, inwieweit die berechnete Referenzdosis (H_{Ref}) das Dosisspektrum einschließt oder, anders ausgedrückt, weil eindeutig angegeben werden könnte, mit welcher Wahrscheinlichkeit Strahlenexpositionen jenseits der Referenzdosis vorkommen können.

Nun ist aber die Äquivalentdosis prinzipiell nicht meßbar, so daß man zur Beantwortung der anfangs gestellten Frage darauf angewiesen ist, die real existierende Individualdosisverteilung wenigstens theoretisch so realitätsnah wie möglich zu approximieren. Diese Approximation hat einen statistischen und einen systematischen Aspekt, die sich z.T. gegenseitig beeinflussen [1]. Unter dem statistischen Aspekt wird im folgenden

insbesondere die Untersuchung der Form der Verteilung verstanden, während unter dem systematischen Aspekt die Analyse der Lage ihres Mittelwertes zu verstehen ist.

Zum statistischen Aspekt der Dosisberechnung sind in der Vergangenheit verschiedene Untersuchungen für die Ingestionsdosis durch J 131, Sr 90 und Cs 137 durchgeführt worden, [2], [3], [4], [5], [7]. Diese Arbeiten zeigen, daß kritische Individuen einer Personengruppe etwa um eine Größenordnung höher exponiert werden können als der Durchschnitt der Personengruppe oder, anders ausgedrückt, das Verhältnis $H_p/\bar{H}$ beträgt für die 99-Perzentile ($H_p=H_{99}$) der Verteilung (vgl. Abb. 1) etwa eine Größenordnung.

Die zitierten Arbeiten haben jedoch den Nachteil, daß sie die radioökologischen Referenzmodelle a priori als richtig voraussetzen und nicht hinterfragt wird, inwieweit die Referenzmodelle wegen der in ihnen enthaltenen zahlreichen Vereinfachungen und pessimistischen Annahmen, überhaupt geeignet sind, als Basis für eine statistische Analyse der Individualdosis zu dienen. Es bleibt offen, inwieweit diese Modelle systematisch von der Wirklichkeit abweichen, beispielsweise durch die Verletzung des Massenerhaltungssatzes oder inwieweit sie prinzipiell geeignet sind, die für eine statistische Analyse erforderliche Stichprobengröße im Rahmen ihrer mathematischen Struktur wirklichkeitsgetreu zu realisieren. Letzteres erfordert z.B. die Einbeziehung eines hinreichend großen räumlichen Bereichs um den Standort, um sicherzustellen, daß eine statistisch signifikante Anzahl exponierter Personen erfaßt wird. Diese Forderung gilt u.a. auch für die jeweils kritische Personengruppe, wie etwa für Säuglinge. Aus den genannten Gründen könnten die zitierten Arbeiten hinsichtlich der absoluten Lage ihrer Dosisverteilungen zu falschen bzw. voreiligen Schlüssen verleiten.

Zur Beantwortung der anfangs gestellten Frage nach dem Schutzbeitrag radioökologischer Modelle wird hier deshalb versucht, die o.g. statistischen Analysen der individuellen Strahlenexposition durch eine systematische Analyse zu ergänzen, d.h. auf der Basis eines möglichst realitätsnahen Szenarios und einer entsprechenden Modellstruktur wird versucht, eine möglichst gute Schätzung der zu erwartenden mittleren, individuellen Strahlenexposition ($\bar{H}$) zu erarbeiten, vgl. Abb. 1. Danach wird am Beispiel des derzeit in der Bundesrepublik Deutschland angewendeten Verfahrens durch Bildung des Quotienten $H_{Ref}/\bar{H}$ geprüft, inwieweit die mittlere Strahlenexpositon ($\bar{H}$) von der Referenzdosis (H_{Ref}) zur sicheren Seite hin abgeschätzt wird und ob der Sicherheitsfaktor ($H_{Ref}/\bar{H}$) ausreicht, um auch die Höherexposition kritischer Personengruppen gegenüber dem Durchschnitt, die sich durch den Quotienten ($H_{99}/\bar{H}$) ausdrücken läßt, zu kompensieren. Ist dies der Fall, d.h. gilt $H_{Ref}/H_{99} > 1$, so kann davon ausgegangen werden, daß das untersuchte Referenzverfahren seinen intendierten Beitrag zum Strahlenschutz der Bevölkerung in der Umgebung kerntechnischer Anlagen mit hoher Wahrscheinlichkeit leisten kann.

Die Konkretisierung des o.g. Verfahrens erfolgte an Hand von 3 exemplarischen Expositionspfaden:

Ingestion

J 131: Weide-Kuhmilch-Säugling (Schilddrüse)
Sr 90 Wurzelgemüse - Erwachsener (Knochen)

Äußere Bestrahlung

Cs 137: Bodenstrahlung (Gesamtkörper),

die in den betroffenen Organen bis zu 85 % der nuklidspezifischen Dosis verursachen. Gleichzeitig repräsentieren diese Expositionspfade die wichtigsten radioökologischen Prozesse, die im Referenzmodell Berücksichtigung finden.

Die Untersuchungen, deren Einzelaspekte an anderer Stelle [1] ausführlicher erläutert und diskutiert werden, zeigen, daß das deutsche Referenzverfahren die mittlere, realitätsnahe, individuelle Strahlenexposition im Falle der untersuchten Ingestionspfade um etwa zwei Größenordnungen systematisch überschätzt.

Im einzelnen ergibt sich für die Schilddrüsenexposition von Säuglingen durch J 131 über den Weide-Kuhmilch-Pfad eine Überschätzung des Durchschnittswertes um den Faktor 125. Diese Überschätzung ist im wesentlichen dadurch bedingt, daß Säuglinge entsprechend dem Referenzszenario ständig größte Mengen höchstkontaminierter Milch trinken (Faktor 64), vgl. [1].

Hinsichtlich der Strahlenexposition der Knochen erwachsener Personen durch den Verzehr von Sr 90-kontaminiertem Wurzelgemüse liegt die Referenzdosis, je nach Bodentextur um den Faktor 124 bis 210 oberhalb des Durchschnittswertes ($\bar{H}$). Dies ist im wesentlichen bedingt durch eine systematische Überschätzung der Aktivitätskonzentration im eßbaren Teil der Pflanze (Faktor 7 bis 12) und eine Überschätzung des Dosiskonversionsfaktors zur Berechnung der Knochendosis bei gegebener Inkorporationsmenge von Sr 90 (Faktor 6), vgl. [1].

Unter Beachtung der Tatsache, daß die $H_{99}/\bar{H}$-Werte für die Schilddrüsenexposition von Säuglingen durch J 131 bei etwa 7 und die für die Knochenexposition erwachsener Personen durch Sr 90, je nach Nahrungspfad , zwischen 7 und 21 liegen, ergibt sich für die hier untersuchten Ingestionspfade ein Sicherheitsabstand (H_{Ref}/H_{99}) der Referenzdosis gegenüber der Dosis kritisch exponierter Personen von etwa einer Größenordnung.

Für den Fall der äußeren Strahlenexposition durch Cs 137 am Boden ergibt sich für das Referenzverfahren gegenüber der Durchschnittsexposition eine systematische Überschätzung von einer Größenordnung. Für diesen Expositionspfad sind bisher keine statistischen Untersuchungen bekannt. Wegen der wenigen zu berücksichtigenden biologischen Parameter, die meist eine wesentlich größere Varianz aufweisen als beispielsweise rein physikalische oder chemische Parameter, ist jedoch anzunehmen, daß dieser Sicherheitsabstand ausreichen wird, um auch kritische Expositionsfälle einzuschließen.

Zusammenfassend bleibt festzustellen, daß das hier untersuchte, in der Bundesrepublik Deutschland benutzte, Referenzkonzept zum Schutz der allgemeinen Wohnbevölkerung vor zu großer Strahlenexposition für einzelne Expositionspfade gegenüber der mittleren Strahlenexposition am Standort kerntechnischer Anlagen Sicherheitsabstände bis zu zwei Größenordnungen bereithält, die durch die Höherexposition kritischer Gruppen gegenüber dem Mittelwert um etwa eine Größenordnung, nicht aufgezehrt werden. Der verbleibende Sicherheitsabstand zur kritischen Gruppe beträgt danach etwa eine Größenordnung. Es ist somit wenig wahrscheinlich, daß der verbleibende Sicherheitsabstand nicht ausreichen wird, um die Bevölkerung hinreichend zu schützen. Dies dürfte auch dann gelten, wenn man in Betracht zieht, daß die Szenarien,

auf deren Basis die angegebenen Zahlen zur statistischen Variabilität ($H_{99}/\bar{H}$) und zur systematischen Überschätzung ($H_{Ref}/\bar{H}$) der Strahlenexposition beruhen, nicht vollständig kompatibel sind. Dieses Faktum erzeugt eine gewisse Restunsicherheit in der Gesamtbeurteilung der Ergebnisse. Es ist jedoch zu hoffen, daß der verbleibende Sicherheitsabstand ausreichen wird. Diese Hoffnung wird durch die vorliegenden Meßergebnisse aus dem Bereich der Umgebungsüberwachung bei kerntechnischen Anlagen, z.B. in [6] bisher bestärkt. Für solche Fälle kann davon ausgegangen werden, daß das untersuchte Referenzverfahren seine Schutzfunktion mit hoher Wahrscheinlichkeit erfüllt. Von überzogenem Pessimismus kann dabei jedoch nicht die Rede sein.

LITERATURANGABEN

1. H.D. Brenk, K.J. Vogt: Quantifizierende Bemerkungen zur Frage der Konservativität der "Allgemeinen Berechnungsgrundlagen", Jül-1821, Dez. 1982

2. Dunning, D.E., Schwarz, G.: Variability of Human Thyroid Characteristics and Estimates of Dose from Ingested Dose, Health Physics, Vol. 40, pp. 661-675, 1981

3. Hoffman, F.O., Baes, C.F. (editors): A Statistical Analysis of Selected Parameters for Predicting Food Chain Transport and Internal Dose of Radionuclides, Oak Ridge Nat. Lab. Rep. No. NUREG/CR-1004, ORNL/NUREG/TM-282 Oct. 1979

4. Schwarz, G., Hoffman, F.O.: Imprecision of Dose Predictions for Radionuclides Released to the Environment: An Application of a Monte Carlo Simulation Technique, Environment International, Vol. 4, pp. 289-297, 1981

5. Schwarz, G.u.a.: Untersuchungen zur Entwicklung eines probabilistisch gestützten Bewertungskonzepts für Aktivitätsfreisetzungen nach Reaktorstörfällen, Brenk Systemplanung, Aachen, Interner Bericht BSU 8101/1, Febr. 1982

6. Wiechen, A.: Erste Erfahrungen mit der J 131-Überwachung der Milch aus der Umgebung von Kernkraftwerken, Milchwissenschaft 32, (5), 1977

7. Hoffman, F.O., Gardner, R.H., Eckerman, K.F.: Variability Dose Estimates Associated with the Food Chain Transport and Ingestion of Selected Radionuclides, Draft-Report No. NUREG/CR-2612, 1982

"REFERENCE JAPANESE MAN" AS A MODEL OF MAN FOR DOSE EQUIVALENT ESTIMATION

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Introduction

A program on physical and physiological characteristics of the Japanese is in progress in conjunction with importance of such data in establishing "Reference Japanese Man", a mathematical model for the dose equivalent estimation for radionuclide intakes. This is more important with respect to the general public including children. Moreover, data on variations to apply ICRP Reference Man to populations with different racial characteristics are needed(1-3). It is also expected that the Reference Man concept will need revision and extension(4).

Mass of organs, daily consumption of various nutrients in the diet, daily intake and skeletal content of some alkaline earth elements, and metabolic parameters for radioiodine in the thyroid gland was reported for the Japanese were reported previously(1,2).

In the present paper, age dependency of the mass of organs, mass of the mineralized bone and some results of determination of elements in tissues as well as further examination of radioiodine metabolism in the Japanese are to be reported.

1. Mass of Organs of the Normal Japanese of Various Ages

Age dependency is one of the factors of great interest in dose and risk evaluation and growth in mass of organs and tissues is essential data in dosimetry. Substantial data are available for the Japanese children and adolescents(1). Age of reference is often given as 1, 5, 10 and 15 years. Presently mass of organs is shown for the age groups, 1-2, 5-6, 9-10, 15-16 yr and the adult as in Fig. 1.

2. Mass of Mineralized Bone and Tissues in Bone

Estimated weight of bone as well as the distribution of mineralized bone in skeleton is being studied along with mass of cells near bone surfaces and active red bone marrow in the Japanese because of significance of these data in dosimetry(5). In the previous papers(1, 2), mass of the total skeleton of a "Reference Japanese" adult male was estimated as 8.4 kg using a weight ratio between the skeleton and total body, 0.14 which is assumed by ICRP(4). Since then, by a thorough investigation of the published data on the weight of bones and by measurements of a considerable number of complete sets of bone specimens(6), estimates of the weight of bones were obtained for the Japanese adult male and female, which are believed to be equivalent to masses of the mineralized bone(4). The data are shown in Table 1. According to the result, mass of the mineralized bone is 4.2 and 3.2 kg for the "Reference Japanese" adult male and female weighing 60 and 51 kg, respectively(5). Mass of cells near bone surfaces and active red bone marrow was estimated approximately as 100 and 1000 g, respectively.

3. Concentration and Distribution of Elements in Tissues

Up to now, results of the determination of elements using trace analysis techniques were obtained for eleven elements in twelve types of tissue, including the brain, kidney, liver, spleen, pancreas, small and large intestine, skin and diaphragm. More number of analyses is required, but, it may be of interest to note some difference in the concentration of some elements between the tentative result and some of the ICRP assumptions(4).

Uptake of Radioiodine by and Biological Half Time in the Thyroid Gland in the Japanese

In Japan, high levels of natural stable iodine intake, about 1 mg per person per day is known to be common because several kinds of marine algae including iodine-rich kelp or tangle, raw and often dried, are almost essential foodstuffs in the Japanese diet(1).

Results of *in vivo* measurement for an experimental administration of radioiodine in the volantary, healthy and normal Japanese adult males to clarify influences of stable iodine intake on the metabolism of orally ingested radioiodine, I-131, were reported(1,2). The data were further examined recently(7). Peak uptake rate and biological half-time in the thyroid was 10.3 % and 28.9 d, respectively in a subject who continued to eat "normal" Japanese meals throughout the course of the study, and 28.9 % and 38.5 d for another subject who was prohibited from eating any algae and their products for two weeks until the administration to simulate a low level of stable iodine intake as has been assumed in European and North American nations(*i. e.* 0.2 mg). The data, along with other literature data for the Japanese, may confirm the present assumption that, in the normal Japanese adult whose intake level of natural stable iodine is 1 mg per day, the fraction of radioiodine deposited and the biological half-time in the thyroid gland is 0.15 and 35 d, respectively. These values are considerably lower than those assumed by the ICRP(8) and will have significance in the estimation of dose equivalent commitment for radioiodine, particularly of long half-lives, such as I-129 (8). This was shown theoretically using a five compartment model in which release of excess inorganic iodine from the thyroid is assumed by Kai(8).

References

1) Tanaka, G., Kawamura, H. and Nakahara, Y.: Reference Japanese Man-I. Mass of Organs and Other Characteristics of Normal Japanese, Health Phys. 36, 333-346(1979).
2) Tanaka, G. Kawamura, H. and Nomura, E.: Physical Characteristics of the Normal Japanese in Relation to Reference Man, in Radiation Protection-A Systematic Approach to Safety, Vol. 1, Pergamon Press (1980), 288-281.
3) Stannard, J. N.: Internal Emitter Research and Standard Setting, Health Phys. 41, 703-708(1981).
4) "Report of the Task Group on Reference Man" ICRP Publication 23, Pergamon Press(1975).
5) Tanaka, G., Kawamura, H. and Nomura, E.: Reference Japanese Man-II. Distribution of Strontium in the Skeleton and Mass of the Mineralized Bone, Health Phys. 40, 601-614(1981).
6) Tanaka, G. and Nomura, E.: Unpublished data.
7) Uchiyama, M., Tanaka, G. and Akiba, S.: Radioiodine Retention by 2 Japanese Male Adults after a Single Oral Dose, J. Radiat. Res. 23, 358-370(1982).
8) Kai, M: Biological Half-Life of Radioiodine in Normal Japanese Thyroid, Hokenbutsuri(Tokyo) 18, 3-10(1983).

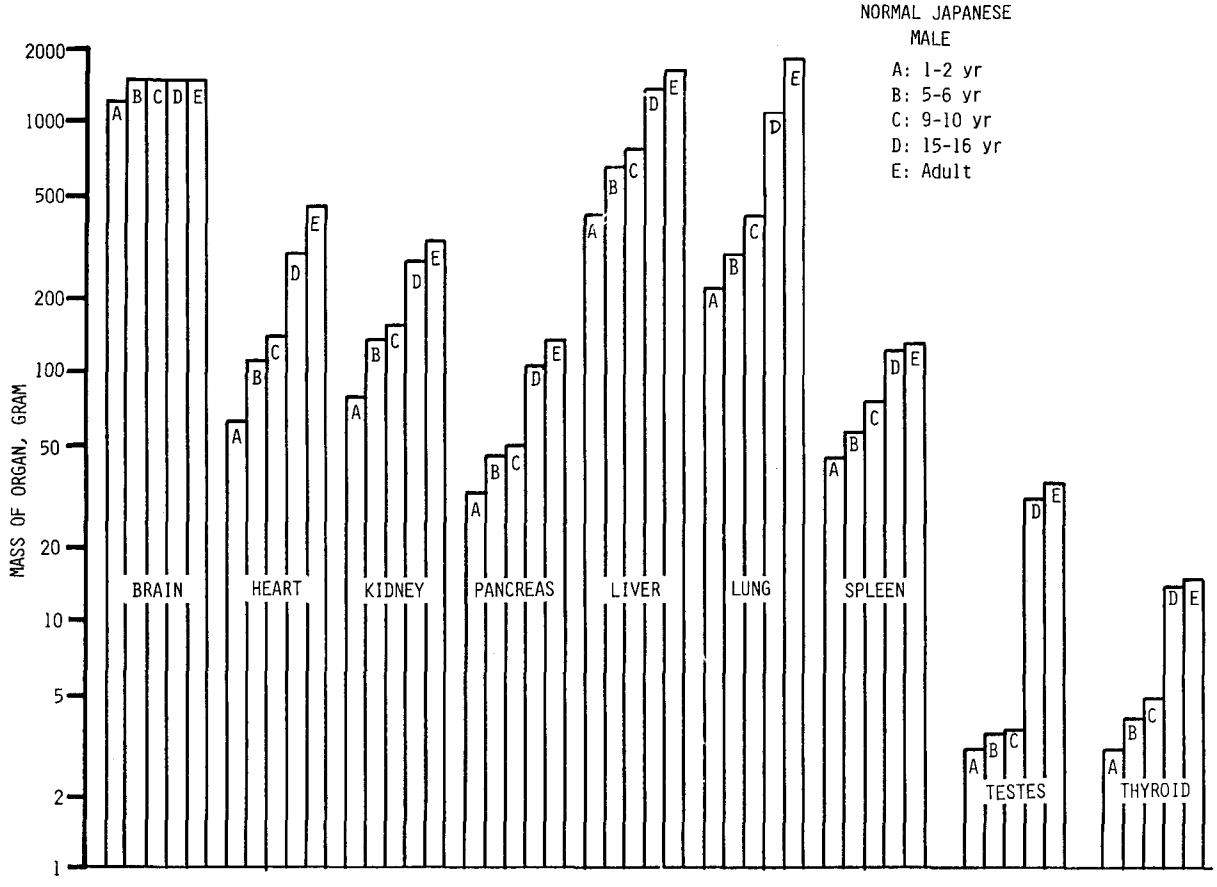

Fig. 1. Mass of organs by four different age groups

Table 1. Mass of the mineralized bone of the Japanese

Bone	Male	Female	Female/Male ratio
Skull(incl. mandible and teeth)	694	632	0.91
Scapula	129	97	0.75
Clavicle	50	38	0.75
Rib	307	230	0.75
Sternum	21	16	0.75
Vertebral column	384	288	0.75
Sacrum	83	62	0.75
Innominates	354	280	0.79
Humerus(2)	284	190	0.67
Radius(2)	90	60	0.67
Ulna(2)	112	76	0.67
Hands(2)	107	72	0.67
Femur(2)	781	571	0.73
Patella(2)	28	21	0.73
Tibia(2)	443	324	0.73
Fibula(2)	104	76	0.73
Feet(2)	229	167	0.73
Total	4200	3200	0.76

DOSE RATES DURING EXPERIMENTS WITH HEAVY IONS (ENERGY UP TO 20 MeV/u)

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The UNILAC (universal linear accelerator) at GSI, Darmstadt accelerates heavy ions - mainly argon to uranium - up to energies till 20 MeV/u.

The resulting dose rates during the normal course of the nuclear chemistry and nuclear physics experiment are only estimated in a rough way or calculated by simple methods. Therefore it is necessary to measure the dose rates in order to compare experimental results with the "calculations".

The measurements are done by "rem"-counters, activations foils and G.-M tubes with filters for different combinations of projectile and targets.

More than 90 % of the dose rate are coming from neutrons; therefore angular distributions of neutrons were measured. The obtained data are within an order of magnitude in agreement with the "calculations". The results may be useful for shielding calculations at heavy ion accelerators.

ASSESSMENT OF POPULATION DOSE FROM THE OPERATION OF ATOMIC POWER STATIONS IN INDIA.

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INTRODUCTION

Environmental surveillance is carried out at the atomic power station sites in India for evaluating radiation exposure of the general population. At present, there are two atomic power stations under operation for many years and the third has been commissioned last year. The Tarapur Atomic Power Station (TAPS) located at the coastal site at Tarapur has two BWR units (210 MWe each) and the Rajasthan Atomic Power Station (RAPS) located inland at Rawatbhata has two PHWR units (220 MWe each). The work presented here deals with evaluation of dose to the maximum exposed individual of the critical group of population and dose to population within 15 km from site from the environmental monitoring at the two power stations during 1980-82 period.

METHODS

Radiation dose to the public from the power plant releases is evaluated from the environmental surveillance data. The external dose due to stack plume is evaluated using TLDs installed at different locations in the environment.

Average daily intake of different food components produced in the area has been evaluated earlier(1). The daily intake of radionuclides by the individual public is estimated from the radioactivity data in the environmental samples of food materials and the dietary intake data. The committed effective dose equivalent (CEDE) is estimated from the ICRP(2) values of uSv/Bq intake.

From the measured external dose rate due to stack release and the population distribution in the concerned region, the population dose due to external exposure is evaluated. From the evaluated individual internal EDE and the population distribution, the population EDE due to radionuclide intake is evaluated. Addition of these two components give the power station contribution to the population EDE in the specified region.

TAPS (BWR) STATION

External exposure:

The external exposure showed a detectable increase over the natural background only at the two nearby population centres existing within 1.6 to 5 km region. The village in the north with a population of 2066, had the annual increase in exposure of 0.2 mGy during 1980-82 period. The villages in the south of site having population of 4678 had the annual increase in exposure

during the same period as 0.15 mGy. Thus the annual effective population dose equivalent as evaluated from TLD measurement is 1.115 person-Sv. The annual individual dose increase at the site fencing was 0.283 mGy which is the external exposure dose of the maximum exposed individual.

The gaseous release rate during 1980-82 period was 1.682 GBq/sec (10% Tech.Spec.Limit). The estimated dose from the gaseous release agreed well with the above evaluated dose.

Dose due to Dietary Intake

Terrestrial food products grown and consumed in the Tarapur environment beyond 2 km from site have not shown pick up of any power station released radionuclides. Cs-137 in the terrestrial food products (0.2 to 1.1 Bq/kg) is attributable to the pick up of stratospheric fallout radiocesium. Thus the terrestrial food samples collected beyond 2 km from site do not contribute any radiation exposure due to deposition or stack releases. Sea food samples from the coastal environment have been regularly analysed since this has been found as the critical food item for population exposure at Tarapur(3). I-131, Cs-134, Cs-137 and Co-60 are the only detectable station released radionuclides in the organisms collected from the coastal area. The last 3 years (1980-82) annual average radionuclide intake through diet and the resulting CEDE to individual public (4.1 uSv) is shown in Table-1 for region 2 to 15 km from site.

On a conservation estimate, it can be considered that population within 15 km from TAPS site have the intake of nuclides through sea food. The population within 15 km is 65000. Thus the population internal dose equivalent is estimated as 65000 x 4.1 uSv = 0.266 person-Sv. At Tarapur BWR environment the total population dose (within 15 km from site) evaluates to 1.4 person-Sv.

Table - 1.

Internal dose due to intake of radionuclides released from TAPS.

Critical food (sea food) intake = 52.2 kg/yr.

Nuclide	1980-82 Average annual intake for individual public Bq	Common from annual intake uSv
Co-60	56.89	0.14
I-131	53.24	1.33
Cs-134	55.33	1.51
Cs-137	65.77	1.12
	Total	4.10 uSv

Dose to the Maximum Exposed Individual from the Critical Group:

The fishermen group has been found as the critical group of population in the Tarapur environment(1). The CEDE to the maximum

exposed individual due to annual intake of radionuclides worked out as (i) CEDE through nuclides in sea food was .396 mSv and (ii) CEDE through nuclides in terrestrial food grown at site fencing was 0.11 mSv giving total CEDE of 0.407 mSv.

Since the nuclides considered have short effective half life, the dose is delivered within the same year of intake. The total of external and internal annual effective dose equivalent to the maximum exposed in the critical group was 0.690 mSv.

RAPS Station

External Exposure :

A-41 and H-3 are the main gaseous activity in the stack release. H-3 does not contribute to the external dose of the general public to any significant extent. Increase in external radiation dose due to A-41 could not be detected in the RAPS environment from the TL dosimeters distributed in different locations.

The average stack release rate of A-41 was 16 mBq/sec (0.75% of Tech.Spec.) and that of H-3 was 8 mBq/sec (.03% of Tech.Spec.). The estimated annual dose from this release rate was only 5 to 7 uSv at the station site boundary and it is insignificant beyond this region.

Dose Due to Dietary Intake & Inhalation :

The critical radionuclide in the RAPS environment for population exposure has been identified as tritium. H-3 in air has been measured even up to 10 km from site and H-3, Zn-65, Sr-90 I-131 and Cs-137 were the nuclides identified in the RAPS environment. The observed levels of Sr-90 and Cs-137 activity in the dietary material could be attributed to pick up of weapon fallout. Thus H-3, Zn-65 & I-131 are the nuclides contributing to the dose from the power station releases. The average annual intake of radionuclides for the period 1980-82 and the resulting CELE are given in Table-2.

Table - 2.

Internal dose to individual public due to intake of radionuclides released from RAPS.

Nuclide	1980-82 Average annual intake (Bq)	Cede from the annual intake (uSv)
H-3	2.25 x 10^5 (air + water)	3.82
Zn-65	27.7	0.15
I-131	1.75	0.04
	Total	4.01 uSv

In this case also, the dose can be considered as delivered in the same year as intake. With the individual CEDE as 4.01 uSv and the population within 16 km from the power site as 25000, the population dose equivalent due to power station releases is evaluated as 0.1 person-Sievert.

Dose to the Maximum Exposed in the Critical Group :

The critical group of population are the public living in the housing colonies on the banks of the Ranapratap Sagar lake, since they use lake water for drinking, irrigation and eat fish caught from the lake. The maximum exposed individual had the external exposure of 7 uSv due to A-41 measured at the site boundary. Due to inhalation of H-3 and intake of station released radionuclides through dietary materials and drinking water, the evaluated CEDE to the maximum exposed in the critical group during 1980-82 period was 175.0 uSv. The total internal and external annual dose equivalent is 0.182 mSv.

DISCUSSION

The maximum exposed the critical group at TAPS received about 14 % of the permissible annual dose of 5 mSv, but at RAPS it was only 3.6% of the permissible.

The population dose within 15 km at TAPS is 1.4 person-Sv where as it was only 0.1 person-Sv at RAPS indicating the lower environmental population exposure in case of RAPS.

ACKNOWLEDGEMENTS

We are greatful to Shri S.D. Soman, Head, Health Physics Division for useful discussion and approval of the publication.

REFERENCES

1. I.S.BHAT, P.R.KAMATH & A.K. GANGULY
Dispersal and uptake of radioactive elements in the Tarapur environment, Bhabha Atomic Research Centre, Bombay Report- BARC/644 (1972).

2. ICRP Publication-30 and its supplements, Pergaman Press (1978).

3. KAMATH P.R., BHAT I.S., GANGULY A.K.
Environmental behavior of discharged radioactive effluents at Tarapur Atomic Power Station. Environmental Aspects of Nuclear power stations.
(Proc. samp. New York 1970), IAEA, Vienna (1971), 475.

RADIOLOGICAL CHARGE ON POPULATION DUE TO SPANISH NUCLEAR INSTALATIONS UP TO 1981

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The paper presents in a tabular and graphic form the dose assessment to workers and public associates with the operations of the spanish nuclear power plants up to 1981.

From 1968 (date the first Spanish NPP was commissioned) dose received by workers have been collected. A distinction of the dose assigned to contract and plant personal or due to normal and maintenance work has been established.

Radioactive releases of NPPs have been reported in a yearly basis so with the knowledge of diffusion parameters, demographic distribution and population habits it can be known, through numerical models, the individaul (critical) and collective dose in a reasonable way.

Radiological surveillance around each NPP gives an approach of the radioactive situation derived from the operation of the plant. The determination of concentration of some radionuclides in different exposure pathways provides a method to ascertain the dose received through it.

All data considered in this report have a common objective: to estimate the nuclear risk of the population. Also can be used in epidemiological studies over cancer incidence in people affected by radiation coming from the NPP but it is possible to conclude in advance that any result in this sense has no logical meaning.

ORGAN DOSES IN ORAL RADIOLOGY

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Using the MIRD 5 phanthom modified by Kramer et al.(Ka82) and the Monte Carlo technique (Sn69), organ doses in patients undergoing external dental examinations were calculated taken into account the different beam geometries and the various incidences of the central beam with regard to the head of the patient. It was necessary to introduce in the original computer program a new source description, specific to dental examinations. To have a realistic evaluation of organ doses during dental examinations,it was necessary to introduce new regions in the phantom head which characterize the jaw (mandible and maxilla) with the teeth and the parotid, submaxillary and sublingual glands.

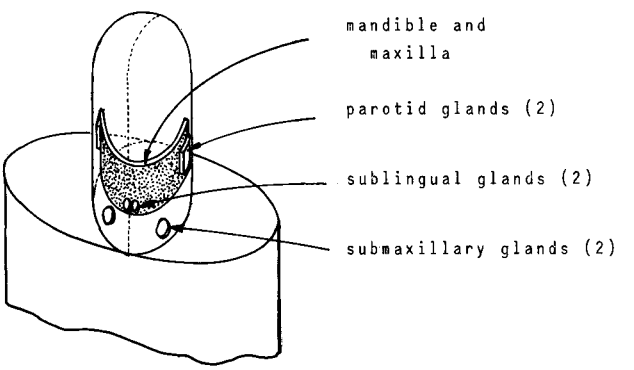

Fig. 1 . Head section of phantom illustrating approximate location of salivary glands and mandible.

Dose calculations were performed for 105 regions in the phantom. The X-rays beam was simulated by typical spectra generated at 50 kV and 70 kV and filtered by 2 mm of aluminium equivalent total filtration. Three regions of the jaw were taken as representative of routine dental examinations: incisor, cuspid and molar regions; and for each one, two incidences were simulated: superior and inferior. Also, three typical focus-skin distances of dental X-rays units were chosen: 10, 18 and 24 cm. Field sizes simulated in the calculations

have diameters of 7.0 , 9.0 and 11.0 cm.

Thirty six tables of organ doses normalized to the exposure at the skin entrance were generated according to the variation of the parameters mentioned above. As examples of the obtained results, tables 1 and 2 show figures of organ doses in typical irradiation geometries in dental examinations.

Table 1. Organ doses in dental examinations for variable incidence.

High Tension: 70 kV
Total Filtration: 2 mm Al
Ø Field at Skin: 7 cm
Focus-Skin Distance: 10 cm

Organ or Tissue	Dose Equiv./Exp. Skin Ent.(rem/R)*					
	Incisor		Cuspid		Molar	
	Inf.	Sup.	Inf.	Sup.	Inf.	Sup.
Thyroid	0.612	-	0.351	0.016	0.079	0.038
Red Marrow						
Facial Skeleton	0.031	0.010	0.039	0.019	0.058	0.059
Head	0.011	0.022	0.016	0.023	0.024	0.029
Total Body	0.002	0.003	0.002	0.003	0.004	0.004
Bone Surface						
Facial Skeleton	0.138	0.040	0.180	0.083	0.259	0.266
Head	0.010	0.127	0.014	0.113	0.023	0.062
Total Body	0.010	0.013	0.011	0.013	0.016	0.018
Skin						
Head	0.040	0.028	0.045	0.034	0.042	0.040
Total Body	0.004	0.003	0.005	0.003	0.004	0.004
Sublingual Glands	0.147	0.012	0.159	-	0.202	0.114
Submaxillary Glands	0.052	-	0.212	-	0.394	-
Parotid Glands	-	-	0.212	0.028	0.394	0.287
Lens of the Eyes	-	0.091	-	0.083	-	-

* - coeff. of variation: less than 10%

Table 2. Organ doses in dental examinations for variable focus-skin distance.

High Tension: 70 kV
Total Filtration: 2 mm Al
Ø Field at Skin: 7 cm
Incidence: Inferior Incisor

Organ of Tissue	Dose Equiv./Exp. Skin Ent.(rem/R)* Focus-Skin Distance (cm)		
	10.0	18.0	24.0
Thyroid	0.612	0.580	0.593
Red Marrow			
Facial Skeleton	0.031	0.033	0.033
Head	0.011	0.012	0.012
Total Body	0.002	0.002	0.002
Bone Surface			
Facial Skeleton	0.138	0.149	0.149
Head	0.010	0.010	0.010
Total Body	0.010	0.010	0.010
Skin			
Head	0.040	0.048	0.043
Total Body	0.004	0.005	0.005
Sublingual Glands	0.147	0.205	0.216
Submaxillary Glands	0.052	0.051	0.052
Parotid Glands	-	-	-
Lens of the Eyes	-	-	-

* - coeff. of variation: less than 10%

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CONCLUSIONS

Tables 1 and 2 present organ and tissues, where large amount of energy were deposited. In other phantom organs and regions not mentioned in the tables, the calculated values were extremely low. It can be observed that significant figures are found for the head and neck regions.

Figures presented in table 1 show the variation of organ doses per unit exposure at skin entrance for 3 representative regions of the jaw and 2 different incidences. From this table it can be seen large variations of thyroid and salivary glands doses with respect either to the region examinated or to the incidence of the central X-rays beam (upper and lower teeth).In table 2, figures are presented for different focus-skin distances. In this table it is shown that organ doses are independent of the focus-skin distances used in the dental units. Some additional tables were generated when the X-rays tube kilovoltage and the field size were modified. As expected, the higher were the field size and the kilovoltage,the higher were the organ doses obtained in the calculations.

In regard to the tissues localized in every part of the human body, namely: red marrow, bone surface and skin, both tables present figures for dose calculations taking into account either regions of the body or the total body. It can be observed that bone marrow and bone surface doses calculated only for the facial skeleton are in average ten times greater than those calculated for the total body. Skin dose is also significantly higher when dose calculations were performed using the head skin mass instead of the total body skin mass.

The results of the calculated organ doses clearly express the dependence on the irradiation geometry and the radiation energy. Therefore, when calculating collective effective dose equivalent for oral diagnostic radiology, organ doses should be weighted for all geometries, the weighting factors being the relative frequencies of dental examinations.

BIBLIOGRAPHY

Ka82 . Kramer,R.; Zankl,M; Williams,G. ans Drexler,G. : " The calculation of dose from external exposures using reference human phantoms and the Monte Carlo method".GSF Bericht S-885 (1982).

Sn69 . Snyder,W.S.; Ford,M.R.; Warner, G. G. and Fisher, H.L.: Journal of Nuclear Medicine,Supplement Number 3,Pamphlet 5, Society of Nuclear Medicine (1969).

EXPOSURE SURVEY IN ORAL RADIOLOGY USING A POSTAL SYSTEM

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The authors present the results of an exposure evaluation program in Oral Radiology carried out by the Institute of Radiation Protection and Dosimetry (IRD) of the Brazilian Nacional Commission on Nuclear Energy. They are related to approximately 1000 dental X-rays units surveyed in the area of Rio de Janeiro from 1981 to 1983 (Pe82).

The postal system used in the survey has to be exposed by the dentists and they also have to answer questions concerning the technical characteristics of the X-rays machine, the radiographic film and the average number of examinations performed per month. The exposures at the skin entrance for a single exposure of a standard dental irradiation (upper molar region) were measured with TLD dosemeters.

This survey shows four main results ofinterest in future programs:

1. the exposure distribution at the skin entrance for the standard examination ranges from 2 mGy to 40 mGy, what can be seen in the histogram presented in figure 1. The exposures measured above 5 mGy are mainly due to the lack of proper use of the exposure time control, and consequent incorrect film processing (HEW76).

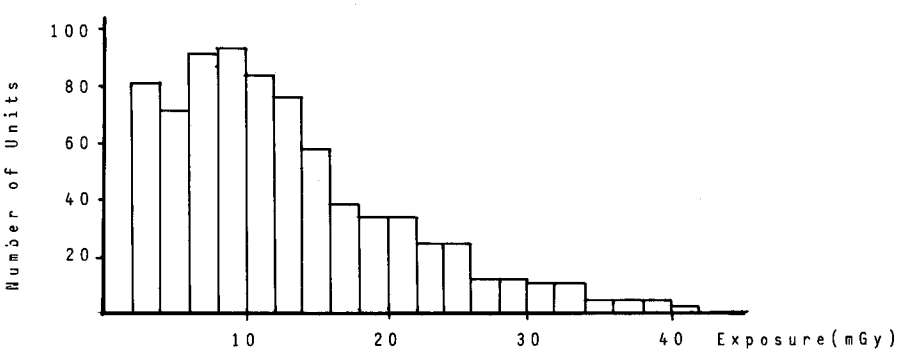


Figure 1. Distribution of the skin entrance exposures for the upper molar region examination measured with the postal kit.

2. the frequency distribution of the number of examinations performed monthly indicates that is possible to identify groups of dentists responsible for the exposure of large fractions of the

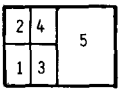
examinated population. Two groups of professionals were identified. To the first group belong the dentists who make from 10 to 100 examinations per month. They are 85% of all professionals but they make only 26% of the examinations. This means that in the second group are found just 15% of the dentists, who are responsible for 74% of the examinations.

3. making provisions so that the data obtained in the survey will be used in the evaluation of organ doses and, consequently, collective effective dose equivalent to the population undergoing dental examinations, the postal system measures the field size at the plane of the skin entrance and the half value layer of the X-rays beam.

4. X-rays machines surveyed come from different factories with the predominance of equipment made in Brazil. In regard to the film supply, it is done only by two factories, and Kodak has approximately 90% of the market.

As it was stressed before, incorrect processing of the film is one of main causes of the high exposures detected. In order to provide a more comprehensive approach to the program, it was recently developed a method using a postal system for testing the development of intraoral radiographs in dental offices.

Basically, the method consists of sending to a dentist a postal kit containing two pre-exposed periapical films, both having the same image. The right half registers the latent image of the molar region of a mandible phantom, and the left half contains four optical densities (base+fog, 0.25, 1.00 and 2.00 above base+fog density, when the film is appropriately developed). The four optical densities are used to determine the mean optical density ($\overline{OD}$) of the image.



1. BASE + FOG DENSITY
2. REFERENCE DENSITY 1
3. REFERENCE DENSITY 2
4. REFERENCE DENSITY 3
5. MANDIBLE PHANTOM IMAGE

Figure 2. Standard image used in the film development tests.

A questionaire is also included in the kit to obtain the necessary data for the evaluation of the resulting images. The questions concern to the developing procedure adopted by the dentist, the darkroom or portable sinkes maintenance procedures, dentist's speciality, comparison of the obtained images in the routine praxis and the standard image sent in the kit, etc...

Currently, the following scheme has been adopted: one film is developed according to the dentist's normal procedure, and the other for three minutes in the same developer solution. By comparing these two developed films with a similar standard image (processed at the IRD), and by applying a statistical method specially developed for this purpose, it is possible to evaluate the procedure followed by the dentist in developing the radiographs, the developer solution activity, and the darkroom or portable sink integrity (HHS83).

The statistical method used to evaluate the developing procedure adopted by the dentists correlates the variation of the mean optical density ($\Delta\%\overline{OD}$) of the images sent to them, and developed in their offices, with the probability of concordance that these images are identical to a standard one. The probabilities of concordance were obtained when a so called "judges test" was applied. In this test, films having the same latent image and developed from 0.5 minute to 8.0 minutes, were submited to dentists of different specialities (radiology, endodontics, surgery, geral practice, etc..). They pointed out those in wich the obtained image was considered to be identical to the standard image. Taking the judges's answers , the probabilities of concordance were determined and plotted against the percent variation of the mean optical density($\Delta\%\overline{OD}$) of the image, as shown in figure 3.

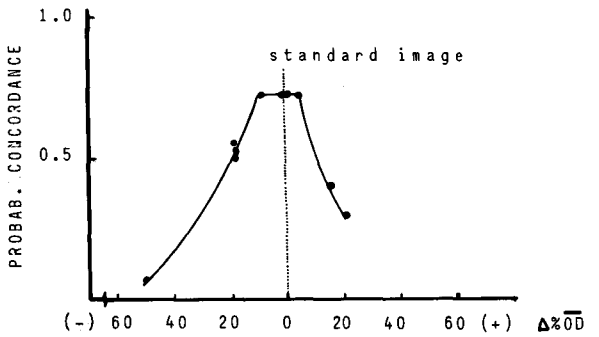


Figure 3. Probability of concordance of judgements as function of variation of the mean optical density of radiographic image.

According to the "judges test" , it was established acceptance limits for the mean optical density of the image developed in the dentists' offices. Images having mean optical density ($\overline{OD}$) between -10% and +5% of the $\overline{OD}$ of the standard image have 72 % of probability of being considered as identical to the standard image.

In 31 dental offices surveyed by this program,26 have presented images with $\overline{OD}$ either lower than -10% or higher than +5% of the $\overline{OD}$ of the standard image, with predominance of films processed at very short developing times. It can be seen from the fact that in 18 images the $\overline{OD}$ was lower than -30% ! Besides, 7 images have presented intense film darkening (fog) and only 8 developing solutions could be considered well active.

BIBLIOGRAPHY

Pe82 . Peixoto,J.E. and Ferreira,R.S.:" Programa de Avaliação Via Postal de Exposições em Radiologia Oral na Área do Rio de Janeiro".Technical Note 001/82 IRD/CNEN (1982).

HEW76. US Department of Health, Education and Welfare: " Dental Normalization Technique Instruction Manual". Publication HEW/FDA 76-8042, (1976).

HHS83. US Department of Health and Human Services:"Checklist for Establishing a Diagnostic Radiology Quality Assurance Program". Publication HHS/FDA 83-8219, (1983).

EFFECTIVE DOSE EQUIVALENT IN NUCLEAR MEDICINE INVESTIGATIONS

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When the concept of "effective dose equivalent" was introduced by ICRP (1,2) the intention was to use it for the radiation protection of workers. With this concept it is possible to make an assessment of the risk from radiation by a single figure, independent of the distribution of the absorbed dose within the body. To achieve this, different weighting factors for various organs, reflecting the relative risk from irradiation of that specific organ, is applied. The estimation of the radiation sensitivity of the various tissues is, however, very rough, but the weighting factors attempt to take into account lethal cancer, as well as genetic effects up to the second generation.

Even if the effective dose equivalent primarily was intended for occupationally exposed workers, it has been approved also for patients undergoing nuclear medicine or X-ray diagnosis. Since patients are not a representatitive part of the general adult population, the weighting factors used by the ICRP are not adequate in this case. For older persons, who are in great majority among nuclear medicine patients (about 3/4 is more than 50 years old (3)), the genetic risk is of no significance. Calculated somatically effective dose equivalents has therefore been published both for X-ray diagnosis (4), and for nuclear medicine investigations (5,6). In the latter case the calculations are based on the ICRP weighting factors, excluding the gonads. In all cases the normalized somatic weighting factors, w_{S1}, according to Table 1, is used. However, since the gonads ofter recieve a relatively low absorbed dose, the resulting figure for the somatically effective dose equivalent, calculated in this way, is often higher than that for the effective dose eqiuivalent. This is due to the renormalization of the ICRP weighting factors, and gives a false impression of a higher risk for a population in which the genetic risks can be neglected. We therefore suggest to use the original weighting factors, except for the gonads, w_{S2}, in order to make a simple comparison with the total effective dose equivalent. Following our proposal the same risk figure, 0.0165 Sv^{-1}, can be applied both for the total and the somatically effective dose equivalent.

In Table 2, we give, for some commonly used radiopharmaceuticals, the effective dose equivalent, both including and excluding the contribution from the gonads, together with the mean absorbed dose in the total body. The figures in this table are derived from a compilation of absorbed doses from radiopharmaceuticals by Johansson, Mattsson and Nosslin (7). In some cases, however, calculations based on more recent biokinetic data, have been performed, using the MIRD formalism (8). Differences in the numerical value for the effective dose equivalent and the mean absorbed dose in the total body is due to an inhomogeneous distribution of the radiopharmaceutical. The greatest differences arise when the radiopharmaceutical is concentrated in an organ with a high weighting factor relative to it's mass. The magnitude of the ratio between the effective dose equivalent and the mean absorbed dsoe in total body is indicated in Table 2.

The effective dose equivalent is of great value in comparing the risk from various medical radiodiagnostic procedures. The concept is also useful in estimating the collective dose from various sources of radiation.

Table 1.

ICRP based weighting factor for calculation of the total effective (w_T) or somatically effective (w_{S1} and w_{S2}) dose equivalent.

Tissue	w_T	w_{S1}	w_{S2}
Gonads	0.25	0	0
Breast	0.15	0.20	0.15
Red bone marrow	0.12	0.16	0.12
Lung	0.12	0.16	0.12
Thyroid	0.03	0.04	0.03
Bone surface	0.03	0.04	0.03
Remainder	0.30	0.40	0.30
Sum	1.00	1.00	0.75

The weighting factor for the remainder is equally divided between the five remaining organs or tissues recieving the highest dose equivalent.

Table 2.

Nuclide	Radiopharmaceutical	Total (H_E) effective dose equivalent (mSv/MBq)	Somatically* (mSv/MBq)	Mean abs. dose in total body ($\bar{D}_{TB}$) (mGy/MBq)	$\frac{H_E}{\bar{D}_{TB}}$
H-3	Water	0.015	0.011	0.015	++
C-14	Aminopyrine	0.0029	0.0022	0.0029	+
F-18	Fluoride ion	0.016	0.016	0.013	+
Na-22	Sodium ion	3.1	2.4	2.9	+
Na-24	Sodium ion	0.34	0.26	0.29	+
P-32	Phosphate ion	1.7	1.5	1.5	+
Ca-47	Calcium ion (oral)	1.5	1.4	0.40	++
Sc-47	Scandium ion (oral)	0.42	0.41	0.040	++++
Cr-51	Chromium (III) ion	0.11	0.10	0.058	+
Cr-51	Chromate ion (oral)	0.035	0.030	0.0063	+++
Cr-51	Chromium labeled denat. erythrocytes	0.40	0.39	0.049	++++
Cr-51	Chromium-EDTA	0.0024	0.0021	0.00087	++
Cr-51	Chromium labeled erythrocytes	0.21	0.20	0.099	++
Fe-55	Iron (III) ion, citrate	1.3	1.2	0.62	++
Fe-59	Iron (III) ion, citrate	13.	11.	7.9	+
Co-57	Vitamin B-12 (oral)	2.9	2.9	1.5	+
Co-58	Vitamin B-12 (oral)	5.9	5.7	3.1	+
Ga-68	Gallium-EDTA	0.053	0.050	0.011	+++
Se-75	l-Selenomethionine	2.9	2.1	2.2	+
Sr-85	Strontium ion	0.85	0.69	0.85	+

Table 2, cont.

Tc-99m	Pertechnetate	0.011	0.0090	0.0038	++
Tc-99m	Insoluble compunds (oral)	0.025	0.023	0.0048	+++
Tc-99m	Technetium labeled albumin	0.0064	0.0048	0.0043	+
Tc-99m	Technetium labeled denat. erythroc.	0.053	0.053	0.0052	++++
Tc-99m	Technetium labeled DMSA	0.016	0.015	0.0046	++
Tc-99m	Technetium DTPA	0.0065	0.0057	0.0030	++
Tc-99m	Technetium labeled erythrocytes	0.0076	0.0066	0.0049	+
Tc-99m	Technetium labeled phosphorus comp.	0.0061	0.0049	0.0024	++
Tc-99m	Technetium labeled gluconate compl.	0.0091	0.0083	0.0027	++
Tc-99m	Technetium lab. liver-biliary subst.	0.020	0.019	0.0034	+++
Tc-99m	Technetium labeled colloids	0.013	0.012	0.0051	++
Tc-99m	Technetium lab. makroaggr./microsph.	0.012	0.012	0.0031	++
Tc-99m	Technetium labeled plasmin	0.0080	0.0075	0.0033	++
In-111	Indium DTPA	0.026	0.023	0.0082	++
In-111	Indium labeled leukocytes	0.63	0.62	0.17	++
In-111	Indium labeled lymphocytes	0.65	0.63	0.17	++
In-111	Indium labeled thrombocytes	0.74	0.72	0.17	+++
In-113	Indium ion	0.012	0.011	0.0047	++
I-123	Iodide ion (thyroid uptake: 35%)	0.17	0.17	0.0087	+++++
I-123	Iodide ion, blocked thyroid	0.013	0.011	0.0077	+
I-123	Iodine labeled albumin	0.021	0.016	0.015	+
I-123	Iodine labeled fatty acids	0.11	0.11	0.011	++++
I-123	Orthoiodohippuric acid	0.015	0.014	0.0015	++++
I-125	Iodide ion (thyroid uptake: 35%)	10.	10.	0.19	++++++
I-125	Iodide ion, blocked thyroid	0.014	0.012	0.0073	+
I-125	Iodine labeled albumin	0.29	0.22	0.22	+
I-125	Iodine labeled fibrinogen	0.11	0.10	0.063	+
I-125	Orthoiodohippuric acid	0.011	0.011	0.0015	+++
I-131	Iodide ion (thyroid uptake: 35%)	16.	16.	0.26	++++++
I-131	Iodide ion, blocked thyroid	0.077	0.065	0.040	+
I-131	Iodine labeled albumin	0.68	0.51	0.48	+
I-131	Iodine labeled albumin, microaggr.	0.037	0.035	0.017	++
I-131	Orthoiodohippuric acid	0.065	0.062	0.0062	++++
I-131	Iodine labeled makroaggr./microsph.	0.39	0.37	0.051	+++
I-131	6-Iodomethylnorcholesterol	1.0	0.65	0.35	++
Xe-133	Xenon gas (inhal., with rebreathing 5 min)	0.00069	0.00057	0.00038	+
	(inhal., single breath, hold 30 s)	0.00024	0.00024	0.00012	++
Xe-133	Xenon gas in solution (breathhold 30 s, rebreath 5 min)	0.0010	0.00083	0.00057	+
	(wait 10 s, breathhold 30 s)	0.00043	0.00036	0.00024	+
Yb-169	Ytterbium DTPA	0.048	0.044	0.013	++
Tl-201	Thallium ion	0.094	0.058	0.035	++

* With the genetic contribution (0.25x(Absorbed dose to the gonads)) subtracted.

** The following intervals for the ratio $\frac{H_E}{D_{TB}}$ are indicated:

+: 1 - 2, ++: 2 - 4, +++: 4 - 8, ++++: 8 - 16, +++++:16 - 32, ++++++: 32 - 64.

REFERENCES

1. International Commission on Radiological Protection: Recommendations of the ICRP. Annals of the ICRP Vol. 1, No. 3, 1977, ICRP Publication 26

2. International Commission on Radiological Protection: Statemant from the 1978 Stockholm Meeting of the ICRP. Annals of the ICRP Vol. 2, No. 1, 1978, ICRP Publication 28

3. Johansson L, Mattsson S: To be published.

4. Laws PW, Rosenstein M. Health Phys 35 (1978) 629 -642.

5. Roedler HD: "Radiation dose to the patient in radionuclide studies", Medical Radionuclide Imaging (Proc. Synp. Heidelberg 1980) Vol. 1, IAEA Vienna 1981, 527-542

6. Persson BRR: "Effective dose equivalent concept in radiopharmaceutical dosimetry", Proc. Symp. on Radiopharmaceutical Dosimetry, Oak Ridge 1980 (Watson EE et al. Eds.), HEW Publ. FDA 81-8166 (1981) 616-624

7. Johansson L, Mattsson S, Nosslin B: Radiation doses from radioactive substances in medical use. 1981. Swedish National Institute of Radiation Protection, Box 60204, S-10401 Stockholm. In Swedish with an English word-list.

8. Snyder WS, Ford MR, Warner GG, et al.: "S" absorbed dose per unit cumulated activity for selected radionuclides and organs. MIRD Pamphlet No. 11, New York, Society of Nuclear Medicine, 1975

EXPOSURE MODEL FOR A GYNECOLOGIC RADIUMTHERAPY. ASSESSMENT OF EFFECTIVE DOSE EQUIVALENT

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INTRODUCTION

An exposure mathematical model has been developed in order to assess the mean organ equivalent dose and the effective dose equivalent, H_E, absorbed by occupationally exposed workers during a gynecological practice with Ra-226.

The H_E -defined in ICRP Pub. 26 (1)- that is absorbed by a worker during a practice involving ionizing radiations is generally hard to estimate. Usually, the knowledge of the dose equivalent absorbed by the personal dosimeter is not sufficient for evaluating the H_E. An analysis must be made, for each type of practice, concerning the working place and consisting in the assessment of all its specific parameters, such as the activity of the emitting source, the spectral and angular distribution of the radiation field, location and displacements of the exposed person, source-person distance, operation times and geometrical configuration of the practice.

For some types of practices, the application of theoretical conversion factors will be enough for the evaluation of H_E (2). Other practices will require a simulation through the development of physical or mathematical models for the evaluation of the organ dose equivalent and of H_E. The last procedure has been selected and a mathematical model was developed based on the application of the Monte Carlo method to the transport of photons in an anthropomorphous phantom.

A gynecological practice was recently considered by Eckerl, H.(3) In our case, a different procedure for the same practice was selected, where the H_E resulted from the application of the model. Both results were finally compared.

DESCRIPTION OF THE PRACTICE

An analysis was made of a gynecological practice consisting in the implantation of Ra-226 sources in utherus. This practice is usually performed by a gynecologist, an assistant, an instrumentist, a nurse and a physicist. During the therapy, all those occupationally exposed remain behind a 5 cm-thick lead shield and are mainly exposed in upper limbs, head and upper trunk.

The H_E absorbed by the gynecologist during a given implantation was assessed by means of an exposure mathematical model. This model is based on the analysis of a working place performed during actual implantations at the Krankenhaus München-Schwabing. A 100 mg Ra-226 punctual source is considered and the exposure is decomposed into four components called A, B, C and C'. Component A is related with the initial handling of the source, which lasts 5 seconds. Component B occurs during the implantation of the source in uthe-

rus, as the gynecologist works seated behind the lead shield for a period of 2.75 minutes. Component C takes place at the end of the application, when the physician stands up and remains standing in front of the patient for 1 minute, while the physicist performs control measurements in rectum and bladder. For comparison purposes, the results that would be obtained if the physician were sitted behind the shield have been considered as a fourth component, C'.

MATHEMATICAL MODEL

The dose equivalent distribution in the human body and the H_E were assessed by developing a code through the application of the Monte Carlo method to the transport of photons in a MIRD V phantom. The code was based on Kramer's code (4).

The history of each photon started by making it to fall upon a mathematical cylinder surrounding the phantom. In this way, a simulation is made of the interactions suffered by the photon in its way through the air from the source to the phantom surface.

The interaction point was selected through the usual procedure:

$$l = -\frac{\ln r}{\mu_0}$$

where:

r: is an alleatory number uniformly distributed between 0 and 1.

μ_0: is the total linear attenuation coefficient.

The difficulty caused by the various regions of the phantom in the choice of the mean free path was solved by applying Colemann's technique(5). A statistic weight factor was introduced for allowing the photon survival and its value was reduced after each interaction as per the following expression:

$$W_n = W_{n-1} \frac{\mu_{co}(E_{n-1})}{\mu_T (E_{n-1})}$$

where:

W_{n-1} and W_n : are statistical weights for the $(n-1)^{th}$ and the n^{th} collisions.

μ_{co} and μ_T : are Compton and total attenuation coefficients.

The photon was forced to continue with its history, with Compton interactions, and then, the energy and the direction cosines of the outgoing photon were calculated. The history of the photon comes to an end if it escapes from the phantom, if its energy is lower than a given threshold or if the statistical weight is lower than a given minimum value. In the latter case, an additional correction of the weight, as described by Kramer, R.(2), was performed. When the energy of the photon is higher than 1,022 MeV, 2 photons of 0.511 MeV are generated at the interaction point and the histories of these two photons are analyzed independently.

The energy, E, deposited in the n^{th} interaction was calculated as follows:

$$E^{(n)} = W_{n-1} \left[\frac{\mu^{Ph}(E_{n-1})}{\mu^{T}(E_{n-1})} E_{n-1} + \frac{\mu^{Co}(E_{n-1})}{\mu^{T}(E_{n-1})} (E_{n-1}-E_n) + \frac{\mu^{PP}(E_{n-1})}{\mu^{T}(E_{n-1})} (E_{n-1} - 1{,}022\ MeV) \right]$$

in which $\mu_{Ph}(E_{n-1})$, $\mu_{Co}(E_{n-1})$, $\mu_{PP}(E_{n-1})$ and $\mu_{T}(E_{n-1})$ are the mass-attenuation coefficients for the photoelectric, Compton, pair-production and total processes previous to the collision at the site considered, respectively.

The code considers the particular geometry of the practice and of the Ra-226 source. The initial energy of each photon introduced is obtained by comparing an alleatory number with the accumulated probability of the various spectral lines emitted by the Ra-226 source. The geometrical consideration corresponding with each one of the components A, B, C and C' in Table 1 are taken into account independently. Thus an evaluation is performed of the various contributions to H_E in the gynecological practice.

The H_E was calculated by means of the following equation:

$$H_E = \Sigma_T W_T H_T$$

where:

W_T is the weighting factor representing the proportion of the stochastic risk resulting from tissue T to the total risk, when the whole body is irradiated uniformly. The male W_T factors were used in this work(5).

H_T is mean dose equivalent in tissue T.

The H_E absorbed by the gynecologist during the therapy may be assessed by means of the following superposition:

$$H_E = H_E^{(A)} + H_E^{(B)} + H_E^{(C)}$$

Since H_E, as defined by ICRP 26 (1), is not additive, its value was calculated by using a fixed-remainder model composed by thymus, SI, ULI, stomach and liver. Thus, the total H_E may be expressed as a superposition of the H_Es due to each component, A, B and C or C'.

RESULTS

The mean organ dose equivalents obtained for components A, B, C and C' are shown in Table 1.

Table 1

Mean organ dose equivalent (mrem)

Tissue	A	B	C	C'
Testes	< 0.10	2.02	0.70	0.73
Lungs	< 0.10	0.77	2.44	0.28
Red Bone marrow	< 0.10	1.29	2.10	0.47
Thyroid	< 0.10	8.01	2.03	2.91
Bone surface	< 0.10	1.29	2.10	0.47
Skin	< 0.10	2.74	2.09	0.99

The variation coefficients obtained depend on the size of the organ, the maximum value being 15% for testes and thyroid. The H_E absorbed by the gynecologist is shown in Table 2.

Table 2

Effective dose equivalent

Components	H_E (mrem)
A + B + C	4.3
A + B + C'	2.7

CONCLUSIONS

A comparison was made between results obtained from our model and the assessments made by Eckerl, H.(3). The difference between the H_Es absorbed by the gynecologist was only 16%, indicating a very good correlation, considering the associated errors.

On the other hand, it must be noted that the H_E absorbed by the gynecologist may be reduced in a 1.6 factor, if the physician remains sitted behind the shield while the physicist performs the control measurements. Such factor was obtained by comparing the H_Es obtained independently considering exposure components C and C'.

The developed exposure Monte Carlo model, considering the spectral and angular distribution of the field, the irradiation geometry, the superposition of components A, B and C or C' and some other specific parameters, allows for a correct simulation of the gynecological practice under analysis.

BIBLIOGRAPHY

1. ICRP Publication 26. Oxford, Pergamon Press, 1977. Annals of the ICRP, v.1: No. 3, 1977.
2. Kramer, R. 1979 GSF-S-556.
3. Eckerl, H., Thomasz, E. and Drexler, G. Strahlentherapie (Germany, F.R.) v.158: 422-426, 1982, No. 7.
4. Kramer, R. Private communication.
5. Colemann, W.A. Nucl. Sci. Eng. (USA) v.32: 76-81, 1968.

ACKNOWLEDGEMENT

The authors are grateful to Dr. G. Drexler and Dr. R. Kramer, for the discussions held on their codes.

ASSESSMENT OF POPULATION DOSE COMING FROM HOSPITAL RADIOACTIVE WASTES

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The purpose of present work is the assessment of population dose due to releases of radioactive liquid wastes from the Spedali Civili of Brescia, where the following non-sealed isotopes are mainly employed: J-131 (40 Ci/year), Tc-99m (30), J-125 (.04), Cr-51 (.03). The knowledge of impact onto environment and man from such releases is achieved by either measuring radioisotope activities and concentrations in concerned water bodies and by studying radioactivity diffusion through food chains to man.
Measurement of the activity in contaminated water is performed by doing spectrometric analysis of treated water samples with a Ge(Li) detector connected to a MCA. Resulting spectra are processed by a computer program which performs peak search and identification, giving the activity of each identified isotope.

Measurements on J-131 have been performed, by which we obtained a dilution curve which allows one to calculate concentrations downstream from release point. By means of parameters of a linear compartmental model we calculated concentrations in critical pathways, i.e. fruit, cereals and milk, and consequently dose to concerned population.

Dose values obtained for J-131 are very low: 5.41×10^{-4} mrem/year/person to thyroid, 9.49×10^{-7} to total body. Collective dose values (250.000 concerned people) are very low, too: .135 man. rem/year for thyroid and 2.37×10^{-4} man/rem/year for total body.
Measurements concerning the other isotones are in progress and, hopefully, will be completed in the next few months.

Bibliografia

- United States Nuclear Regulatory Commission: Regulatory Guide 1.109-1976
- Albini E., Belletti, S., Rossetto, E.: Lo scarico radioattivo monitorato presso gli Spedali Civili di Brescia: risultato di una campagna di rilevasione e modello di impatto radiologico con l'ambiente.
Radiologia Medica, to be published

STRONTIUM-90 CONTENT IN HUMAN BONE OF WEST GERMAN RESIDENTS

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INTRODUCTION

Quantitative estimations of the radionuclide strontium-90 in human diet deriving from the fallout after nuclear weapons tests in the atmosphere, are still carried out today all over the world. To gain insight into transfer and discrimination of strontium-90 in comparison to calcium in the food chain, examinations of human bone and other tissues are a necessary addition to the estimation of this radionuclide in the food products.

From 1958 to 1971, in the Federal Republic of Germany extensive studies on the strontium-90 content in human tissues were made by the Institute of Forensic and Social Medicine of the University of Kiel. The results of these studies were the basis for evaluating the radiation exposure according to the recommendations of ICRP and they were published in the annual reports "Environmental Radioactivity and Radiation Exposure" by the Government of the Federal Republic of Germany. Furthermore they were included in the UNSCEAR Reports of the United Nations and represent a significant contribution to the stipulations, put forth within the framework of the Euratom contract, on the surveillance and control of environmental radioactivity. These studies were concluded in 1971. In the year 1977, the estimation of strontium-90 in human bone was resumed in co-operation with the Institutes of Forensic Medicine and of Pathology, both of the University of Munich, and the Institutes of Forensic Medicine and of Pathology, both of the University of Saarland/Homburg. Meanwhile results from five years are available.

MATERIAL AND METHODS

In the years 1977 to 1982, defined sections of the femur or tibia diaphysis were taken from dead male and female persons of all age groups and deepfrozen after the surrounding tissue was removed. The samples collected in this manner were wet ashed with nitric acid and hydrogen peroxide and the subsequent radiochemical separation was made according to the procedures of the Environmental Measurements Laboratory (EML 1982) in a slightly modified method. The content of natural strontium as well as the chemical yield of the strontium carrier were estimated by atomic absorption spectrophotometry in the nitrous oxide/acetylene flame. The beta activity of the separated yttrium-90 counting sample was measured with a low-level omniguard counter and the quantitative determination of calcium was made by complexometric titration with EDTA.

RESULTS AND DISCUSSION

The results from the year 1977 have shown that in the age group of approx. 15 to 25 years, the intake of strontium-90 into human bone was significantly increased as compared to the 25 to 55 years age group. This can be attributed to the fact that the 15 to 25 year age group were in the first stage of bone growth at the time of maximum fallout in the years 1964-65 and therefore incorporated the radionuclide in increased levels. As compared to other bones of the human body, that part of the femur or tibia diaphysis used for the examinations has only a low turn-over rate, so that the deposited strontium-90 is leaving the bone at a very slow rate. Since the fallout has considerably decreased, only a small amount of strontium-90 is incorporated into human bone today. A follow-up of the strontium-90 concentration in this bone section up to the year 1982 demonstrates that the radionuclide content in bone is slowly decreasing and the maximum concentration is simultaneously shifting to older age groups.

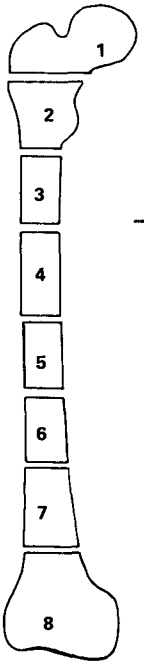
In the year 1982, further bone samples of the 55 years and older age group were examined. On this occasion it was found that a higher intake of strontium-90 is also indicated in this age group. This may be due to an increased osteoporotic or postclimacteric process in the skeletons of this age group which caused, on account of a higher turn-over rate, a stronger incorporation of strontium-90 during or after the maximum fallout period. Natural strontium, however, is absorbed from food and incorporated into the skeleton at uniform levels of concentrations whereby strontium experiences a discrimination in comparison to calcium. For this reason an age-dependent concentration of natural strontium in the examined bone section was not observed.

In the year 1982 a complete right femur (72 years, female) divided into eight sections was examined. The measurement results show that the bone section from slightly below the centre towards the knee joint contains the lowest amount of strontium-90 and the concentration increases in the direction of the epiphyses by about 60 %. This may be explained by the fact that in the maximum fallout period in the years 1964 - 65 the bone growth of the femur was already complete and strontium-90 was incorporated and decorporated again merely by a small turn-over rate in the medium bone section. The increase of the turn-over rate in the direction of the epiphyses caused a higher intake of the fission product. In spite of the also increased decorporation from bone matrix, a higher concentration than in the centre of the femur diaphysis can today still be observed in this region.

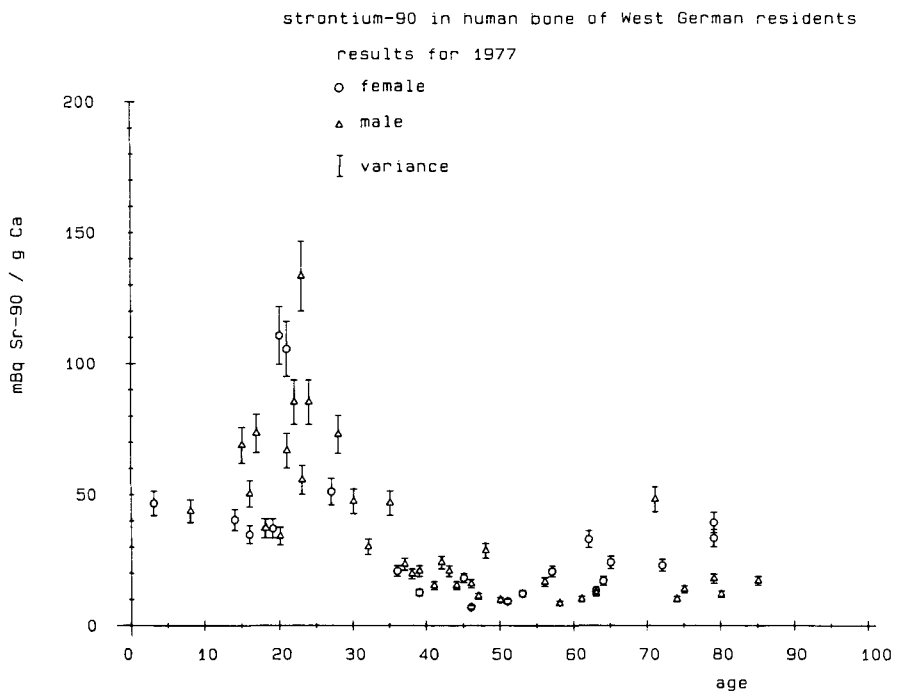
Current studies in bone samples, as, e.g., vertebra or ribs no longer report age-dependent concentration of strontium-90. Apparently the radionuclide has largely left the bone matrix on account of the increased turn-over rate in these bones.

REFERENCE

EML 1982 1982 EML Procedures Manual (HASL 300), 25th Edition Eds. H.L. Volchock, G.de Planque, Environmental Measurements Laboratory, U.S. Dept. of Energy, N.Y.



bone section	mBq Sr-90 pro g Ca	mBq Sr-90 pro mg Sr_{nat}	mg Srnat pro g Ca
1	23 ± 2	97 ± 10	0,24 ± 0,01
2	22 ± 2	98 ± 10	0,22 ± 0,01
3	16 ± 2	74 ± 7	0,21 ± 0,01
4	15 ± 2	73 ± 7	0,20 ± 0,01
5	13 ± 1	63 ± 6	0,20 ± 0,01
6	12 ± 1	56 ± 6	0,22 ± 0,01
7	13 ± 1	63 ± 6	0,21 ± 0,01
8	18 ± 2	80 ± 8	0,23 ± 0,01


strontium-90 in human bone of West German residents
results for 1977
female
male
variance
mBq Sr-90 / g Ca
200
150
100
50
0
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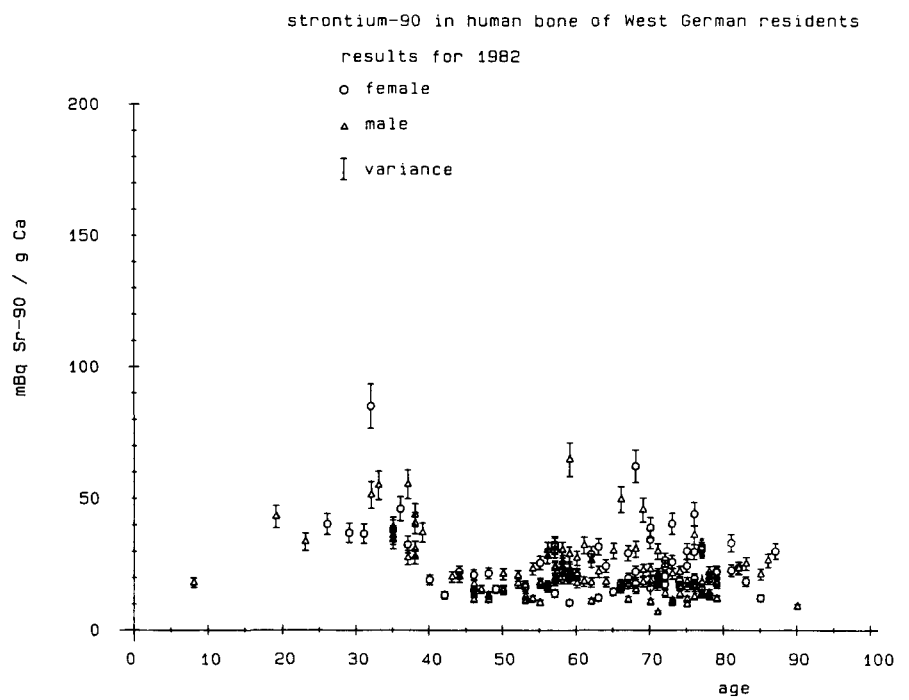
age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mBq Sr-90 / g Ca
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80
90
100
age

COMPARISON OF CRITICAL ORGAN AND EFFECTIVE DOSE EQUIVALENTS

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1. OBJEKTIVES

The earlier method B of limiting stochastic irradiation defects, based on ICRP Publication 2, required the evaluation of organ and whole body dose equivalents and the comparison to the respective annual limits in order to determine the limiting 'critical organ'. In 1977 it is recommended in ICRP Publication 26 to replace this method by the evaluation and limitation of the Effective Dose Equivalent, method A.

In order to compare these two methods of dose equivalent limitation both methods are applied to the radionuclides and radionuclide mixtures listed in Tab. 1, which are assumed to be released continuously through a 100 m stack and to cause irradiation of the surrounding population.

RADIONUCLIDE	CHARACTERISTIC NORMALIZED NUCLIDE SPECTRUM IN PERCENT					EXPECTED CHEMICAL SPECIES RELEASED	ASSIGNED LUNG CLEARANCE CLASS
	PWR AEROSOLS	α-ACTIVE WASTE	LONG LIVED IODINE	NOBLE GAS FR2	NOBLE GAS REPROCESSING PLANT		
Co-58	25	-	-	-	-	OXIDES AND HYDROXIDES	Y
Co-60	35	-	-	-	-		Y
Sr-90	1	-	-	-	-	OXIDE	D
Cs-134	10	-	-	-	-	OXIDES AND HYDROXIDES	D
Cs-137	25	-	-	-	-		D
Ce-144	4	-	-	-	-		Y
Pu-238	-	44.7	-	-	-	OXIDES	Y
Pu-239	10^{-2}	2.8	-	-	-		Y
Pu-240	-	4.5	-	-	-		Y
Am-241	-	5.5	-	-	-		W
Am-242m	-	0.1	-	-	-		W
Am-243	-	0.3	-	-	-		W
Cm-242	-	3.9	-	-	-		W
Cm-244	-	38.2	-	-	-		W
I-131	-	-	∿100	-	-	ELEMENTAL	D
Ar-41	-	-	-	∿100	-	-	-
Kr-85	-	-	-	-	∿100	-	-
RELEASE RATE in Bq/a	$3.7 \cdot 10^{10}$	$3.7 \cdot 10^{9}$	$3.7 \cdot 10^{10}$	$3.7 \cdot 10^{15}$	$3.7 \cdot 10^{16}$		

TAB. 1: RADIONUCLIDE GROUP CHARACTERISTICS.

2. METHODS

Atmospheric dispersion and deposition of radionuclides on the ground as well as their transfer to the man via the various exposure pathways is calculated in both methods according to /1/.

Method A: The Committed Effective Dose Equivalent $H_{50,E}$ was evaluated corresponding to the following equation

$$H_{50,E} = \sum_T W_T \cdot \sum_T H^i_{Ex,T} + H^i_{50,T}$$

$H^{i}_{50,T}$ total committed dose equivalent in tissue (T) resulting from intake of radionuclide (i) (integration time: 50 years) in Sv

$H^{i}_{Ex,T}$ total dose equivalent in tissue (T) resulting from external irradiation of the radionuclide (i) in Sv (accumulation time of ground deposition: 50 years)

W_T weighting factor representing the ratio of the stochastic risk resulting from tissue (T) to the total risk when the whole body is irradiated uniformly, see Tab. 2.

Dose factors for intake of radionuclides have been taken from ICRP 30. The list of dose factors quoted in ICRP 30 has been completed in order to have a set of 20 organ dose factors per radionuclide for both inhalation and ingestion. Dose conversion factors published in reference /2/ have been used to determine dose factors for external γ-irradiation from the ground, taking into account radioactive decay chains. The limit of the effective dose equivalent is assumed to be the same as the whole body dose equivalent limit (300 μSv) specified in the German Radiation Protection Ordinance.

Method B: Dose factors for intake and external irradiation have been taken from reference/1/. Integration and accumulation times are the same as above. Dose equivalent limits refer to the German Radiation Protection Ordinance.

As ICRP 30 quotes dose factors for adults only, all dose calculations are restricted to adults, incl. the thyroid dose by iodine ingestion.

3. RESULTS

The dose equivalents calculated according to method A and B are summarized in Tab. 2. The five organs out of the remainder tissues, which are accounted for, are underlined.

4. CONCLUSIONS

1. The differences of organ dose equivalents in methods A and B are due to
   - improved metabolic data,
   - more detailed metabolic models.

2. Effective dose equivalents, related to the annual limit (method A) are smaller in the examples shown than the respective critical organ dose equivalents, related to their annual limit (method B).
   That means: method B was more conservative.

The difference ist most pronounced in the case of thyroid irradiation by incoporated iodine and of isolated skin irradiation, e. g. by Kr-85, due to the small W_T of the thyroid and the skin. The differences in external irradiation are due to the fact, that the self-shielding of the inner organs is taken into account in method A.

/1/ 'Allgemeine Berechnungsgrundlage für die Strahlenexposition bei radioaktiven Ableitungen mit der Abluft ...' GMBL, Ausgabe A, 30 369 (1979) and GMBL, Ausgabe A, 33 735 (1982)

/2/ D. C. Kocher:
"Dose Rate Conversion Factors for external Exposure to Photon and Electron Radiation from Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities", Health Phys. 38, 543 (1980)

EXPOSURE PATHWAY		INHALATION		INGESTION		γ-IRRADIATION FROM THE GROUND		INHALATION		INGESTION		γ+β-SUBMERSION			
MIXTURE		P W R - A E R O S O L						α-ACTIVE WASTE		J-131		Ar-41		Kr-85	
METHOD	W_T	A µSv	B µSv	A µSv	B µSv	A µSv	B µSv	A µSv	B µSv	A µSv	B µSv	A µSv	B µSv	A µSv	B µSv
GONADS	0.25	.02	-	4.1	-	3.5	-	5	-	.02	-	80	-	2.3	-
BREAST	0.15	.04	-	3.7	-	4.6	-		-	.05	-	85	-	5.1	-
RED BONE MARROW	0.12	.08	-	5.7	-	4.7	-	34	-	.04	-	72	-	1.9	-
LUNG	0.12	.74	.51	3.7	0.8	4.3	-	68	15	.05	-	72	-	1.9	-
THYROID	0.03	.04	-	3.7	-	3.8			-	213	230	90	-	2.5	-
BONE SURFACE	0.03	.43	-	7.6	-	5.1	-	422	-	.04	-	77	-	2.5	-
BONE		-	.93	-	28.8	-	-	-	406		-	-	-	-	-
GASTRO INTESTINAL TRACT		-	.03	-	5.5	-	-	-			4	-	-	-	-
SKIN	0.01	.02	-	2.9	-	5.5	-		-	.04	-	154	182	202	197
REMAINDER	0.3														
ADRENALS		.06	-	4.6	-	-	-		-	.03	-	68	-	1.9	-
BLADDER		.02	-	4.3	-	4.0	-		-	.02	-	-	-	-	-
STOMACH		.05	-	4.2	-	4.1	-		-	.14	-	66	-	1.8	-
SMALL INTESTINE		.03	-	4.7	-	3.5	-		-	.02	-	62	-	1.5	-
UPPER LARGE INTESTINE		.03	-	5.1	-	4.2	-		-	.02	-	72	-	1.8	-
LOWER LARGE INTESTINE		.03	-	6.2	-	3.3	-		-	.02	-	66	-	1.7	-
KIDNEYS		.04	.09	4.1	2.9	3.8	-		49	.02	-	72	-	1.9	-
LIVER		.14	.16	4.2	8.8	3.9	-	95	64	.02	-	68	-	1.8	-
PANCREAS		.06	-	4.0	-	3.8	-		-	.03	-	60	-	1.5	-
SPLEEN		.06	-	4.1	-	4.8	-		-	.03	-	66	-	1.7	-
THYMUS		.09	-	3.6	-	3.4	-		-	.14	-	76	-	2.0	-
UTERUS		.02	-	4.5	-	3.1	-		-	.02	-	60	-	1.5	-
EFFECTIVE DOSE EQUIVALENT		.15	-	4.6	-	4.2	-	32	-	6.4	-	78	-	4.5	-
WHOLE BODY DOSE EQUIVALENT		-	.05	-	5.8	-	6.8	-	12	-	0.4	-	137	-	3.2

CRITICAL ORGAN		LUNG		WHOLE BODY		WHOLE BODY		BONE		THYROID		WHOLE BODY		SKIN	
ANNUAL DOSE LIMIT OF ADULTS IN µSv	300	900	300	300	300	300	300	1800	300	1800	300	300	300	1800	
EFF.DOSE EQUIV. / ANNUAL DOSE LIMIT	.0005		.015		.014		.11		.02		.26		.015		
CRITICAL ORGAN DOSE EQUIVALENT / ANNUAL DOSE LIMIT		.00057		.019		.023		.23		.13		.46		.11	

TAB. 2: DOSE EQUIVALENTS RELATED TO DOSE EQUIVALENT LIMITS.
Dose contributions smaller than 1 ‰ are omitted.

LIFE-SPAN RADIATION EFFECTS STUDIES IN ANIMALS: WHAT CAN THEY TELL US?*

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This presentation bears the same title as the 22nd Hanford Life Sciences Symposium, which was held in Richland, Washington, U.S.A., in September of 1983. The complete Proceedings of that Symposium, edited by Roy C. Thompson and Judith A. Mahaffey, and published by the U.S. Department of Energy as a volume in their DOE Symposium Series, should be available before the end of 1984. This presentation is a personal reaction to the topic of that Symposium, and to some of the papers and discussions heard at that Symposium.

In the absence of data on effects in humans from exposure to low doses of ionizing radiation, life-span studies in a variety of animal species, employing a variety of radiations sources, have been funded by several U.S. government agencies, and by agencies of other governments. These studies employ large numbers of animals, some of which may live for 15 years or longer; they represent a substantial investment of time and money. Despite this effort, the results from these studies have found relatively little application in the development of radiation risk factors for various organs of man, as promulgated by such groups as the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), the International Commission on Radiological Protection (ICRP), or in the United States by the National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation (BEIR). Are these animal data of so little value? Have we been doing the wrong experiments? Have we not learned how to most appropriately interpret our data? Our Symposium last year, in Richland, was convened to explore these kinds of questions.

One might reasonably question whether those attending our Symposium constituted a totally unbiased or representative group. There was a preponderance of attendees from the Western U.S. laboratories engaged in life-span studies in beagle dogs; however, small animal studies were also represented, including the extensive French studies. In all, there were approximately 150 registrants from 7 countries, representing more than 30 laboratories and funding agencies. To focus attention on the interpretation of data, and to insure an objectivity that might otherwise have been suspect, we employed a panel of three eminent statisticians who critiqued papers as they were presented and led summary discussions. This proved to be a generally acclaimed feature of the Symposium, and the discussions will be included in the published Proceedings. The statisticians on this panel were: Dr. Marvin A. Schneiderman of Clement Associates, Arlington, Virginia -- for many years at the U.S. National Cancer Institute; Dr. Kenny S. Crump of Science Research Systems, Rustin, Louisiana; and Dr. Leon S. Rosenblatt of Geneticon, Inc., Walnut Creek, California.

The first session of the Symposium included 8 overview papers summarizing work in progress in major laboratories conducting life-span radiation effects studies. While not covering all such studies in all countries, a brief review of these papers will give some feel for the magnitude of this effort. Studies conducted in laboratories of the French Atomic Energy Commission since 1968,

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mostly in rats, have concentrated on inhalation exposures to radon and radon daughters, and to actinides, and to external neutron irradiation. A study in progress at Colorado State University, Fort Collins, Colorado, has employed nearly 2,000 beagle dogs exposed at 6 ages of pre- and post-natal life to ^{60}Co gamma radiation. Current life-span studies at Argonne National Laboratory, Argonne, Illinois, involve mice and beagle dogs in comparative studies of gamma and neutron irradiation under various temporal exposure patterns. At the University of California, at Davis, California, beagles have been exposed chronically to ingested ^{90}Sr and injected ^{226}Ra, and externally to X-ray and ^{60}Co gamma irradiation. Single dose, intravenous injection has been the exposure mode for studies involving more than 1000 beagle dogs at the University of Utah, Salt Lake City, Utah, where study of the comparative toxicity of ^{239}Pu and ^{226}Ra has been the principal objective. At Pacific Northwest Laboratory, in Richland, Washington, we have underway experiments involving the inhalation exposure of more than 500 beagles to plutonium, and more than 2,000 rats to radon and radon daughters; also a very low-level exposure experiment involving the inhalation of $^{239}PuO_2$ by nearly 3,000 rats. Other inhalation studies with beagle dogs are conducted at the Lovelace Inhalation Toxicology Research Institute, Albuquerque, New Mexico, involving exposure to fission product beta-gamma emitters as well as plutonium.

All of these studies have the primary objective of evaluating risks to humans, in areas where direct human data are unavailable. Some of these objectives include:

- evaluation of the bone-cancer risks of plutonium exposure by comparing plutonium/radium toxicity ratios in the dog to the known human risks from radium;
- evaluation of the liver-cancer risks of plutonium exposure by comparing plutonium/thorium toxicity ratios in the dog to the known human risks from thorotrast;
- evaluation of the lung-, bone-, and liver-cancer risks of inhaled plutonium by exposure of several animal species to aerosols of various plutonium compounds;
- clarification of the risks to uranium miners by experimental study of the comparative effects of various components of uranium mine atmospheres;
- evaluation of the comparative risks of exposure to neutrons -- particularly significant in view of the virtual elimination of neutron information in the re-evaluation of Hiroshima-Nagasaki dosimetry;
- evaluation of age effects on the sensitivity to radiation of various types -- particularly important to the evaluation of population effects.

Attainment of these objectives involves unavoidable uncertainties of extrapolation between animal species, and extrapolation from experimentally accessible doses to the much lower doses of primary concern in human exposure. Methods of performing these extrapolations, and of estimating the uncertainties involved in these methods, were the subject of much of the Symposium discussion.

It has seemed to me that these life-span animal studies can be ultimately useful in one, or all, of three basically different ways. First, one might hope to accumulate sufficient data in a variety of species to convince oneself

that these data could be directly applied to man, or applied with some mechanistically, physiologically, or other confidently derived correction factor. We seem to have some prospect of achieving this objective for a few of the most studied radionuclides, like plutonium, and radon and daughters. There are always some aberrant species, however. As reported in our Symposium, burros seem immune to any kind of radiation induced cancer, as are hamsters, under many circumstances, to lung cancer.

A second way of utilizing the results of life-span animal data would be to employ the extensive and detailed information, potentially available from such experiments, in acquiring a basic understanding of mechanisms of radiation damage. Thus, the aberrant burro and hamster data just mentioned could be most useful if they led to an understanding of why these species behave differently from others. Such understanding might enable us to predict, with some assurance, whether man would be apt to behave more like the burro or more like the beagle. The potential data from these studies affords many such exciting opportunities. Unfortunately, the cost of the basic study, with its original narrowly focussed objective, is so large that funding agencies are loath to supply the additional funds required to explore these ancillary possibilities. This problem was repeatedly voiced at the Symposium, and the desirability of intensive and cooperative use of the experimental resource was stressed, perhaps through funding of different aspects of the study by different agencies.

Still a third useful purpose may be served by these studies, even though the data may not permit confident extrapolation to the human, and may afford little in the way of basic understanding. Such a scientifically disappointing study may nevertheless offer essential reassurance to the decision maker, who has no alternative but to make some kind of decision, and who can at least point to the fact that an effort was made to study the problem, and that some qualitative assurance is afforded by the observation of no effects, or no unexpected effects. The political importance of such qualitative observations should not be underestimated. They have provided a sufficient basis for most past decisions, and I have the feeling that our current emphasis on statistical precision may often obscure the more important political and human issues.

Undoubtedly, the most important contribution of the Symposium consisted of the new ideas and new approaches provided to the participants by both the authors of the papers presented and by the discussions with the statistics panel. It is hardly possible to summarize these, but I would like to mention a few that impressed me.

If there could be said to be a single prevailing theme it might have been the importance of not neglecting the factor of time in the evaluation of dose-response relationships. The day has past when raw incidence data can be reported, even as preliminary data, without raising critical questions of when, in relation to exposure, the effect occurred. It not only makes an obvious difference to the animal, or human, whether he dies early or late; but it is critically important to any mechanistic modeling of the effect, or any statistical interpretation of uncertainties. It would also seem to me that this same criticism can be directed at most current formulations of cancer risk to man, which are couched in such terms as cancer deaths per million man rem, ignoring the time factor. A measure of years of life lost per million man rem would seem a more pertinent measure of detriment, and a better basis for the establishment of permissible exposure limits.

Many cautions were expressed concerning the myriad of unforeseen and possibly unappreciated biases that may influence an experimental result. These become of increasing concern as the effects to be evaluated become smaller and smaller relative to those observed in controls, and as the period of observation becomes longer. The lesson to be learned is not that all such biases can be avoided -- some that we might know how to avoid today, were not as well appreciated 10, 15, or 20 years ago when some experiments still in progress were started. And often we have only a choice of which bias to accept; for example, do we bias our female beagle results by removing mammary glands before exposure, by removing mammary tumors as they develop, or by losing animals through early deaths from spontaneous mammary tumors? The important thing is that these biases be recognized, and acknowledged, and allowance made for them in the interpretation of results.

It was suggested on several occasions that it might be useful to establish some basic statistical guidelines -- a minimum set of procedures that would be required in any analysis of life-span data. These suggestions were not followed up, however, and there were contrary opinions to the effect that no constraints should be placed on the scientist's judgement regarding the proper approach to the analysis of his particular data. A corollary to this latter position, however, should be that the collected raw data be available to anyone wishing to employ a different analytical approach.

The problem of unappreciated bias is particularly prevalent when, as often happens in these studies, data are interpreted by persons or groups not involved in the collection of the data. The cooperation of diverse groups in such interpretation is certainly to be encouraged -- there were several examples of it in our Symposium -- but every effort should be made to involve the data collector in such interpretations. It was stressed that no matter how detailed the published data, there is always information that remains unpublished, and is known only to the experimenter -- or, increasingly I fear, may be known only to the experimenter's computer, which is, unfortunately, not smart enough to recognize the problem.

There seemed general agreement that discussions such as those held at this Symposium were useful, and that persons engaged in life-span radiation effects studies should meet at regular intervals to compare progress and procedures. This would seem particularly appropriate in the case of the U.S. beagle studies, which in many respects can be thought of as separate facets of a single experiment.

ETUDE EXPERIMENTALE DE LA RADIOSENSIBILITE DES DIFFERENTES POPULATIONS CELLULAIRES AUX EFFETS CARCINOGENIQUES DES NEUTRONS DE FISSION

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Une partie de notre programme de recherche expérimentale sur l'induction de cancers par irradiation concerne les irradiations globales par des neutrons de fission à TEL élevé. Ce travail nous a permis d'étudier la radiosensibilité des différents organes chez le rat, ainsi que pour certains d'entre eux des différentes lignées cellulaires qui les composent.

I - SCHEMA EXPERIMENTAL

Ce schéma a été décrit dans une publication précédente (LAFUMA, MORIN, MASSE 1982). 1 370 rats mâles Sprague-Dawley âgés de 3 mois ont été irradiés et comparés à 606 rats témoins. Les doses d'irradiation varient de 0.01 Gy à 8 Gy. Les techniques d'histopathologie sont celles utilisées dans nos laboratoires habituellement (LAFUMA, 1974) et la classification des tumeurs est faite en se référant à celle de l'ICCD.

II- RESULTATS EXPERIMENTAUX

1. Durée de vie

La diminution de la durée de vie dépend de la dose d'irradiation ; nulle à 0,016 Gy, elle est de 50% à 4,4 Gy et de 68% à 8 Gy (MORIN, 1983).

2. Incidence des Cancers et Radiosensibilité des Organes

L'incidence croît de façon différente avec la dose selon les organes (MORIN, 1983). Un doublement du taux des cancers apparaît dès 0,016 Gy (39,3 % + 13,7% de cancers thyroïdiens) par rapport aux rats témoins (17,5% + 13,8% de cancers thyroïdiens).

La radiosensibilité des organes est calculée en faisant le rapport entre la fréquence (en pourcentage) de cancers apparus par organe chez les rats irradiés et la fréquence de cancers d'un même organe chez les rats témoins.

Les organes les plus sensibles à l'irradiation neutronique sont les poumons et l'os, organes pour lesquels le nombre de cancers constatés est multiplié par plus de 15 par rapport aux témoins. Apparaissent ensuite avec une radiosensibilité décroissante : le système urogénital, les glandes mammaires, la peau et les vaisseaux sanguins. Le système digestif (tractus digestif, foie, pancréas), la surrénale et le système hématopoïétique sont beaucoup moins radiosensibles.

Tableau 1 - Radiosensibilité des différents organes

	Fréquence des cancers (%)		Rapport
	Témoins 606 rats	Neutrons 1.370 rats	Neutrons/Témoins
Poumon	0,83	13,1	15,8
Os	0,33	5,3	15,1
Urinaire	0,83	7,5	9,0
Génital	0,33	2,4	7,3
Mammaire	0,33	2,5	7,6
Peau	1,80	11,7	6,5
Vaisseaux sanguins	1,32	8,4	6,4
Tissus mous	3,00	15,0	5,0
Surrénale	1,65	4,4	2,7
Digestif	1,30	3,7	2,8
Hématopoïétique	2,15	4,1	1,6
Thyroïde	13,90	7,7	0,6
Nerveux	1,20	0,9	0,75

3. Cinétique

On constate une accélération du processus d'apparition des cancers en fonction de la dose (MORIN, 1983).

Les organes les plus marqués par un raccourcissement des médianes d'apparition des cancers sont le système hématopoïétique et l'os.

La radiosensibilité d'un organe n'est pas forcément la même pour toutes ses lignées cellulaires. Nous avons comparé dans un même organe des lignées différentes pour voir si leur évolution vis à vis de l'apparition des cancers est ou non identique.

Tableau 2 - Raccourcissement des Médianes d'Apparition des Cancers (jours)

	Témoins 606 rats	Neutrons 1.370 rats	Neutrons/Témoins
Sang	764	396	0,52
Os	853	463	0,54
Poumon	867	543	0,63
Vaisseaux sanguins	746	473	0,63
Urinaire	852	570	0,67
Digestif	781	541	0,70
Tissus mous	663	473	0,71
Génital	711	514	0,72
Thyroïde	827	593	0,72
Peau	688	504	0,73
Mammaire	720	537	0,75
Surrénale	770	622	0,81
Nerveux	604	489	0,81

4. Etude des Différentes Lignées Cellulaires

L'OS : Chez le rat, les tumeurs osseuses sont en majorité des ostéosarcomes ostéogéniques (plus de 60%).

Les ostéosarcomes télangiectasiques et les chondrosarcomes apparaissant aussi de façon significative. Les variations des doses d'irradiation ne semblent avoir aucun effet ni dans la répartition de ces diverses formes, ni dans leur localisation : pattes 57% , côtes 22% , vertèbres 8% , tissus mous 8% , crâne 5% ; on peut comparer ces résultats à ceux obtenus avec des actinides : pattes 78%, côtes 8%, vertèbres 14%.

Tableau 3 - Pourcentage de Cancers pour Différentes Lignées Cellulaires

	Poumons*		Hématopoïétique*		Surrénale*		Thyroïde*				
	1	2	1	2	1	2	1	2	3	4	5
Témoins	20	80	67	33	60	40	66	4	22	8	0
Neutrons											
0,016	67	33	100	0	72	28	66	17	17	0	0
0,08	60	40	60	40	67	33	67	11	0	22	0
0,4 (s)	57	43	50	50	75	25	63	12	25	0	0
0,4	55	45	50	50	58	42	88	0	0	12	0
1,5	26	74	83	17	33	67	58	14	0	14	14
2 (s)	36	64	0	100	50	50	60	20	0	20	0
2,3	32	68	75	25	63	37	100	0	0	0	0
2,6	71	29	100	0	100	0	100	0	0	0	0
3,5	54	46	100	0	100	0	100	0	0	0	0
4,4	40	60	100	0	100	0	25	50	0	25	0
5,6	50	50	0	100	0	0	0	0	0	0	0
8	50	50	33	67	0	0	0	0	0	0	0
Actinides	52	48	67	33							

* Poumons - 1 Bronchioloalvéolaire, 2 Epidermoïde
Hématopoïétique - 1 Lymphosarcome, 2 Leucémie myéloïde
Surrénale - 1 Corticosurrénale, 2 Médullosurrénale
Thyroïde - 1 Vésiculaire et Trabéculaire, 2 Papillaire, 3 Médullaire, 4 Indifférencié, 5 Epidermoïde.

Des effets dûs à la nourriture (biphényls) interviennent de façon prépondérante dans l'apparition des cancers de la thyroïde ; l'apparition de ces cancers est tardive, l'irradiation les accélère mais la forme médullaire n'existe que chez les témoins ou les très faible doses.

III- CONCLUSION

La radiosensibilité des lignées cellulaires à l'intérieur des organes montre une réponse à l'irradiation relativement comparable, donc on peut ne pas tenir compte des différentes lignées cellulaires en protection radiologique.

Comme chez l'homme, la lignée médullaire thyroïdienne ne semble pas sensible à l'irradiation ; les cancers du système nerveux semblent aussi indépendants de l'irradiation.

Le système hématopoïétique présente une divergence d'évolution entre la fréquence et la vitesse d'apparition des cancers en fonction de la dose d'irradiation. La vitesse d'apparition est très accélérée et si au lieu de garder les rats jusqu'à leur mort naturelle on avait fait des sacrifices précoces, on aurait pu avoir l'impression fausse que le système hématopoïétique était un système très radiosensible.

Les deux organes les plus sensibles à l'irradiation par des neutrons de fission chez le rat sont l'os et le poumon.

REFERENCES

- LAFUMA J. et al, 1974. Etude expérimentale des polluants radioactifs inhalés. In : Réactions broncho-pulmonaires aux polluants atmosphériques, Pont-à-Mousson, Paris, INSERM, pp. 307-324.

- LAFUMA J. , MORIN M. , MASSE R. , 1982. Cancer induction in rats after fission neutron irradiation with special emphasis on lung cancers. In : Proceedings European Seminar on Neutron carcinogenesis, 57-73,EUR 8084 EN, Luxembourg.

- MORIN M. , MASSE R. , LAFUMA J. , 1983. An experimental study on fission neutron carcinogenesis. In : Twenty-Second Hanford Life Sciences Symposium - September 27-29, 1983, Richland, Washington.

INHALATION TOXICOLOGY OF $^{144}CeCl_3$ IN THE BEAGLE DOG

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INTRODUCTION

To ensure that radiation protection guidelines provide adequate protection against long-term health effects from internally deposited radionuclides, it is necessary to know what organs and tissues incur the highest risks and how different physical, chemical, and biological factors may alter the dose-response relationships. Data from exposed groups of people provide a critically important foundation for our understanding of radiation risks but they do not provide all of the necessary information required to address the above concerns. To provide this information for inhaled fission product and actinide aerosols, life-span studies are being conducted using Beagle dogs and other species at the Lovelace Inhalation Toxicology Research Institute (ITRI).

The fission product ^{144}Ce was chosen for study because of its prevalence in irradiated fuel of operating nuclear reactors, the energetic beta emissions from its short-lived daughter, ^{144}Pr, and its relatively long physical half-life, 284 days. Life-span studies were conducted using dogs that inhaled ^{144}Ce in a relatively soluble form, $^{144}CeCl_3$, and a group of control dogs. All of the $^{144}CeCl_3$-exposed dogs are dead; this report gives a brief summary of the results obtained to date.

MATERIALS AND METHODS

Two groups of dogs were used in these studies; 70 dogs were assigned to the life-span dose-response study and 27 dogs were assigned to a parallel, serial sacrifice study of the radiation dose distributions. All dogs assigned to the dose-response study were purebred Beagles 12 to 14 months of age and weighing between 6.7 and 13 kg at time of exposure. Of the 70 dogs, 55 were exposed to ^{144}Ce and 15 served as controls. The exposure aerosol was produced by nebulizing a solution containing ^{144}Ce in a 0.7 to 1% CeCl solution in a 0.1 or 1.0 N HCl solution. The resulting aerosol had an activity median aerodynamic diameter of 1.5 to 2.4 μm and σ_g = 1.6 to 2.1. After a brief, nose-only inhalation exposure (4 to 28 min), each dog was whole-body counted to determine the initial body burden. Whole-body counting was used at periodic intervals in these studies to determine the patterns of long-term retention. After 20 to 60 days, the dogs were transferred from individual metabolism cages to a kennel facility where they were housed two per run by sex. The health status of each dog was evaluated periodically throughout the dog's life and illnesses not associated with the radiation exposure were treated using standard veterinary practices. All dogs were maintained until they died or were euthanized. Complete necropsies were performed. Tissue specimens collected for histopathologic examination were embedded in paraffin, sectioned at 5 μm, and stained with hematoxylin and eosin routinely as well as special stains where appropriate. The clinical and pathologic findings were used to establish the cause of death and occurrence of major diseases.

Dosimetry calculations were estimated on the organ-average absorbed beta dose from ^{144}Ce-^{144}Pr. The small gamma contribution was ignored. Dosimetry informa-

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tion obtained from dogs that were serially sacrificed over the first 512 days after exposure in the dose distribution study (1) were used along with the whole-body counting data from each exposed dog in the life span study to determine organ doses individually (2).

The lifetime risk of cancer was calculated from the exposed dogs that lived more than 2 years after inhalation exposure. The risk for a given type of cancer was first computed on an annual basis by dividing the number of cancers of that type seen in a given year by the sum of all absorbed doses to that organ for all dogs living at the beginning of the year being calculated. This result was corrected for the observed incidence of that cancer in unexposed control dogs and for survival of exposed dogs to that time after exposure. These annual risks were summed to obtain the lifetime risk. The variance of the annual risk for exposed and control dogs was estimated for each year using the bionomial distribution. The square root of the sum of the variances of the annual risks was used to construct the 95% confidence interval for lifetime risk.

RESULTS

The whole-body retention pattern for ^{144}Ce inhaled as $^{144}CeCl_3$ was similar to that seen for other inhaled materials, with a rapid drop in body burden occurring during the first 2 days as the ciliated portions of the respiratory tract were cleared by mucociliary activity by way of the oropharynx to the gastrointestinal tract. Because of the low solubility of ^{144}Ce in the gastrointestinal tract, most of this material passed through unabsorbed and was excreted in feces. The long-term retained burden (LTRB) ^{144}Ce remaining after early clearance was complete, represented, on the average, 43% of the initial body burden. The LTRB had a whole-body retention half-life that increased with time after exposure and approached the physical half-life of ^{144}Ce. Although the dogs were originally entered into the study in defined exposure groups, there was sufficient variability among the achieved long-term retained burdens that they formed a continuum ranging from 0.096 to 13.3 MBq/kg body weight. For dogs that lived more than 5 years, the cumulative absorbed beta doses were 24, 60, 18 and 92 Gy per MBq ^{144}Ce LTRB/kg body weight for lung, liver, skeleton and nasal cavity, respectively (1,2).

Fourteen dogs exposed at the highest levels died of acute effects during the first two years. Eight of these died from bone marrow aplasia during the first 44 days after exposure, and 6 had pulmonary, hepatic or marrow injury that caused death at 138 to 510 days after exposure. Except for one case of osteosarcoma in a vertebra, deaths with cancer in various organs did not begin until 4.5 years after exposure. All ^{144}Ce-exposed dogs and all but one control dog are dead as of 9/30/83. A summary of important late biological effects is given in Table 1.

DISCUSSION

The biological effects seen in this study reflect the patterns of deposition, retention and dosimetry of ^{144}Ce after inhalation in a relatively soluble form. During the first week after inhalation exposure, after rapid clearance of the naso-pharyngeal and tracheobronchial regions by way of the gastrointestinal tract, the pulmonary-deposited ^{144}Ce constituted the main part of the long-term retained burden. Most of this ^{144}Ce was translocated during the first week, primarily to the liver and skeleton. High local concentrations were also noted in the nasal turbinates, either from ^{144}Ce deposited there at the time of exposure or translocated there after absorption from the pulmonary region (1,4).

Although the most prevalent early effect of inhaled ^{144}Ce was bone marrow aplasia in the dogs exposed at the highest level, early effects were also seen in the lung and liver. Dogs that survived this early phase had non-eventful clinical

TABLE 1. Summary of Biological Effects Seen in Dogs that Inhaled $^{144}CeCl_3$ or the Associated Control Dogs

Diagnosis	Number[a]	Death Days PE	Cumulative Dose to Organ Involved, Gy
Early Effects (< 2 yr.)			
Exposed Dogs			
Pulmonary Injury	3	138-375	86-170
Hepatic Injury	2	309-336	160-190
Bone Marrow Aplasia	9	21-510	6.1-58
Late Effects (> 2 yr.)			
Exposed Dogs			
Neoplastic Disease			
Lung	4	1632-5139	11-62
Liver	10	1759-5485	10-240
Skeleton	1	799	81
Sinonasal	7	1632-4085	40-250
Bone Marrow	2	1806-1811	9.4-37
Other	8	2935-5498	
Non-neoplastic Disease	11	874-5120	
Control Dogs			
Neoplastic	7	2545-5366	0
Non-Neoplastic	7	3065-5974	0
Alive	1		0

[a]One dog had both a pulmonary adenoma and squamous cell carcinoma of the nasal cavity and one dog had both a bronchial adenocarcinoma and a squamous cell carcinoma of the nasal cavity.

courses except for a single case of vertebral osteosarcoma until cancers began to appear beyond four years after exposure.

Cancers or benign neoplasms were observed in all four organs or tissues receiving the largest relative doses: lung, liver, bone, and nasal epithelium. In the lung, 3 adenocarcinomas and an adenoma were seen. In the liver, there were 7 hemangiosarcomas, 1 hepatocellular carcinoma, 1 bile duct carcinoma and 1 bile duct cystadenoma. One osteosarcoma of bone was seen. Two cases of myelogenous leukemia were observed. In the nasal cavity, 1 hemangiosarcoma and 6 squamous cell carcinomas were seen. One of these latter tumors was found *in situ*. Cancers found in other organs were also observed in control dogs and are currently assumed to not be associated with the radiation exposures.

The malignant tumors listed above were used to calculate lifetime risk of cancer in each organ or tissue (Table 2). The risk of cancer from ^{144}Ce in the dog's liver is 2 to 3 times greater than that in the lung or sinonasal epithelium. The calculated bone cancer risk is the lowest value in Table 2, but it may also be least reliable because only one bone cancer was observed.

TABLE 2. Comparison of Estimated Lifetime Risks for Chronic Irradiation from Internally-Deposited ^{144}Ce in Dogs and People

Organ or Tissue	Lifetime Cancers/10^4 Gy	
	ICRP	Dogs[a]
Lung	20	34 (4-64)
Bone Marrow	20	27 (0-66)[b]
Liver	10	84 (61-107)
Sinonasal	10?	31 (16-46)
Bone	5	14 (4-42)[b]

[a]Lifetime risk (95% confidence interval)

[b]Based on average bone dose

How do these data apply to possible human exposures? The current ICRP exposureguidelines for internally-deposited radionuclides account for a number of different organs that may be at risk, depending upon the dose patterns associated with the radionuclide form inhaled (5). For this approach to be successful, it is necessary to estimate both the risk of cancer in each heavily irradiated tissue and the doses received by each of these tissues. Currently, dosimetry calculations are not available for the sinonasal area. Analyses of all sinonasal cancers seen in dogs exposed to beta-emitting radionuclides in a soluble form at the ITRI have shown that use of the average skeletal dose to approximate dose to nasal tissues is not a useful method especially for lanthanide radionuclides (3). This is an area that needs further attention by standards-setting bodies.

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REFERENCES

1. Boecker, B. B. and Cuddihy, R. G. (1974): *Radiat. Res.*, 60, 133.
2. Benjamin, S. A., Boecker, B. B., Chiffelle, T. L., Hahn, F. F., Hobbs, C. H., Jones, R. K., McClellan, R. O., Pickrell, J. A. and Redman H. C. (1973): Radionuclide Carcinogenesis, 181, AEC Symposium Series 29, CONF 720505, NTIS.
3. Boecker, B. B., Hahn, F. F., Cuddihy, R. G., Snipes, M. B. and McClellan, R. O. (1983): To be published in the Proceedings of the Twenty-Second Hanford Life Sciences Symposium, "Life-Span Radiation Effects Studies in Animals: What Can They Tell Us?", Richland, WA, September 27-29, 1983.
4. Benjamin, S. A., Boecker, B. B., Cuddihy, R. G and McClellan, R. O. (1979): *J. Natl. Cancer Inst.*, 63, 133.
5. International Commission on Radiological Protection (1979): *Ann. ICRP*, 2, 1.

RADIOBIOLOGICAL EFFECTS OF TRITIUM IN MICE *)

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INTRODUCTION

Tritium, the radioactive isotope of the most ubiquitious element hydrogen, has access to all living organisms in form of tritiated water or along the food chain,inherent in essential organic molecules.Incorporated as precursors for long lived biochemical compartments as amino acids for proteins the efficiency of radiation may be by a factor of 4 higher compared to HTO or when incorporated into genetically important substances such as DNA by a factor of 10 to 50 /FEI 78/, /STR 82/.

Besides the limited experience from a few cases of fatal tritium incorporation by man only limited data on biological effects of tritium from animal experiments had been documented under the particular conditions of dose and dose rates corresponding to a single uptake of HTO. With regard to the special biological endpoints, body weight and mortality of offspring, there was a lack of information with studies below 10 MBq/g b.w. No data were available on the fertility or reproductive performance of male and female offspring following treatment of dams with HTO or other tritiated compounds on different gestation days. With these circumstances in mind, investigations of biological effects in pregnant mice were initiated.

METHODS

NMRI mice were caged in air-conditioned animal rooms at 23°C with artificial light from 6 a.m. to 6 p.a. All animals received standard diet and water ad libitum. Dams (P), 2.5 - 3 month old were mated between 9 - 11 a.m. The day after detecting vaginal plug was considered as day 1 post conceptionem (p.c.). The vaginal plug positive females were injected intra-peritoneal (i.p.) with the appropriate amount of tritiated water on days 5, 7, 9, 11 and 13 p.c. On day 19 p.c.or after delivery biological parameters as abnormalities or disorders of organs, body weight or organ weight variations and deviations in fertility and reproductive performance were analysed.

The build-up of dose and dose rate in pregnant mice follows exponential conditions with a biological half life of 1.8 days for the elimination as HTO, a value which was calculated on the basis of initial and terminal (on 18. day p.c.) tissue levels of tritium in dams injected on day 7. p.c.

Dose calculations were performed by equations from Jaeger and Hübner with an average energy of 1/3 of the maximum energy for tritium (18 KeV) /JAE 74/.

*) In memoriam Peter Török, 1981.

Incorporation of tritium by single injection during organogenesis resulted in an irradiation time period of 3,3 half lives due to biological regulated elimination of tritiated water. During that time 90% of tritium has been eliminated. 39% of total dose is delivered on the first day, the injection day, 65% in the first two days.Statistical evidence of the experimental data was shown by Student's t-test and its bivariate generalisation the Dunnett-test /DUN 55/.

RESULTS

Mortality in F_1:

The intra-uterine mortality rate was 27% after injection of 20 MBq/g b.w. on gestation day 7 (g.d.) 19% on g.d. 9 and 6% on g.d. 11. 60% with 30 MBq/g b.w. on g.d. 9 and 100% with 40 - 50 MBq/g b.w. on g.d. 7. Control animals showed an intra-uterine mortality of 4%. With a single injection of 5 MBq/g b. w. on gestation day 5, 7, 9, 11 and 13 mean litter size in treated dams (P-generation) was normal, compared to the controls. This rules out significant effects on the prenatal viability resp. intra-uterine mortality of the offspring by this amount of tritium. The smallest effective amount, where mortality effects could be detected was 20 MBq/g b.w. which corresponds to a total dose of 450 rad resp. 4,5 Gy. The next ineffective amount in our experiments was 5 MBq/g b.w. corresponding to a dose of 111 rad or 1 Gy.

Mortality in F_2:

Corresponding to the decreased fertility of F_1, some groups exhibited a lower number of F_2 than required for mortality studies. Therefore they were pooled. In groups consisting of approx. 100 or more suckling mice, no increased mortality could be verified. The smallest effective amount was 10 MBq/g b.w. the ineffective amount 5 MBq/g b.w.

Body weight:

HTO showed clear effects on body weight development in F_1 mice. The weight differences correlate to the prenatal injection day, the postnatal development state, the sex of the exposed embryo and the dose.

In adults the weight differences were less substantial in comparism to controls. But in one group (injected on day 9 p.c.) the weight of the females was significantly higher than that of control females of the same age. The smallest effective amount was 5 MBq/g b.w., whereas the ineffective amount seemed to be 2,5 MBq/g b.w.

Effects on organs:

After 4,5 months offsprings showed significantly decreased weight of brain and genital tract, depending on prenatal dose. 10 MBq/g b.w. to the dam, reduced the weight of gonads by 10 - 20% as compared to those of controls. The smallest effective amount for weight reduction of gonads and other organs seemed to be 2,5 MBq/g b.w., the ineffective amount was 1,25 MBq/g b.w.

Developmental effects:

The weight of terminal foetusses (on day 18.p.c.) in dams treated with 5 - 30 MBq/g b.w. was significantly decreased. A drastic stunting (cleft palate 51,3% of the controls) and other anomalies of mouse skeleton were detected.

The incidence rate of sceleton anomalies was low even in the experimental groups treated with 20 - 30 MBq/g b.w. The smallest effective amount of HTO was 20 MBq/g b.w. The ineffective amount was 10 MBq/g b.w.

Reproduction:

The reproductive performance of mice exposed in utero to a total dose of approximately 1 Gy (5 MBq HTO/g body weight) was investigated. Because of the rapid decrease of dose rate, due to rapid biological elimination of HTO with a biological half life of 1,8 days, the effectiveness of 5 MBq HTO/g b.w. was restricted to a relatively short period after the application.In contrast to treatment on days 9, 11 and 13, the injection of tritium on gestation days 5 and 7 did not influence the fertility. The prenatal exposure of females onward from gestation day 9 and of males onward from gestation day 11 resulted in a 50 - 100 percent decline of fertility, as compared to controls. Taking into account the substantially decreased fertility of both sexes, the administration of HTO on gestation day 13 had the greatest effect.

In the study on dose dependent effects, initial dose rates of 0.125, 0.5 and 1 Gy prenatal doses were focussed on the particularly sensitive day 13 p.c. The study showed that tritiated water from a dose of 0.5 Gy on effected the fertility of females and males as demonstrated by reduced litter size, lower percentage of pairs producing offspring and calculated from both of these,a reduction of the mean number of neonates per mated pair.

In other experiments adult offspring exposed in utero to 0.125 and 0.5 Gy radiation doses (injected activities: 0.6 and 2.5 MBq/g b.w.) were examined on fertility for longer periods of time. The irradiated mice were permanently mated during one year with similarly aged control mice or with each other. The reproduction in groups consisting of 20 - 25 separately caged pairs,is presented in Figure 1 as number of neonates per dam, as litter size and as frequency of pregnancies during the investigated period.

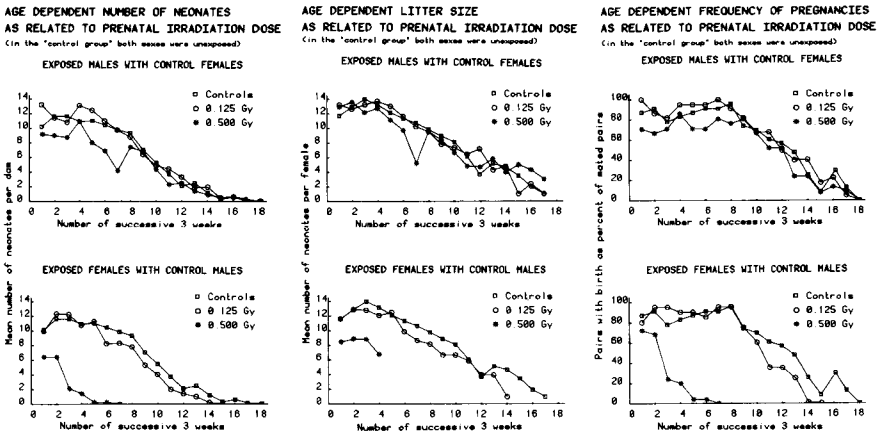

Figure 1 Reproductive capacity and reproductive period shown by mean number of neonates per dam,mean litter size and frequency of pregnancies in long-term reproduction experiments with tritium exposed male and female mice. (* = 0,500 Gy,o = 0,125 Gy, and □ = 0 Gy,control)

CONCLUSION

Single injections of HTO, even at activities causing stunting and perinatal death of litters (20 MBq/g b.w.) failed to be a potent inducer of gross malformations in foetus. However those prenatal doses, permitting survival of offspring, excerted severe consequences on the postnatal development. This could be explained by the relatively low dose rates of the injected radionuclide in our experiment.

At constant dose rates (chronical ingestion) the threshold dose of HTO was between 1.3 - 3.3 Gy /CAH 70/. In our experiments the 0.5 Gy from HTO showed similar effects on the reproduction of females as acute dose of 2 Gy x-rays,applied on gestation day 12,5 /RUS 59/.

The minimal dose affecting fertility of males is between 0.125 and 0.5 Gy and that of females is approximately 0.125 Gy. The threshold character of the 0.125 Gy dose is suggested by the minimal differences between fertility parameters of treated and control females, which became apparent only with aging /TÖR 80/.

REFERENCES

/CAH 70/ Cahill, D.F. et al.; Rad.Res.44,727-737,1970;.
/DUN 55/ Dunnett,C.W.; J.of Amer.Stat.Ass.50,1096,1955;
/FEI 78/ Feinendegen,L.E.;VI.Sym.on Microdos.Brüssel 1978;
/JAE 74/ Jaeger,R.B.,Hübner,W.;Thieme Verlag,Stuttgart 1974;
/RUS 59/ Russel,L.B.,et al.; Int.Conf.Venice, 1959;
/STR 82/ Streffer, C., Europ.Sem.;Mol, Belgien, 1982;
/TÖR 80/ Török, P. et al.; Int.Symp.,Neuherberg, 1980.

Radiation risk during prenatal development: Preimplantation period.

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The prenatal mammalian development is divided into three periods: The preimplantation period, the major organogenesis and the fetal period. Several studies have shown that the radio-sensitivity of the preimplantation mouse embryo is very high in general, but especially in the early 1-cell stage and somewhat less in the late 2-cell embryo. The major effect after irradiation of preimplantation embryos of different mammalian species is the early embryonic death. The embryo generally fails to implant into the uterus wall and organogenesis which follows the preimplantation period is lacking. We have investigated details of the radiation effect under very clear in vitro conditions (1, 2, 3).

Mouse embryos were cultivated in vitro (4) beginning from the 2-cell stage. Irradiation of 2-cell embryos either with X-rays (240 kV) or neutrons (7 MeV) was performed in the late G_2-phase. The following results were obtained:

1) Disturbances of blastocyst formation and a reduction of the number of hatched blastocysts occurred in the dose range between 0.5 and 1.0 Gy X-rays (5). In the case of neutron irradiation such damage of preimplantation development was already observed after a dose of 0.12 Gy. The RBE-values for blastocyst formation for doses of 0.12, 0.25 and 0.5 Gy were 9.0, 6.1 and 4.5 respectively.
2) Concomittant to the morphogenetic damage impairment of cell proliferation was found. At the end of the preimplantation period an X-ray dose of 1.0 Gy had caused a reduction of the average cell number to 67 cells per embryo in comparison to 100 cells in the unirradiated embryo. After neutron doses of 0.25 and 0.50 these values were 72 and 63 cells respectively (5).
3) Initially after irradiation a mitotic delay due to a G_2-block was

observed. An X-ray dose of 1.88 Gy induced a G_2-block of about 4 hours which was measured by cytophotometry. During the later preimplantation period after the morula stage cell death was the dominant event (5).

4) Cell death can be explained with the occurrence of chromosomal damage which led to hypoploid cells (DNA content of the cell nucleus: <diploid; cytophotometrically investigated). Hypoploid cells were found 1 and 2 cell divisions after irradiation in 4-cell and 8-cell embryos and in morulae (3 mitoses p.r.). The hypoploid cells lost their ability to divide after progression through 2-3 cell cycles post radiation (5).

5) Acentric chromosomal fragments were the reason for the formation of hypoploid cell nuclei. The fragments were lost from the cell nuclei during cell division. They were observed as micronuclei (6) in the cytoplasma of interphase cells or they were scored directly in metaphases with chromatid breaks (7). When compared with X-rays the higher frequency of cytogenetic damage after neutron irradiation explained the higher frequency of hypoploid cells and thus the stronger impairment of preimplantation cell proliferation and embryological development (8).
The development of micronuclei was studied during various stages of the embryos in dependance on the radiation dose, the number of cell cycles and other factors. The micronucleus test turned out to be a good assay for radiation damage on the genome and its cellular consequences.

6) Two different types of recovery processes were observed: intercellular and intracellular mechanisms. After X-ray doses of 1.0 and 2.0 Gy a small percentage of the embryos showed cell numbers at the end of the preimplantation period which were in the range of the control values (100 cells per embryo). Transplantation of these embryos into foster female mice proved that even after an X-ray dose of 2.0 Gy a complete recovery from radiation damage was possible. Some of the embryos developped to normal newborn mice which were free of gross malformations. Concluding from autoradiographic results it is assumed that during the preimplantation period an increased cell proliferation had compensated for the radiation induced cell death (intercellular mechanisms!).

When X- or neutron-irradiation was performed either in the late or in the middle of G_2-phase a higher frequency of acentric fragments and other chromosome aberrations as well as the higher impairment of proliferative and morphogenetic development was found in the case of irradiation at the end of the G_2-phase of 2-cell embryos. Those embryonic cells which had the longer interval between irradiation in G_2-phase and the 1^{st} mitosis post radiation had a better chance to repair radiation damage than cells with the shorter interval. Damage which remained unrepaired until the 1^{st} mitosis became manifest. The frequency of acentric fragments was less in the daughters of those cells which were irradiated at the middle of the G_2-phase and had divided comparatively late after irradiation. Apparently an appreciable repair of chromosomal damage had taken place before cell division (8, 9).

These studies demonstrate the development of radiation damage after exposure to X-rays and fast neutrons in preimplantation mouse embryos on the cellular and embryonic basis. The transplantation experiments show that even after high radiation doses the prenatal development is either inhibited completely or occurs normally. The recovery processes can be very extensive. The conditions of such mechanisms have been studied.

References

1. UNSCEAR, 1977 Sources and Effects of Ionizing Radiation, United Nations Publication Sales No. E.77.IX.1

2. C. Streffer, Prenatal irradiation and late effects: A review of data and open questions. Proceedings of the 7th international congress of radiation research, Amsterdam. Somatic and genetic effects, Martinus Nijhoff Publishers (1983).

3. M. Molls, U. Weißenborn, C. Streffer, Irradiation of mouse embryos in pronucleus and 2-cell stages: dependence of micronucleus formation and cell proliferation upon DNA amount and cell cycle phase. Strahlentherapie 158, 504-512 (1982).

4. C. Streffer, D. van Beuningen, M. Molls, N. Zamboglou, S. Schulz, Kinetics of cell proliferation in the pre-implanted mouse embryo in vivo and in vitro. Cell Tissue Kinet. 13, 135-143 (1980).

5. M. Molls, C. Streffer, D. van Beuningen, N. Zamboglou, X-irradiation in G-2 phase of two-cell mouse embryos in vitro: cleavage, blastulation, cell kinetics, and fetal development. Radiat. Res. 91, 219-234 (1982).

6. M. Molls, C. Streffer, N. Zamboglou, Micronucleus formation in pre-implanted mouse embryos cultured in vitro after irradiation with X-rays and neutrons. Int. J. Radiat. Biol. 39, 307-314 (1981).

7. M. Molls, C. Streffer, Irradiation of 2-cell mouse embryos in G-2-phase: Kinetics of chromosomal aberrations, cell proliferation, development and recovery. Proceedings of the 7th international congress of radiation research, Amsterdam. Somatic and genetic effects, Martinus Nijhoff Publishers (1983).

8. M. Molls, C. Streffer, B. Fellner, U. Weißenborn, M.-L. Steimann, W. Breipohl, Development of cytogenetic effects and recovery after irradiation of preimplantation mouse embryos. Proceedings EULEP Symposium, 17th Annual meeting of the european societv for radiation biology, Bordeaux (1982, in press).

9. M. Molls, C. Streffer, N. Zamboglou, Development of X-irradiated 2-cell mouse embryos in relation to micronucleus formation and G-2-delay. In: Developmental effects of prenatal irradiation, Gustav Fischer Verlag, Stuttgart-New York (1982).

LONGTERM ANIMAL STUDIES ON THE EFFECT OF INCORPORATED RADIOACTIVE AND NONRADIOACTIVE PARTICLES AND THEIR SYNERGISM - RADIOCHEMICAL PREPARATION OF TISSUE SAMPLES AND ALPHA-SPECTROSCOPY

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Introduction

When evaluating longterm effects of incorporated collodial ThO_2 (Thorotrast) from a radiobiophysical view of quantifying the radiation risk, the nonradiation effect of the foreign body must be considered. There are well-founded indications that in the induction of tumors the late detrimental effects observed originate not only from radiation but apparently also from physico-chemical stimulus of a foreign body in the sense of a synergism. In search of clarifying the participation of the nonradiation effect in the genesis of tumors after the incorporation of Thorotrast, a comparison is made on hand of animal experiments - within the scope of a combined research project - between the longterm effects of Thorotrast, Th-230 enriched colloidal ThO_2 and radioactive as well as nonradioactive colloids of identical physicochemical characteristics (We 83, Ri 83). The latter part of the project deals with colloidal zirconium dioxide injected into rats as nonradioactive zirconotrast or as a radiocolloid (radiozirconotrast) labelled with Th-230 and Th-228.

It was the objective of this study to develop methods for the radiochemical preparation of samples along with physical measuring and evaluating techniques for calculating tissue doses on the basis of results from the thorium isotopes Th-230 and Th-232 (with daughter products) in tissue samples of animals after administration of normal and enriched Thorotrast.

1. Radiochemical preparation of samples

A method was developed that allows a complete radiochemical work-up of tissue samples containing Thorotrast. For the subsequent electrodeposition, the method described by Schieferdecker, inter alia, (Sch 68, HA 77) was used as a starting point. A procedure was developed that resulted in a nearly mass-free (self-absorption, energy resolution) preparation of samples for alpha-spectroscopy.
The radiochemical sample preparation consists of five steps:

1. Th-229 added as internal standard
2. Wet ashing of Thorotrast tissue with concentrated nitric acid using a reflux condenser

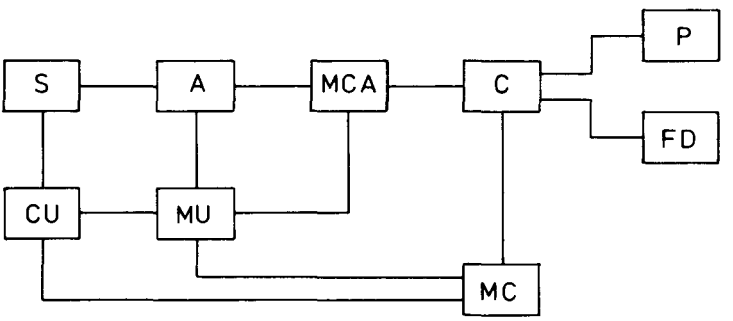
3. Separation of thorium from other elements by anion-exchange

4. Removal of an aliquot for the radiochemical work-up by electrodeposition

5. Electrodeposition of thorium from solution in sulfuric acid.

2. Design and function of the alpha-spectroscopy measurement system

For assessing thorium activity from different tissue samples, an alpha-spectroscopy measurement system was designed with a 500 mm^2 Si-surface barrier detector. Since the counting times are longer than 1000 min and a large number of samples from long-term tests had to be measured and evaluated, it was necessary to automate the system for routine measurements (Figure 1).



S	Sample changer		
A	Amplifier	CU	Control unit
MCA	Multichannelanalyzer	MU	Monitoring unit
C	Main computer	MC	Monitoring computer
P	Printer		
FD	Floppy disk		

Fig. 1: Block diagram of automated alpha-spectroscopy measurement system

System guidance and control is performed by the main computer C (Apple II Europlus). In order to use this computer also for other functions, e.g. spectrum analysis, a monitoring computer was installed for continuous system guidance and control (sample changer S, amplifier A, multichannel analyzer MCA and control unit CU). The spectrum fed into the multichannel analyzer automatically goes into the main computer and, with appropriate sample classification, is put out by printer P as hard-copy and/or stored on a floppy disk FD, as needed. The control unit CU continuously controls the functional parameters of the

highly sensitive semiconductor, e.g. vacuum and detector bias. In case of malfunction or error, indicated by the control unit, the monitoring computer gives a highest priority interrupt signal to the main computer. The main computer interrupts its program and registers the error. Depending on a pre-programed command, the main computer will again start the measurement program or signal its discontinuance.

Computer assistance is provided for evaluating the alpha-radiation spectra.

3. Results

3.1 Radiochemical sample preparation

For determining the activity of the thorium isotopes Th-232 Th-230 and Th-228, Th-229 is added to the tissue samples as a tracer prior to work-up. This procedure permits an individual determination of recovery for each single tissue sample. With good reproducibility the yield from organ samples containing Thorotrast is about 80 %.

Due to the extremely inhomogenous activity distribution in organs, one each complete organ sample is worked up and the mean activity derived.

3.2 Alpha-spectroscopy

The efficiency yielded via an Am-241 reference source is about 25 %. The energy resolution of the semiconductor is reported to be 25 keV FWHM; in practice, however, the energy resolution of single peaks for electrodeposition from physically pure standard solutions of thorium ranges from 50 to 60 keV FWHM, and from 60 to 80 keV FWHM for the electrodeposition of worked up Thorotrast tissue samples (Figure 2).

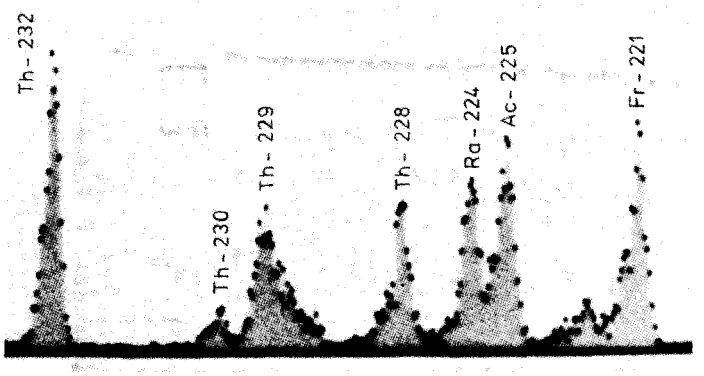


Fig. 2: Typical thorium spectrum following electrodeposition of a Thorotrast sample

Due to the low activity in tissue samples (approx. 1 pCi), counting times range from 1000 to 1700 min, depending on the sample, whereby the mean background rates are as follows:

Th-232	0.028 cpm
Th-230	0.025 cpm
Th-228	0.138 cpm.

The higher background rate in the Th-228 range can be attributed to detector contamination by recoil atoms.

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References:

We 83 Wegener K., Hasenöhrl K. and Wesch H., Health Physics 44, Supplement No. 1, 1983, pp 307-316

Ri 83 Riedel W., Dalheimer A., Kaul A., Said M. and Walter U., Health Physics 44, Supplement No. 1, 1983, pp 239-298

Sch 83 Schieferdecker H., KFK 810, 1968, Kernforschungszentrum Karlsruhe

HA 77 HASL-Report, 1977, "Radiochemical Determination of Thorium in Urine", E-TH-03-01

DEUTSCHE THOROTRASTSTUDIE
STRAHLENWIRKUNG UND FREMDKÖRPERWIRKUNG IM TIEREXPERIMENT IM VERGLEICH ZUR MENSCHLICHEN THOROTRASTOSE - PATHOLOGISCHE ANATOMIE

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Die kanzerogene Wirkung des Kontrastmittels Thorotrast wurde seit 1930 in zahlreichen Tierexperimenten nachgewiesen. Jedoch erbrachten nur wenige dieser Versuche Ergebnisse, die denen bei Thorotrastpatienten vergleichbar waren. Dies hatte folgende Gründe:

1. In vielen Tierexperimenten wurde Thorotrast nicht i.v. injiziert.
2. Die Tierzahl war in fast allen Experimenten für statistische Aussagen zu gering.
3. Die Tierversuche wurden zum größten Teil nicht als Überlebensversuche geplant und ausgeführt.

In einer Überlebensstudie injizierten wir 1920 weiblichen Wistarratten i.v. unterschiedliche Mengen Thorotrasts variabler Aktivität. Wir versuchten folgende Fragen zu klären:

1. Entwickelt die Ratte in den bei der menschlichen Thorotrastose kritischen Organen Leber und Milz die gleichen Tumoren wie der Mensch ?
2. Stimmt die Morphologie dieser Tumoren bei Mensch und Ratte überein ?
3. Ist das Verteilungsmuster des Thorotrasts beim Menschen und bei der Ratte in den Organen des RHS identisch?
4. Lassen sich die nichttumorösen Veränderungen in Organen des RHS bei Mensch und Ratte vergleichen ?
5. Läßt sich der Anteil der Strahlenwirkung und der Fremdkörperwirkung an der Tumorgenese bestimmen (vgl.Poster: H. Wesch et al.) ?

Das i.v. verabreichte Thorotast wird wie beim Menschen in allen Organen und Geweben der Ratte, insbesondere aber im RHS abgelagert. Der größte Teil wird in der Leber, extrazellulär und intrazellulär in Hepatozyten und von Kupffer' schen Sternzellen abgelagert. Weitere wichtige Speicherorgane sind Milz, hiläre und parapankreatische Lymphknoten und Knochenmark.

Bei den 1920 Versuchstieren entwickelten sich 239 Tumoren der beim Menschen kritischen Organe Leber und Milz:

69 Gallengangsadenome,
13 Leberzelladenome,
18 Haemangiome der Leber,
20 Haemangiome der Milz,

8 Gallengangscarcinome,
25 Leberzellcarcinome,
47 Haemangiosarkome der Leber,
5 Haemangiosarkome der Milz,
35 sogenannte "Kupfferzellsarkome" der Leber.

Diese Tumoren traten statistisch signifikant häufiger in den Versuchsgruppen als in den Kontrollgruppen auf.

Die Tiere der Versuchsgruppen und Kontrollgruppen entwickelten darüber hinaus zahlreiche andere Tumoren wie z.B. Adenocarcinome des Uterus, Fibroadenome und Adenocarcinome der Mamma, Plattenepithelcarcinome der Haut und Stromasarkome des Uterus. Bei diesen Tumoren fanden sich keine statistisch signifikanten Häufungen.

Die Verteilung und das Muster der Ablagerung von Thorotrast sind bei Mensch und Ratte identisch. Als späte Reaktion auf die Thorotrastablagerung in der Leber werden beim Menschen häufig Leberfibrosen und Lebercirrhosen beobachtet. Unsere Versuchstiere dagegen haben nur selten Leberfibrosen und nur in zwei Fällen postnekrotische Lebercirrhosen entwickelt. Diese unterschiedliche Reaktionsweise der Leber von Mensch und Ratte geht möglicherweise auf die sehr viel bessere Regenerationsfähigkeit der Leber beim Versuchstier zurück.

Das Auftreten von Gallengangscarcinomen, Leberzellcarcinomen und Haemangiosarkomen der Leber bei unseren Versuchstieren bestätigt ältere Untersuchungen an der Ratte und an anderen Versuchstieren (Übersicht bei Wegener, 1979). Thorotrastpatienten entwickeln in vielen Fällen nach i.v. Injektion des Kontrastmittels die gleichen oben genannten malignen Lebertumoren. Die Tumoren bei Mensch und Ratte sind morphologisch völlig identisch und zeigen ein sehr ähnliches biologisches Verhalten z.B. in der Metastasierung, der Infiltration der Leber und dem Übergreifen auf benachbarte Organe und Gewebe.

In unserem Tierexperiment haben sich im Verhältnis zur Zahl der Lebertumoren nur wenige Milztumoren entwickelt. Dies entspricht den Beobachtungen anderer Untersucher und den Beobachtungen beim Menschen. Thorotrastpatienten zeigen nur sehr selten primäre Milztumoren ausschließlich in Form benigner Haemangiome.

Leberzelladenome und Gallengangsadenome sind in der Literatur bei Thorotrastpatienten nicht beschrieben. Bei unseren Versuchstieren sind diese Tumoren aufgetreten. Das Fehlen dieser Tumoren beim Menschen könnte darauf zurückgeführt werden, daß sich aus ihnen zum Zeitpunkt der Untersuchung der Patienten Jahrzehnte nach der Thorotrastapplikation bereits Carcinome entwickelt hatten.

Den interessantesten Tumor bei Thorotrastpatienten wie bei unseren Versuchstieren stellt zweifellos das Haemangiosarkom dar. Dieser Tumor tritt beim Menschen spontan ausgesprochen selten auf. Edmondson berichtet 1958, daß unter 52000 Autopsien im Los Angeles County Hospital nur ein einziges spontanes Haemangiosarkom der Leber diagnostiziert worden sei. Ein gehäuftes Auftreten dieses Tumors beim Menschen wird außer nach Gabe von Thorotrast auch nach Arsen- und Vinylchloridexposition gesehen. Die beim Menschen und unseren Versuchstieren aufgetretenen Haemangiosarkome sind morphologisch vergleichbar. Die Tumorzellen des Haemangiosarkoms wachsen entlang der Oberfläche der Leberzelltrabekel. Sie kleiden unterschiedlich weitgestellte Bluträume aus oder bilden schwammartige Formationen. Eine Speicherung von Thorotrast in Haemangiosarkomzellen haben wir lichtmikroskopisch weder bei Thorotrastpatienten noch beim Versuchstier beobachten können. Bis heute ist nicht eindeutig bestimmt, welche von den bekannten Arten von Sinuswandzellen der menschlichen Leber die Mutterzelle des Haemangiosarkoms darstellt. Wir haben bei unseren Versuchstieren in der Leber zwei Sarkome von Sinuswand-

zellen unterschiedlicher Struktur gefunden. Das Haemangiosarkom und ein sog. "Kupfferzellsarkom". Die Tumorzellen des sog. Kupfferzellsarkoms sind größer und zytoplasmareicher als die des Haemangiosarkoms. Sie kleiden die Sinusoide nicht tapetenartig aus, sondern füllen die Sinusoide mehr oder weniger vollständig aus. Die Zellen dieses Sarkoms speichern Thorotrast, auch wird das Kontrastmittel bei der Metastasierung mit den Tumorzellen verschleppt. Elektronenmikroskopische Untersuchungen der beiden verschiedenen Tumoren haben gezeigt, daß es sich bei den Haemangiosarkomen um Sarkome von Sinusendothelien, bei den sog. Kupfferzellsarkomen tatsächlich um Sarkome der von Kupffer' schen Sternzellen handelt (persönliche Mitteilung Professor Hüber/München).

Die Ablagerung von Thorotrast in der Lunge bei Thorotrastpatienten und Versuchstieren sowie die dauernde Abstrahlung des beim radioaktiven Zerfall entstehenden Edelgases Thoron haben sowohl beim Menschen wie bei Versuchstieren gehäuft Lungentumoren erwarten lassen. Entgegen dieser Erwartungen sind weder bei den Thorotrastpatienten noch bei unseren Versuchstieren im Vergleich zu Kontrollgruppen gehäuft Lungentumoren aufgetreten. Bisher ist nicht geklärt, worauf dieses "negative Phaenomen" zurückzuführen ist.

In der Literatur wird immer wieder über einzelne Fälle von Knochensarkomen nach Thorotrastapplikation beim Menschen berichtet. Eine statistisch signifikante Häufung von Knochentumoren hat sich bisher bei Thorotrastpatienten auch nach einer Latenzzeit von über 30 Jahren nicht nachweisen lassen. Wir können diese Befunde durch unsere Tierversuche bestätigen: Bei unseren Versuchstieren sind einige ganz wenige Fälle von Knochentumoren aufgetreten (Osteosarkom, Osteoklastom). Doch sind auch diese Zahlen gegenüber denen der Kontrollgruppe statistisch nicht signifikant.

Unsere Versuche haben gezeigt, daß es möglich ist, durch Gabe von Thorotrast bei Ratten als Versuchstieren die gleichen Tumoren zu erzeugen, wie sie gehäuft und statistisch signifikant vermehrt bei Thorotrastpatienten auftreten. Auch das biologische Verhalten dieser Tumoren entspricht dem beim Menschen. Zum dritten hat sich nachweisen lassen (vgl.Poster Wesch), daß eine lineare Dosis-Wirkungsbeziehung bei der Erzeugung dieser "Thorotrasttumoren" beim Versuchstier besteht. Nicht eindeutig haben wir mit diesem Versuch klären können, welchen genauen Anteil bei der Entstehung der malignen und benignen Tumoren der Strahlenwirkung und welcher der Fremdkörperwirkung bzw. welcher der Wirkung beider Faktoren zukommt. Lediglich das Ergebnis, daß durch den Fremdkörper die Latenzzeit für die Entstehung der Tumoren verkürzt wird, läßt sich aus unseren Untersuchungen eindeutig ablesen.

Aus diesen Gründen haben wir in einer zweiten Versuchsreihe, wiederum mit weiblichen Wistarratten (1260 Versuchstiere) die Frage der Fremdkörperwirkung und Strahlenwirkung in einer anderen Anordnung zu klären versucht. Einem Teil der Versuchstiere haben wir wiederum Thorotrast unterschiedlicher Aktivität injiziert (siehe Versuch oben), einem anderen Teil haben wir nichtradioaktive Kolloide identischer physikalisch-chemischer Zusammensetzung mit und ohne Anreicherung durch Th-228 und Th-230 ("Zirkonotrast") injiziert.

Dieses Experiment ist noch nicht abgeschlossen. Es lassen sich jedoch aufgrund der Untersuchung an inzwischen 1000 Tieren folgende

Aussagen machen:

1. Auch in diesem Experiment sind die gleichen Tumoren aufgetreten, wie sie oben für das Thorotrastexperiment beschrieben worden sind.
2. Diese Tumoren haben ein gleiches biologisches Verhalten wie die Tumoren im Thorotrastexperiment und die Tumoren bei Thorotrastpatienten gezeigt.
3. Es scheinen vermehrt Haemangiosarkome der Milz entstanden zu sein.
4. Die Verteilung des Thorotrasts wie des Zirkonotrasts ist in den Organen des RHS identisch.

Edmondson, H.A. Tumors of the Liver and Intrahepatic Bile Ducts; Armed Forces Institute of Pathology, Section VII-Fascile 25, 1958.

Wegener, K. Systematic Revies of thorotrast data and facts: Animal experiments; Virch. Arch. A. Path. Anat. Histol.; 381, 245, 1979.

EVALUATION OF THE RADIATION AND NON-RADIATION EFFECTS AFTER INJECTION OF RADIOACTIVE THOROTRAST OR ZIRCONOTRAST*

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INTRODUCTION

Since the introduction of Thorotrast as a radiographic contrast medium, its carcinogenic effect has been demonstrated in many experiments (We79). None of these studies defined the relationship of dose-rate and tumor incidence, or answered the question of possible foreign body involvment ("non-radiation" effect).

In our first long-term study a Th-230-incorporated thorium dioxide colloid (enriched Thorotrast) was applied. This experiment demonstrated that the frequency of liver and spleen tumor occurence is linearly dependent on the dose-rate, but was not correlated with the number of particles of injected Thorotrast (Wes83). However, these investigations provided only limited information on the pure contribution of the "non-radiation" effect, since Thorotrast itself is radioactive. The goal of our second long-term study was to determine, by means of suitable non-radioactive and radioactive colloids of identical physical and physicochemical properties, the foreign body effect.

This paper presents preliminary results on the incidence of primary hepatic and splenic tumors after a single injection of Th-228/Th-230 enriched Zirconotrast as a function of dose-rate and the applied amount of the colloid.

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MATERIAL AND METHODS

The radioactive Zirconotrast was produced by adding different amounts of Th-228/Th-230 during the preparation of Zirconotrast. Due to the different rates of decay of Th-232 and the Th-228/Th-230 mixture it was necessary to define the point in time when the accumulated dose would be the same, for calculating the initial dose-rate. An acceptable compromise seems to be 1.5 years, considering the assumed latency period of the liver tumor in the rat (Ri83).

Six stock solutions of Zirconotrast were prepared. Compared with Thorotrast the following α-energy emission rates were sought: 0 (inactive Zirconotrast), 1 (the α-dose is equal to the α-dose of Thorotrast 1.5 years after injection), 2.5, 5, 10 and 25. Each radioactive stock solution was divided into three subgroups. Appropriate amounts of inactive Zirconotrast were added to enhance the foreign body effect by a factor of 2.5 and 5. In addition solutions of Thorotrast and physiological saline were injected. All animals were injected with the maximum volume of 600 µl via the tail vein. The animals treated with 120 or 300 µl of the colloid received this amount adequately diluted with physiological saline solution.

Table 1 gives the experimental design. For better comparison with the first long-term study (Wes83) the relative dose-rates in the different groups were standardized to 60 µl of normal Thorotrast. Each experimental group included 60 animals.

Table 1. Relative dose-rates in different experimental groups, standardized to 60 µl of normal Thorotrast.

		Injected volume (µl)				
		Zirconotrast			Thorotrast	
		120	300	600	120	600
Enrichment factor	0	0	0	0	–	–
	1	2	2	2	2	10
	2.5	5	5	5	–	–
	5	10	10	10	–	–
	10	20	20	20	–	–
	25	50	50	–	–	–
	NaCl	–	–	0	–	–

RESULTS AND DISCUSSION

In Table 2, the mean survival time (median) of the animals is recorded. Compared with the control animals the survival time was significantly reduced in the enrichment groups 10 and 25. At constant dose-rates no significant decrease of survival time was observed as a function of the injected volume.

Table 2. Mean survival time (median) in the different experimental groups.

Enrichment factor	Injected volume (µl)				
	Zirconotrast			Thorotrast	
	120	300	600	120	600
0	808	793	784	–	–
1	790	767	773	772	769
2.5	781	771	823	–	–
5	794	710	814	–	–
10	716	732	679	–	–
25	574	523	–	–	–
NaCl	–	–	824	–	–

Table 3. Percentage of rats with primary hepatic or splenic tumors. Numbers not underlined may change slightly since some animals in these groups are still alive.

Enrichment factor	Injected volume (µl)				
	Zirconotrast			Thorotrast	
	120	300	600	120	600
0	0	0	2	–	–
1	0	2	3	2	2
2.5	3	0	2	–	–
5	8	8	10	–	–
10	28	22	18	–	–
25	57	46	–	–	–
NaCl	–	–	2	–	–

The percentage of rats with primary hepatic or splenic tumors, in the different groups, is shown (Table 3). As in the first study (Wes83) the following histological types were grouped together: liver cell adenoma, bile duct adenoma, hemangioma, liver cell carcinoma, bile duct carcinoma, hemangiosarcoma and so-called Kupffer cell sarcoma.

Although not all animals of the lower dose-rate groups have died, a significant increase in hepatic or splenic tumor incidence was observed when the dose-rate was raised from zero to 20 or zero to 50. At a constant dose-rate (20 and 50) a decrease in the number of tumors was observed when the injected volume (number of particles) was raised by a factor of 5. This decrease may be due to the lower retention of the applied colloid, or due to the conglomerate size in the high volume groups (Ri83).

In Figure 1 we compared the percentage of hepatic and splenic tumor bearing animals from the first animal study with the dose-rate groups 20 and 50 from the second study, as a function of the relative dose-rate. The close agreement between the two studies can be

seen even if final results of the lower dose-rate groups are not yet available. The slightly higher tumor incidence in the second study may be explained with the following necessary assumptions used for calculating the initial activity concentration of Th-228/Th-230-Zirconotrast. The average retention of the colloid, the diameter of the conglomerates, the self-absorption coefficient and the point in time (1.5 years) were considered. The influence which each individual parameter has, still requires clarification, and will demand a final calculation.

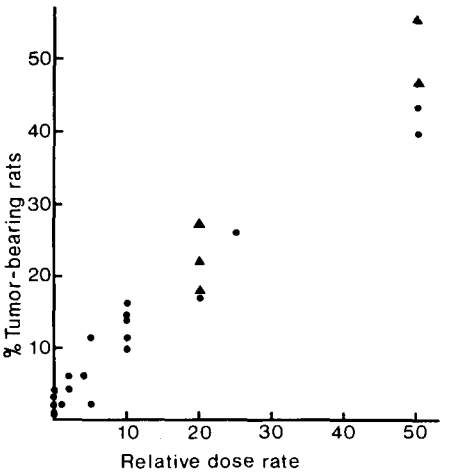

Figure 1. Percentage of hepatic and splenic tumor bearing animals in relation to the relative dose-rate. (●) First animal experiment with Th-230-enriched Thorotrast; (▲) this study.

The results of the two studies demonstrated that the frequency of liver and spleen tumor occurence following a single injection of Th-230-enriched Thorotrast or Th-228/Th-230-Zirconotrast, is approximately linear dependent on the dose-rate. The inactive colloid (Zirconotrast) failed to enhance the tumor incidence nor did it increase the spontaneous tumor incidence at constant dose-rates.

REFERENCES

Ri83 Riedel W., Dahlheimer M., Walter U. and Kaul A., 1983, "Recent Results of the German Thorotrast Study:Dose Relevant Physical and Biological Properties of Thorotrast Equivalent Colloids", Health Phys., **44**, Supplement No. 1, 293-298.

We79 Wegener K., 1979, "Systematic Review of Thorotrast Data and Facts: Animal Experiments", Virchows Arch. A. Path. Anat. Histol. **381**, 245.

Wes83 Wesch H., van Kaick G., Riedel W., Kaul A., Wegener K., Hasenöhrl K., Immich H. and Muth H., 1983, "Recent Results of the German Thorotrast Study:Statistical Evaluation of Animal Experiments with Regard to the Non-radiation Effects in Human Thorotrastosis", Health Phys., **44**, Supplement No. 1, 317-321.

RISK ANALYSIS OF COMBINED EFFECT ON MICE EXPOSED TO MULTIPLE ENVIRONMENTAL FACTORS

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The simultaneous effect of multiple factors on organisms is not simple. The scientists have usually analysed the effect of each factor alone, however, in reality organisms live not under single stimulant but under multiple ones and they respond to these all together.

The aim of the present study is to evaluate the response of the mouse exposed to multiple risk factors including sublethal doses of radiation and metals.

Seven hundred and twenty male mice of six week old were utilised. They were divided into 36 groups of equal size, and the twenty mice in each group were exposed to varied risk sources under the following four conditions : (1) exposure to gamma radiation giving intestinal dose of 524, 571 or 617 rad, (2) with or without constant administration of 1000 ppm zinc in drinking water, (3) with or without a spot injection of 1.2 or 4.7 mg Cd/kg weight and (4) constant intake of either calcium deprived diet or normal mouse feed. The mortality of the mice within 30 days after irradiation was recorded.

We analised the data by assuming following mathematical models : (1) log-linear model, where the logarithm of the mortality of a mouse is expressed by a linear function of risk factors. The optimal model was chosen from various unsaturated models by comparing AIC statistic and (2) multiple-logistic model which expresses the mortality of a mouse as $p = [1 + \exp(-\beta_0 - \beta_1 x_1 - \beta_2 x_2 - \beta_3 x_3 - \beta_4 x_4)]^{-1}$ where x_L 's denote the risk factors and β_i 's are unknown parameters. The maximum likelihood estimates of parameters were obtained.

The results obtained can lead to the following : (1) the log-linear model is more effective than the logistic model for detecting the association among factors, while the latter is better than the former for predicting the effect of risks.

BIOLOGICAL EFFECTS BY ANTENATAL IRRADIATION OF MICE WITH LOW LEVELS OF REACTOR RADIATION

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Considering the biological effects of low levels of ionizing radiation, developing mammals are particularly sensitive to radiation during their intrauterine and early postnatal stages. Recent information indicates that measurable damage can be produced by doses 10 - 19 cGy(kerma), and also the effects of radiation are related to the developmental stage at which exposure occurs, and correspondence has been demonstrated in this respect between man and other mammals [1]. However, many experimental studies on the effects of ionizing radiation on the mammalian embryo and fetus have been conducted using mainly X-rays or gamma-rays irradiation[2,3]. As an approach to the studies on the biological effects of low dose rates and low doses of reactor radiation, the experiments on antenatal irradiation of mice using UTR-KINKI, a low power research reactor operating at 1 Watt have been conducted. The experiments are in progress, and this paper deals with the death rate, growth retardation, histopathological changes in lymph nodes and tumorigenesis of mice irradiated at fetal age of 14 days after conception, from birth to age of 600 days.

MATERIALS AND METHODS

The pregnant female ICR/JCL mice, 10 to 15 weeks old were used. Nine pregnant females, as one group, were confined in a circular plastic cage, and irradiated at fetal age of 8 or 14 days after conception at the central top of graphite reflector of the reactor operating at 1 Watt for 6 hrs(21.4 cGy), 12 hrs(43.1 cGy) and 24 hrs (86.2 cGy), respectively. During irradiation, mice were allowed to drink water *ad libitum*. The experimental mice including unirradiated control mice are allowed to complete their life spans under conventional rearing conditions with high sanitary care at the room temperature of 23 - 25°C.

The irradiation doses were measured using tissue equivalent TLD (BeO) for gamma-rays, gold foil activation method for thermal neutrons and silicon diode dosimeter for fast neutrons, respectively. The dose rates and their fractions under experimental conditions were obtained to be 2.56 cGy/hr and 0.73 for gamma-rays, 0.56 cGy/hr and 0.16 for thermal neutrons using a flux to dose conversion factor[4], and also 0.40 cGy/hr and 0.11 for fast neutrons, respectively. The cadmium ratio of 8.1 and gamma-ray energy index of 0.83 based on the ratio of TL response of BeO to that of $CaSO_4$ were also obtained.

RESULTS AND DISCUSSION

Early Death and Growth Retardation: sex-averaged observed death rates during growth of mice irradiated at fetal age of 14 days after conception are shown in Table 1. No obvious dose dependency in the death rates was observed between the mice exposed to 21.4 cGy and

43.1 cGy, however, the effectiveness of the given doses was noticeable in comparison with the unirradiated control group, although their life spans have not been completed yet.

Table 1. Sex-Averaged Observed Death Rates during Growth of Mice Irradiated at Fetal Age of 14 Days After Conception. (Up to Age of 600 Days)

	21.4 cGy	43.1 cGy	86.2 cGy	Control
Number of mice	60	60	60	60
Stillbirth	0 %	0 %	15.0 %	0 %
Infant (1 - 20 days)	0	0	0	0
Juvenile (21 - 100 days)	0	0	6.7	0
Adolescence (101 - 200 days)	11.7	6.7	8.3	3.3
Adult (201 - 300 days)	10.0	6.7	1.7	0
(301 - 400 days)	6.7	15.7	8.3	3.3
(401 - 500 days)	15.0	11.7	15.0	3.3
(501 - 600 days)	18.3	11.7	18.3	18.3
Total	61.7 %	52.5 %	73.3 %	28.2 %

Thomson, et al.,[5, 6] reported that the radiation-specific excess mortality rate(EMR) of mice irradiated at the age of about 110 days can be fitted to a power function,

$$EMR = aD^b,$$

where D is the radiation dose in cGy, both a and b are the parameters under the study. In their studies, the doses of fission neutrons were varied from 20 to 240 cGy, and also from doses below 20 cGy down to doses of 5 and 10 cGy in a single exposure. In the above expression, they stated that approximation of the radiation-specific excess mortality rate will be valid only if the distribution of death times in the control and irradiated populations show the same symmetry. In the present study, however, somewhat accelerated occurrence of early deaths were observed at the highest dose of 86.2 cGy, so further studies are needed for estimation of the radiation-specific excess mortality rate under the experimental conditions.

To check the effect of antenatal irradiation on growth of mice after birth, body weight changes from birth to age of about 300 days were followed, however, tumors were found hereafter in the irradiated groups. Nearly linear reduction of the relative body weight normalized to the control was observed in both males and females, however, more obvious reduction was found in males than in females with dose as shown in Fig. 1. Sasaki, et al.,[2] pointed that the long-term effects of X-irradiation at the middle intrauterine stage(12 days post coitum) were persistent growth retardation and degenerative diseases such as amyloidosis, more than tumorigenesis. In our experiments, growth retardation seemed to be more obvious in males than in females.

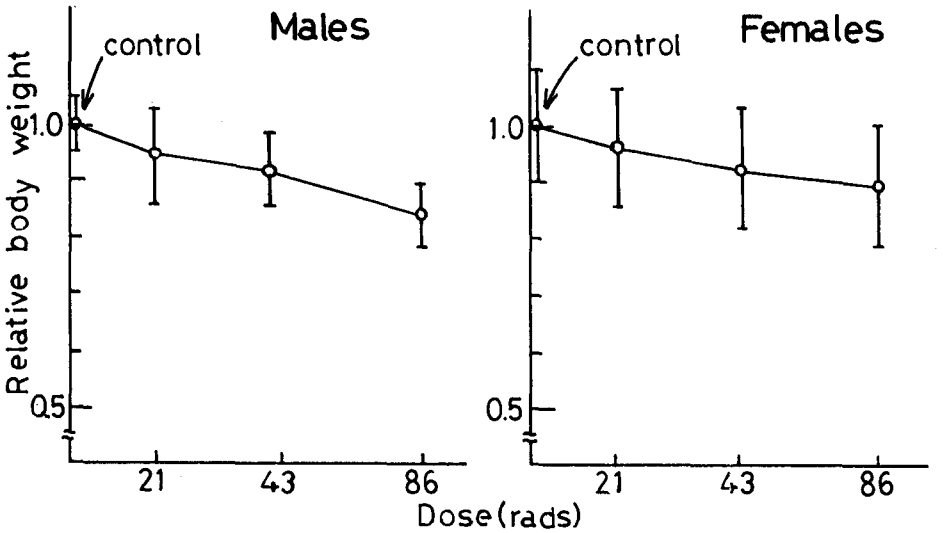

Fig. 1. Relative Body Weight of Mice Irradiated at Fetal Age of 14 Days After Conception (At Age of 316 Days)

Histopathological Changes in Lymph Nodes: in the life of the irradiated mice up to age of about 300 days, somewhat systemic changes of lymph nodes were found in non-tumor bearing mice. The reactive and degenerative changes such as hyperplasia, infiltration of round cells and plasma cells, vacuole formation, disappearance of follicles and destruction of reticulum were observed with dose and time elapsed.

Tumorigenesis: ICR/JCL mice have been developed as mice which have less susceptibility to spontaneous breast tumor. The observed incidences of overall tumors up to age of 600 days are shown in Table 2. It indicated that the incidences of overall tumors increased with dose, and also the occurrence of tumors was found in earlier time with dose, as pointed early by Upton[7], although the incidence of tumors did not show so high values(less than 21.1 %). Furthermore, it seemed to be noticeable that the occurrences of tumors were observed in females rather than in males. However, it is not excluded that some bias might occur due to the difference in the effective numbers of females and males, and also the effective number of mice in each group was not provided enough for good statistics. Thomson, et al.,[5] pointed out that females are appreciably more sensitive than males to neutron but to gamma radiation from their survival data on mice exposed at age of about 110 days to single doses of fission neutron and gamma radiation. In the present study, it seemed that females were more susceptible to reactor radiation induced tumors than males, although it is not conclusive. On the other hand, histopathological examination on the tumors showed a variety of tumors

such as malignant lymphoma, malignant mesothelioma, reticulosarcoma, immunoblastosarcoma, erythroleukemia, plasmacytoma, and so on.

Table 2. Observed Incidence of Malignant Tumors in Mice Irradiated at Fetal Age of 14 Days After Conception. (Up to Age of 600 Days)

	21.4 cGy		43.1 cGy		86.2 cGy		Control	
	Female	Male	Female	Male	Female	Male	Female	Male
Effective No. of mice	19	14	19	16	19	16	18	13
Adolescence (101 - 200 days)	0	0	0	0	0	0	0	0
Adult (201 - 300 days)	0	0	0	0	0	0	0	0
(301 - 400 days)	0	0	2	0	2	0	0	0
(401 - 500 days)	1	0	1	0	1	0	0	0
(501 - 600 days)	1	0	0	0	1	0	1	0
Total	2	0	3	0	4	0	1	0
	(10.6%)	(0%)	(15.8%)	(0%)	(21.1%)	(0%)	(5.6%)	(0%)

SUMMARY

The results obtained are summarized as follows:

1. More obvious growth retardation was observed with dose in males than in females.
2. The death rate during growth was accelerated by antenatal irradiation of mice.
3. A variety of malignant tumors was observed in the irradiated groups with dose, and the occurrences of tumors seemed to be prominent in females rather than in males.

REFERENCES

[1] BEIR Committee (1980): The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980, National Academy Press, Washington, D.C., p. 447.

[2] Sasaki, S., Kasuga, T., Sato, F., and Kawashima, N. (1978): Late Effects of Fetal Mice X-Irradiated at Middle or Late Intrauterine Stage, Gann, 69, 167.

[3] Kusama, T., and Yoshizawa, Y. (1982): The Carcinogenic Effects of Fetal and Postnatal Radiation in Female Mice, J. Radiat. Res., 23, 290.

[4] Rindi, A. (1977): An Analytical Expression for the Neutron Flux-to-Adsobed-Dose Conversion Factor, Health Physics, 23, 264.

[5] Thomson, J. F., Williamson, F. S., Grahn, D., and Ainsworth, E. J. (1981): Life Shortening in Mice Exposed to Fission Neutrons and γRays, I. Single and Short-Term Fractionated Exposure, Radiat. Res., 86, 559.

[6] Thomson, J. F., Williamson, F. S., and Grahn, D. (1983): Life Shortening in Mice Exposed to Fission Neutrons and γRays, III. Neutron Exposures of 5 and 10 Rad, Radiat. Res., 93, 205.

[7] Upton, A. C. (1961): The Dose-Response Relation in Radiation-induced Cancer, Cancer Res., 21, 717.

BIOLOGICAL RESPONSE OF BEAGLE DOGS TO INHALED MONODISPERSE AEROSOLS OF $^{239}PuO_2$

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Inhalation has proven to be a likely route for human exposure to ^{239}Pu aerosols. Such exposures have occurred in occupational accidents involving fugitive releases of aerosols, and to the general population as a result of inhaling fallout ^{239}Pu. The studies reported here, in which dogs received single brief inhalation exposures to monodisperse $^{239}PuO_2$ aerosols, are part of a group of studies in which the role of local alpha dose rate from individual particles, initial average dose rate to lung, total average dose, and fractional lung irradiation are being evaluated with respect to the production of biological effects, primarily in the lung. Results from parallel studies in which dogs that inhaled $^{239}PuO_2$ were serially sacrificed have shown that less than 0.1% of the alpha radiation dose was delivered to tissues other than the lung and thoracic lymph nodes by two years after exposure (1).

MATERIALS AND METHODS

A total of 216 Beagle dogs from the Institute's colony have inhaled aerosols of monodisperse particles of $^{239}PuO_2$. Forty-eight dogs were exposed in a nose-only exposure system to 0.75 μm AMAD particles, 96 to 1.5 μm AMAD particles and 72 to 3.0 μm particles. Thirty-six dogs were exposed to the aerosol vehicle, a dilute ammonium hydroxide solution, to serve as controls. Each study had four or more desired activity levels, expressed as initial pulmonary burdens: 0.37, 1.1, 2.6, 5.2 kBq/kg body weight, and with 12 dogs per activity level. Two additional levels were used in the study with 3.0 μm AMAD $^{239}PuO_2$, 10.4 and 20.7 kBq/kg, and four additional levels were used in the 1.5 μm AMAD $^{239}PuO_2$ study, 0.0092, 0.085, 10.4 and 20.7 kBq/kg. Some physical and dosimetric details for this experimental design are given in Table 1. Methods for the preparation of the aerosols and the exposure of the dogs have been previously described (2,3). In addition to pre-exposure clinical evaluation, each dog received daily observation, annual physical and radiographic examination, and semi-annual blood cell counts and serum chemistry tests. Sick dogs were examined and tested to establish a diagnosis. A few dogs died from their illness but most were euthanized when moribund. Necropsies were done on all dogs, and tissues were evaluated histologically. Both tissues and collected excreta samples were analyzed radiochemically for ^{239}Pu content by alpha liquid scintillation counting.

TABLE 1. Design for the Study of the Effects of Inhaled $^{239}PuO_2$ in Dogs

Parameter	0.75 μm (AMAD)	1.5 μm (AMAD)	3.0 μm (AMAD)
Physical size, μm	0.18	0.44	0.96
MBq per particle	0.048	0.74	7.4
Local dose rate, mGray/day	0.65	9.6	99
Number of particles, range	$6x10^7 - 2x10^9$	$6x10^5 - 6x10^8$	$4x10^5 - 1x10^8$
Fraction of lung irradiated	1.0	0.03 – 1.0	0.02 – 1.0
Initial pulmonary burden, kBq	3 – 74	0.3 – 330	2 – 630
Average initial dose rate to lung, Gray/day	0.002 – 0.07	0.003 – 0.3	0.002 – 0.5

Research conducted under U.S. Department of Energy Contract No. DE-AC04-76EV01013 and in facilities fully accredited by the American Association for the Accreditation of Laboratory Animal Care.

RESULTS

Results from the study in which dogs were serially sacrificed subsequent to inhaling monodisperse $^{239}PuO_2$ aerosols have shown that up to two years after exposure, 99% of the burden measured at sacrifice was in lung and thoracic lymph nodes. The largest fraction of the initial pulmonary burden (65-90% IPB) was being retained in the lung with half times of 700-1800 days. Greater retention times were noted with increasing particle size. Of the ^{239}Pu leaving the lung, ≈ 75% was cleared mechanically via the bronchi and trachea to be swallowed and excreted in the feces; the remaining 25% accumulated approximately linearly with time in the thoracic lymph nodes. As an example, the tissue retention curves for lung, thoracic lymph nodes, skeleton, and liver are shown for the 1.5 μm AMAD group (Figure 1) (1). These patterns were markedly different from those of $^{238}PuO_2$, where accelerated clearance from lung and increased solubilization *in vivo* was noted beginning about 100 days after exposure (3). The accumulation of radiation dose in the four organ systems of concern (Figure 2) indicate a range of 10^5 in absorbed dose between bone and thoracic lymph nodes. By 2 years, the lung has accumulated a dose of 0.1 Gray/kBq of initial pulmonary burden.

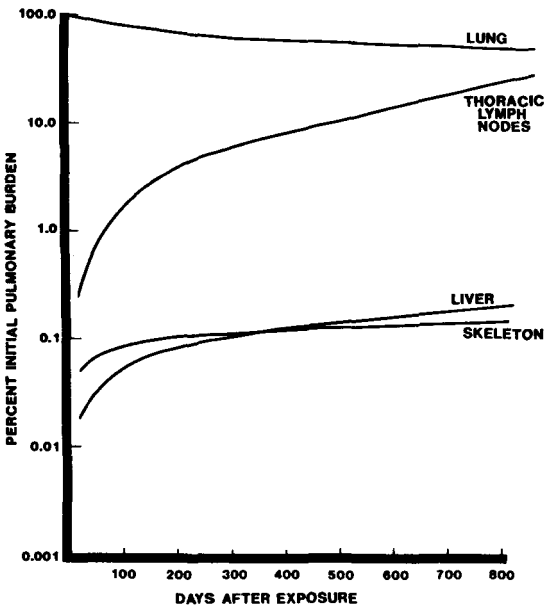

FIGURE 1. Retention of ^{239}Pu in tissues of Beagle dogs exposed to 1.5 μm AMAD monodisperse PuO_2 aerosols.

As of September 30, 1983, all surviving dogs had been on study at least 4.5 years. The biological effects to date have been radiation pneumonitis with or without pulmonary fibrosis, and pulmonary carcinoma (Table 2). The earliest death from radiation pneumonitis occurred 105 days after exposure; dogs are continuing to die at and beyond 4.5 years with pneumonitis and pulmonary fibrosis. Most of these dogs have accumulated alpha radiation doses to lung of 10-80 Gray. The earliest appearance of pulmonary carcinoma at death was 1108 days after exposure. However, the incidence of carcinoma did not increase appreciably until about 1500 days. Of the 18 dogs that have died since 1300 days after exposure, 16 had pulmonary carcinoma. The mean (± SD) dose to lung at death for those dogs dying with pulmonary tumors were 18 (± 7) Gray for the 0.75-μm group, 27 (± 10) Gray for the 1.5-μm group, and 49 (± 24) Gray for the 3.0-μm group.

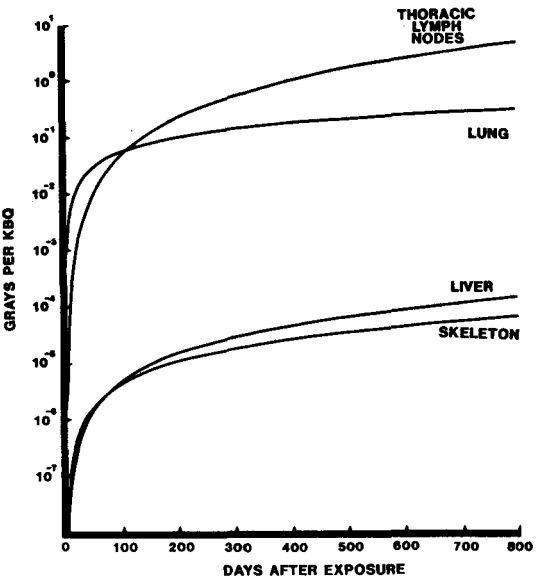

FIGURE 2. Cumulative radiation dose to the tissues of a 10 kg Beagle dog that inhaled 1.5 μm AMAD $^{239}PuO_2$ aerosol.

TABLE 2. Biological Effects from Inhaled $^{239}PuO_2$ as of September 30, 1983*

Study	Dogs Originally on Study	Radiation Pneumonitis/ Pulmonary Fibrosis	Pulmonary Carcinoma	Age Range of Surviving Dogs (days after exposure)
0.75	48	9	4	1681-2458
1.5	96	35	3	1646-2452
3.0	72	35	8	1674-2460
All	216	79	15	1646-2460

*Dogs which had both pneumonitis/fibrosis and carcinoma were counted twice.

DISCUSSION

The initial pulmonary burdens of ^{239}Pu in these studies represent a continuum of activity levels from very low, i.e., equivalent to a "maximum permissible lung burden" in man, to very high, i.e., levels capable of producing death from radiation pneumonitis within a few hundred days. The dogs were exposed 1600-2500 days ago, and thus far only dogs with high pulmonary burdens have shown biological response.

Lymphopenia, the first biological effect observed, was seen as early as 180 days after exposure. Those dogs with severe lymphopenia usually developed clinical signs of radiation pneumonitis. Dogs with only a moderately depressed lymphocyte count did not always develop clinical disease, and in some dogs lymphocyte counts returned to near normal three to four years after exposure. No depression of neutrophil counts have been observed, in contrast to those seen in dogs inhaling $^{238}PuO_2$.

Radiation pneumonitis was the earliest cause of death, as it was for dogs inhaling monodisperse $^{238}PuO_2$ (4). At four to seven years after exposure, pneumonitis and fibrosis continue to be significant findings at death. Given the relative insolubility of the $^{239}PuO_2$ particles used in these studies, and the long retention times of the particles in lung, it is possible that pneumonitis and fibrosis may continue to be important biological effects, perhaps for the entire duration of the study. This cannot be stated with certainty, however, because a dose or dose rate threshold might exist, below which the radiation pneumonitis and pulmonary fibrosis either do not occur, or if they occur, are either subclinical or are repairable.

Several differences in the development of radiation pneumonitis/pulmonary fibrosis have been noted for the dogs inhaling $^{239}PuO_2$ vs $^{238}PuO_2$. The distribution of fibrosis for the former case has been more uniform and widespread than for the latter. This most likely reflects the relatively larger fraction of lung irradiated for equal lung burdens of $^{239}PuO_2$ vs $^{238}PuO_2$. Additionally, beyond 1000 days after exposure, very few cases of pneumonitis/fibrosis have been observed for the dogs with ^{238}Pu. Their major finding at death has been bone cancer.

Three years after exposure, pulmonary tumors began to be found as incidental findings at necropsy, similar to the findings with $^{238}PuO_2$. At later times, the cancers caused death and the incidence rate appears to be increasing. However, it is too early to predict the ultimate pattern of tumor incidence with respect to time or alpha radiation dose. It must be remembered that the tumor incidence observed thus far encompassed dogs that have died between 4.5 and 7.0 years after exposure. Likewise, the surviving dogs have a similar range of times after exposure. Therefore, the number of tumors observed to September 30, 1983, cannot be used as is to determine cancer incidence rates. Presently the average lung doses to death for those dogs dying with lung carcinomas increased with increasing particle size. If this pattern continues in time, these studies may provide key information in evaluating the role of nonuniformity of alpha radiation dose, and of "wasted" radiation on the induction of lung cancer by inhaled insoluble particles containing alpha-emitting radionuclides.

In summary, studies of the biological effects in Beagle dogs from inhaling widely differing activity levels of insoluble monodisperse $^{239}PuO_2$ aerosols are continuing. Presently very few dogs with initial pulmonary burdens greater than 3 kBq/kg are alive. Of the survivors, the projected range of lifetime doses to lung are 0.2 to 70 Gray. These developing results will provide important insights into the role of nonuniformity of alpha radiation in lung (and extra-pulmonary tissues as well) and of the risk of biological effects from inhaling insoluble alpha-emitting aerosols in dogs and ultimately in man.

Acknowledgments:

The authors gratefully acknowledge the contributions of the professional and technical staff of the ITRI, in particular those of the aerosol science and exposure groups, animal care, analytical radiochemistry, pathology, and radiobiology groups.

REFERENCES

1. Guilmette, R. A., Diel, J. H., Muggenburg, B. A., Mewhinney, J. A., Boecker, B. B., and McClellan, R. O., 1983, *Int. J. Rad. Biol.* (submitted).
2. Raabe, O. G., Boyd, H. A., Kanapilly, G. M., Wilkinson, C. J., and Newton, G. J., 1975, *Health Phys.* 28, 655.
3. Mewhinney, J. A. and Diel, J. H., 1983, *Health Phys.*, 45, 39.
4. Muggenburg, B. A., Mewhinney, J. A., Merickel, B. S., Boecker, B. B., Hahn, F. F., Guilmette, R. A., Mauderly, J. L., and McClellan, R. O., 1980, Proceedings of the Vth ICRP Congress, Vol. II, pp 115-118.

CYTOGENETICS IN RADIATION PROTECTION - BENEFITS AND LIMITATIONS

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The purpose of this presentation is to examine the various applications of human chromosome analysis in radiation protection work. The objective is to determine which of these applications have been useful and which have been misleading. To make this assessment the radiation cytogenetic studies carried out in our clinical cytogenetics laboratory since 1960 have been reviewed. We have also reviewed the pertinent radiation cytogenetics literature. The most useful applications have been in the recongnition and quantitation of individual overexposures as well as the ruling out of suspected but not actual overexposures. Problems have been noted relating to control exposure calibrations, and the interaction of extraneous agents. Another major application has been in the evaluation of populations receiving relatively low radiation exposures briefly, or intermittently over prolonged time periods. These have been productive on occasion but have also been misleading when limitations related to sample size considerations, selection of appropriate controls, and the lack of predictive power have not been recongnized. In summary, cytogenetic studies have an important place in radiation protection but their limitations must be recongnized adequately. Examples from our laboratory and the literature will be provided.

DOSE ESTIMATION BASED ON CHROMOSOMAL ABERRATIONS IN HUMAN PERIPHERAL LYMPHOCYTES

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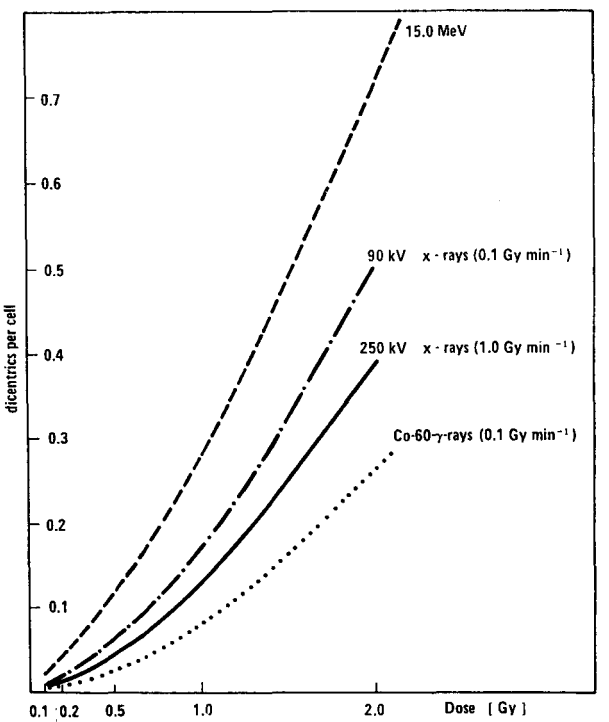
For dose estimation it is necessary to have a calibration curve which relates the radiation dose to the chromosomal aberration yield. In vitro curves were constructed for dicentric aberrations for several qualities of high and low LET radiation (Fig. 1).

Figure 1. Yields of dicentrics plotted against the dose for several radiation qualities (in vitro curves)



Data obtained on the aberration yield after exposure of cells by low LET radiation were found to be best fitted to the linear quadratic function $y = \alpha D + \beta D^2$, where y is the aberration yield, D the dose and α and β are the coefficients (Table 1). However, after irradiation with 15 MeV neutrons the curve shows a closer approximation to a linear function. The quotient α/β describes the dose, where the number of dicentrics produced by a

single and two ionizing track events is equal. Below these doses the majority of aberrations will be produced by single tracks. Thus, modifying the dose rate of low LET radiation, the dicentric rate will be mainly affected by higher doses where most of the dicentrics are induced by pairs of ionizing tracks represented by the term βD^2 in the equation.

Table 1. Coefficients α and β in the equation $y = \alpha D + \beta D^2$

Radiation type	$\alpha \pm SE \times 10^{-2}$	$\beta \pm SE \times 10^{-2}$	α / β
neutrons $\bar{E}$ = 15.0 MeV	24.9 ± 4.3	7.1 ± 2.0	3.5
250 kV X-rays (1.0 Gy min^{-1})	4.8 ± 0.3	7.4 ± 0.1	0.65
250 kV X-rays (0.4 Gy min^{-1})	7.7 ± 0.2	6.85 ± 1.07	1.12
250 kV X-rays (0.1 Gy min^{-1})	8.3 ± 0.1	5.03 ± 0.3	1.65
90 kV X-rays (0.1 Gy min^{-1})	7.3 ± 0.1	9.17 ± 0.4	0.8
Co-60-γ-rays (0.1 Gy min^{-1})	3.0 ± 0.1	5.3 ± 0.5	0.57

From this it is concluded that in practice the dose rate is hardly influenced by the dicentric yields found after an over-exposure of low LET radiation since in most cases of overexposure doses occur being lower than the described quotient α / β .

Each laboratory that performs "chromosome dosimetry" should have its own calibration curve because of the inter-laboratory differences.

From October 1982 to October 1983, 25 blood samples arrived per mail for "chromosome dosimetry". Table 2 shows that in the cases investigated there is a time lapse between exposure and sampling time. Therefore it is of great advantage to have an in-indicator for radiation exposure that remains detectable for some time after exposure.

Table 2. Time interval between radiation exposure and blood sampling

days	cases
7;10	2
11 - 30	1
> 30	8
chronic exposure	8
unknown	7

For dicentric chromosomes it was shown that within a few weeks after the exposure no remarkable alteration in the dicentric rate is observed (Preston, Brewen et al., Radiat. Res. 1974; 60, 516). On the other hand, it is also possible to estimate exposures after a longer interval between exposure and blood sampling. In two cases of an accidental whole body irradiation doses were estimated 30 weeks after the accident which were in good agreement with the film badge values (Stephan, Hadnagy et al. Health Physics 1983; 44, 409).

Table 3 demonstrates the importance of "biological dosimetry" based on chromosomal aberrations, particularly in those cases where physical dosimetry is not sufficient:

- known exposure, but no blackness on film badge
- possible exposure but film badge was not worn
- values of film badges are underestimating whole body dose because of the limited body region measured
- values from film badges are too high to be indicating the actually received dose

Table 3. Cases analysed

known exposure	6
film badge indicates a questionable exposure	5
suspected exposure of persons not wearing a dosemeter	5
chronic exposure	9
external: 4 internal: 5	

Table 4 shows the estimated doses based on chromosomal aberrations indicated as equivalent whole body doses.

Table 4. Numbers of estimates

Dose range (Gy)	cases
0 - 9	11
10 - 14	3
15 - 19	3
20 - 24	0
25 - 29	1

In most cases the equivalent whole body dose was estimated to be less than 0.1 Gy. This value appears to be the lower limit of sensitivity by using the chromosome aberration method. This limit is indicated by two factors: the spontaneous dicentric rate and the confidence limits of dose estimations.

The aberration rate differing from the spontaneous rate (7 dicentrics/13,741 lymphocytes) will give a dose estimate of about 0.05 Gy. The confidence limits are influenced by the number of scored cells (Table 5) and the frequency of dicentrics per unit dose.

Table 5. Influence of the number of cells examined on the 95% confidence limits (250 kV x-rays; 1 Gy min^{-1})

No. of cells	Dose estimate (Gy)	92% confidence limits (Gy)	
250	0.14	0.03	0.36
500	0.14	0.05	0.28
1000	0.14	0.06	0.23
1500	0.14	0.08	0.21
2000	0.14	0.09	0.20

In cases of high LET radiation the lower limit of the estimated doses is less in comparison to low LET radiation.

The results show that "chromosome dosimetry" may be of practical use for dose estimation and provides an important supplement to physical methods.

These investigations were supported by the Bundesministerium des Innern

THE SIGNIFICANCE OF SERUM THYMIDINE CONCENTRATION AS BIOCHEMICAL INDICATOR OF RADIATION EXPOSURE

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In accidental cases of high level radiation exposure of persons, the actual exposure of the organism must be determined independent from physical dosimetric measurements. For suitable medical treatment, risk assessment and also for occupational legislative considerations, such determination is of primary concern. This requires a biochemical in vitro method (or combination of methods) which is specific, reproducible, significantly dosedependent and quickly and simply performed in the laboratory without undue burden to the patient from sample extraction. In addition, the method should allow for a dose assessment immediately as well as after a sufficiently long period of time following the exposure. Upon a comprehensive study of the available literature (1,2), a method was developed that may be suitable for dose assessment using biochemical indicators.

1. Determination of thymidine concentration in mice serum

Serum of whole-body irradiated mice inhibits the incorporation of 125-IUdR into the DNA of L929 cell cultures (3). The effect depends on dose and time and has its maximum at about 4 hours after acute exposure to 0.005 - 1 Gy. The corresponding humoral factor behaves identical to thymidine. Considering the possible use of higher levels of thymidine concentration as a biochemical indicator of radiation exposure, we reproduced Feinendegen's mice experiments (3) as an initial step (Fig.1).

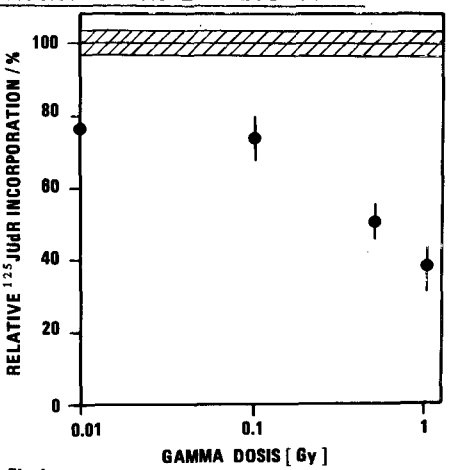

Fig. 1

2. Determination of thymidine concentration in human serum from healthy persons

The extrapolation of the results from mice to man is the objective of this study while considering the possible use of increased levels of thymidine concentration as a biological indicator of radiation exposure. In contrast to mice serum, human serum contains low levels of thymidine concentration - extremely low for a direct indication - and two factors that inhibit incorporation and differ in molecular weight: F_1 < MG 500, F_2 = MG 30.000 - 50.000 (3). All of the above renders an extrapolation to man more difficult.

2.1 Optimization of method, considering the low thymidine concentration in human serum and the interfering serum factors

2.2.1 Preliminary treatment of human serum

For separating the 125-IUdR incorporation inhibiting factor F_2, as well as further serum components of high molecular weight, the blood was ultra-filtrated (Diaflo-membrane, nominal separation limit 500 daltons). The yield was 70 - 80%.

2.2.2 Optimization of incubation time

It had to be clarified whether the assay system developed for mice serum requires the same incubation time for human serum. The incorporation kinetics of 125-IUdR in the DNA of L929 cells was examined after 15, 30, 45 and 60 min., respectively. Fig. 2 shows the results. The relative incorporation rate is fairly identical within these time spans. Therefore, we proceeded with an incubation time of 60 min, as was done in mice experiments.

2.2.3 Optimization of method in view of serum concentration

By increasing the serum concentration in the culture medium, the method can be adjusted to the lower thymidine concentration for improved sensitivity. For this purpose the assay system was tested for 10%, 20%, 40% and 60% serum, repectively. Results are shown in Fig.3.

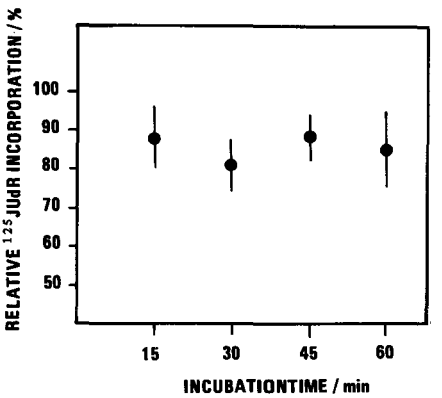

Fig. 2

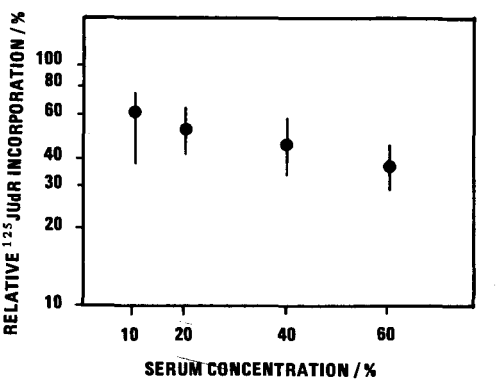

Fig. 3

3. Serum from patients receiving radiotherapy

First examinations on patients receiving radiotherapy (Tab. 1) yielded contradictory results. Therefore additional investigations are necessary.

Literature

1.) Stamm, A., Weitz, B., Bögl, W.
Biochemische Indikatoren für Strahlenexpositionen, Teil I
Bericht des Institutes für Strahlenhygiene des Bundesgesundheitsamtes, ISH 4, Juli 1981

2.) Stamm, A., Bögl, W.
Biochemische Indikatoren für Strahlenexpositionen, Teil II
Bericht des Institutes für Strahlenhygiene des Bundesgesundheitsamtes, ISH 13, März 1982

3.) Feinendegen, L.E., Mühlensiepen, H., Porschen, W., Booz, J.
Acute non-stochastic effect of very low dose whole-body exposure, a thymidine equivalent serum factor
Int. J. Radiat. Biol. 41 (1982) 139-150

Table 1: Some results of examinations on radiotherapy-patients.

Case No. Sex Year of Birth	Diagnosis	Radiation Field	Field Size /cm/ Typ of Radiation	FHA/cm MD/Gy HD/Gy	Sampling (blood) h p.r.	125-IUdR-Incorporation⊕ before Irradiation %	125-IUdR Incorporation⊕ post Irradiation
10 male 1938	Kidney tumor T_2, N_x, M_x	1) Kidney tumor ventr., right 2) Kidney tumor dors. and lat., right	15x15 12 MeV X-rays	100 1) 2,92 2) 2,66 4,0	4	92 70	75 (Serum 10%, UF 25.000, Incubation Period 15 min) 85 (Serum 20%, UF 25.000, Incubation Period 15 min)
12 male 1928	Metastases of Bronchial-Ca.	Skull lat, right	14x19 Co-60	80 2,92 2,0	5	100 50	60 (see above) 54 (see above)
1 male 1913	Carcinoma of the Bladder pT_{2-3}, N_0, M_x	Pelvis	15x15 15 MeV X-rays	100 2,74 2,0	6	58 69 69	63 (Serum 10%, Heat Treatment, Incubation Period 60min) 93 (Serum 10%, UF 50.000, Incubation Period 60 min) 110 (Serum 10%, UF 25.000, Incubation Period 60 min)
2 female 1913	Rectum Ca.	Pelvis	14x12 X-rays	100 2,55 3,0	4	65 82 100	57 (see above) 89 (see above) 103 (see above)
5 female 1906	Cervix Ca. Stage IIb	Pelvis dorsal	15x16,5 15 MeV X-rays	100 2,68 2,0	2,25	83	66 (Serum 100%, UF 25.000)

6 female 1917	Ca. of Uterus, Bladder, Collum	Pelvis ventral dorsal	14,5x14,5 15 MeV X-rays	100 2,42 1,8	2,25	70 60	85(Serum 30%,UF 25.000) 85(Serum 50%,UF 25.000)
3 male 1908	Metasta- ses of Prostata	right Shoulder	14x8 Co-60	100 2,83 2,5	2	80 - 85	75(Serum 40%,UF 500,Incubation Period 60 min)
7 female 1925	Collum- Ca.	Pelvis dorsal	17x16 15 MeV X-rays	100 2,34 2,00	2,25	87	77(see above)
8 male 1913	Bladder- Ca.	Pelvis dorsal	15x15 15 MeV X-rays	100 2,74 2,00	2,25	95	87(see above)
11 female 1931	Met. Mamma-Ca.	Heart, Lungs, Stomach	16x7,5 Co-60	80 3,81 3,00	4	97	89(see above)

⊕The range of variation of the percentage values for incorporation is of 10 to 15 percent on the average

Abbreviations: FHA = distance between focus and tumor
MD = mean dose
HD = tumor dose
Ca. = carcinoma
UF = ultra-filtrate

THE SIGNIFICANCE OF ACID SERUM PHOSPHATASE AND SERUM AMYLASE AS BIOCHEMICAL INDICATORS OF RADIATION EXPOSURE

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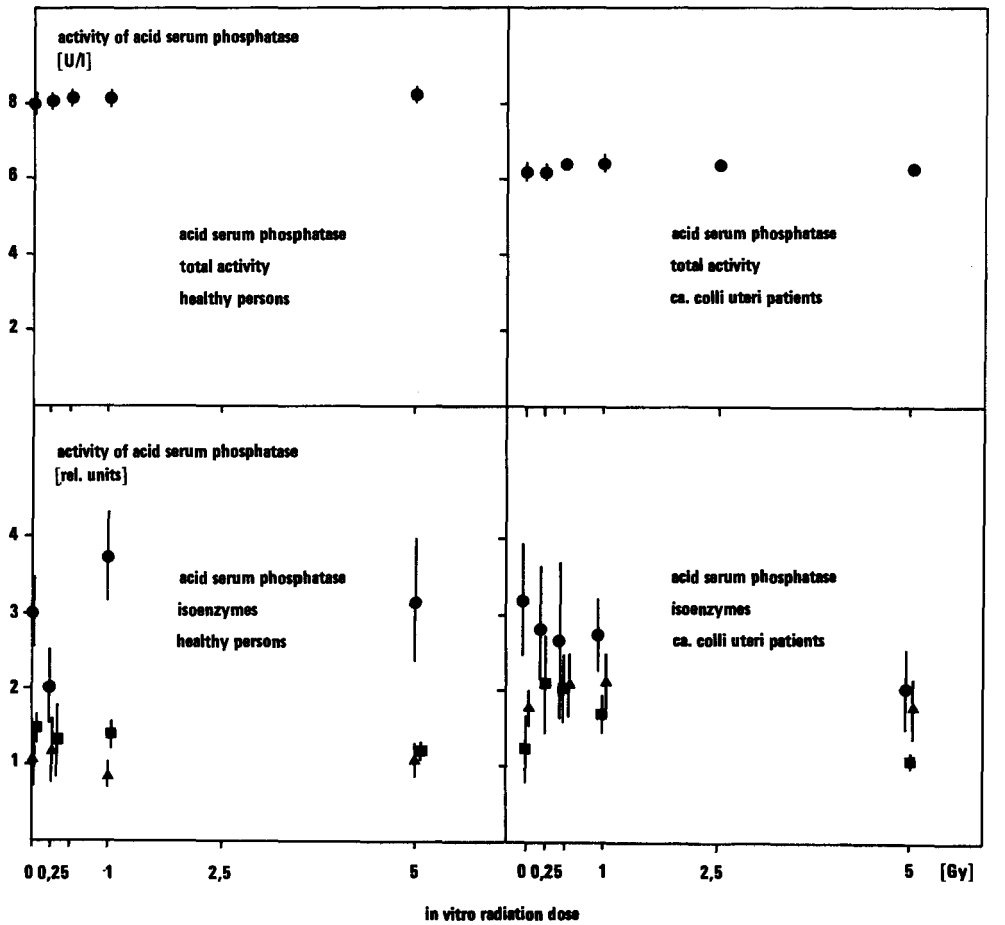
In accidental cases of increased radiation exposure of individuals and prior to medical treatment, the actual exposure of the organism must be determined independent of physical dosimetric measurements. This necessitates a biochemical in vitro method (or a combination of methods) which is specific, reproducible, significantly dosedependent and may be quickly and simply performed in the medical laboratory without undue burdening of the patient during sampling. In addition, it should allow for dose assessment immediately after exposure as well as after a sufficiently long period of time following the exposure.
From an extensive literature review (1,2), two methods were selected that may be suitable for dose assessment via biochemical indicators. These methods are currently examined on patients receiving radiotherapy.

1. Determination of Total- and Isoenzyme Activities of Acid Serum Phosphatase

Maruna et al. (Strahlentherapie 147 6 (1974) 624) reported on changes in total and isoenzyme activities of acid serum phosphatase after in vitro irradiation of blood samples (0.25 Gy) from patients with carcinoma colli uteri. The total activity of acid serum phosphatase was increased by 35-40% after irradiation. The activity of single isoenzyme fractions was either increased by about 20-25%, or decreased, respectively.
The isoenzymes were separated by thin layer electrophoresis with starch gel and activities were densitometrically determined with phenylphosphate as substrate. In view of the possible use of acid serum phosphatase as a biochemical indicator, respective blood examinations were initially performed on healthy persons and later on patients with carcinoma colli uteri as well.
Fresh blood was irradiated by x-rays (250 kV without filter, DL = 0.50 Gy/min) with doses from 0.25 to 5 Gy. The enzymes were separated by electrofocussing on polyacrylamidegel and the activities were densitometrically determined by 4-methylumbelliferylphosphate as substrate. A representative selection of results is shown in the figure.
In healthy persons and in carcinoma patients, there were no changes in the phosphatase activities at radiation doses of up to 5 Gy(the determination of acid phosphatase isoenzyme activities in serum is largely erroneous because of the low content). Not even by using the method according to Maruna (electrophoresis with starch gel, phenylphosphate) were we able to reproduce his results.



2. Determination of Serum Amylase Activity

Chen et al. (Radiation Research 54 (1973) 141) reported on the increase of serum amylase activity after radiation therapy. This increase was observed only after total body irradiation, i.e. after irradiation of the upper part of the body. Patients receiving radiation therapy in the lower body region, showed no significant change of amylase activity in serum. Indications are that the increase in amylase activity is caused by salivary glands in the oral region. The maximum increase occurred approx. 30 h after irradiation. After 60 h, the serum amylase activity returned to its initial value prior to irradiation. In view of a possible use of serum amylase activity as a biochemical indicator, respective examinations of radiotherapy patients were performed.

Some results of examinations on radiotherapy patients have been tabulated below. With patients identified Nr. 6 and 4, the salivary glands in the oral region were at least partially exposed to the irradiated field; a significant increase of serum amylase activity was seen at 24 h p.r.. In the case of patient Nr. 7, nearly 30% of the pancreas was exposed to the irradiated field; there was no increase of serum amylase activity. The remaining four patients with irradiated fields exterior to salivary glands and pancreas also showed no increase in serum amylase activities after irradiation. These results allow the conclusion that the increase of amylase activity is caused by the salivary glands in the oral region. A maximum increase was seen approximately 30 h after irradiation. After 60 hours, the serum amylase activity returned to its initial value before irradiation.

Sex (Case Nr.) Year of birth	Diagnosis	Irradiation	Serum amylase (U/l) before irrad./24h after irrad. (relation = after/before irrad.)
male (6) 1937	Tonsillar-Ca.l.	Epi-Mesopharyngeal fields (2,5 Gy HD) Neck-field (3,0 Gy HD)	189/1282 (6,8)
male (4) 1921	Larynx-Ca.	Larynx-fields (2,5 Gy HD)	158/395 (2,5)
female (7) 1924	Met.Mamma-Ca.	$^{+}_{+}$BWS-LWS-Ileosacral-field (2,5 Gy HD)	223/207 (0,9)
female (1) 1924	Corpus-Ca.	1,8 Gy HD mid-pelvis	200/203 (1,0)
female (2) 1913	Mamma-Ca.r.	HSIA x (3,0 Gy MD) Mamma-field (4,0 Gy MD)	161/139 (0,7)
female (3) 1913	Mamma-Ca.r.	HSIA x (3,0 Gy MD) Mamma-field (4,0 Gy MD)	134/91 (0,7)
female (5) 1950	Mamma-Ca.r.	HSIA x (3,0 Gy MD) Mamma-field (4,0 Gy MD)	219/22 (1,0)

HD = tumor dose
MD = mean dose
BWS = thoracospinal column
LWS = lumbar vertebral column

The following factors remain to be examined:
a) the radiation sensitivity of the different salivary glands in the oral region, and
b) a possible correlation between radiation dose and changes in enzyme activity.

References

1) A. Stamm, B. Weitz, W. Bögl
Biochemische Indikatoren für Strahlenexpositionen. Teil I. Eine Literaturstudie über Blutuntersuchungen nach Exposition mit Röntgen - und γ- Strahlen.
Report of the Institute for Radiation Hygiene of the Federal Health Office, ISH 4/1981.
2) A. Stamm, W. Bögl
Biochemische Indikatoren für Strahlenexpositionen. Teil II. Eine Literaturstudie über biochemische Veränderungen an biologischem Material nach Exposition mit Röntgen - und γ- Strahlen.
Report of the Institute for Radiation Hygiene of the Federal Health Office, ISH 13/1982.

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AGE-DEPENDENT RADIATION RISK IN MAN: CHROMOSOMAL INVESTIGATIONS

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The age-dependent reports about the chromosomal radiosensitivity are contradictory. Because chromosomal aberrations may be the source of both somatic and genetic disorders, age-dependent radiosensitivity is an important factor for radiation protection purposes.

The dicentric chromosome which is mostly used for those investigations will often result in the formation of anaphase bridges at the anaphase stage of mitosis. These bridges interfere with the mechanical separation of the two daughter cells. Such interference frequently leads to cell death. Therefore it is important that dicentrics are scored in first dividing cells after exposure. In earlier times the normal culture time for lymphocytes was 72h. Since it was learned that at this time many lymphocytes have passed the first cell division, culture time was reduced to about 56h and now the standard culture time is 48h. The aim is to avoid the inclusion of second division cells in the analysis as these would reduce the observed aberration yield. This reduction is a consequence of the multiplication of undamaged lymphocytes and the exclusion from the analysis of damaged cells.

It is now possible to identify reliably cells in their 1st, 2nd and subsequent in vitro division by using the technique of "Fluorescence plus Giemsa" (FPG) staining. The technique has also permitted to study the proliferating kinetic of lymphocytes in vitro. It is known that up to 60% of second mitosis (M2-cells) in cultures of human lymphocytes appear in 48h cultures. These findings have therefore cast doubts on the validity of much earlier cytogenetic research.

The following results present the cell cycle of lymphocytes in vitro, the dicentric rate in first and second dividing cells and the dicentric rate in different age groups.

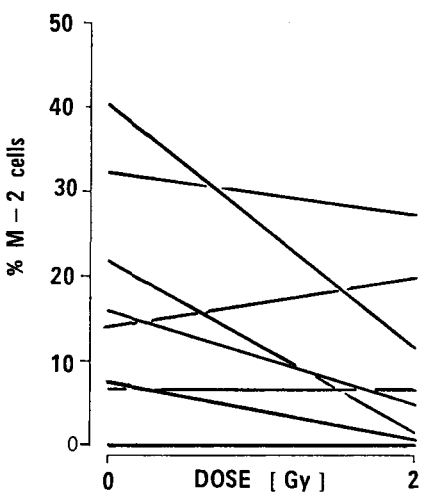
From 36 donors we have studied lymphocyte proliferation in non-irradiated and irradiated blood samples. Peripheral blood was irradiated in vitro with a single dose of 2 Gy and was cultured for 48h in the presence of BUdR (32 µM). As can be seen from Table 1 in non-irradiated samples, the proportions of M2-cells vary between 0 and 40%. In the donors aged 13 to 48 years, the proportions of M2-cells range, in most cases, between 0 and 10%. In contrast hereto it has been found that systematically 50% of cells with second cell cycle appeared in the 48h cultures of all cord blood samples (10 newborns). However, we have found only 30% in this particular age group.

Table 1. Proportions of M2-cells in non-irradiated (Co) and irradiated (2 Gy) blood samples

% M – 2	Co	2 Gy
	donors (13 - 85 yrs)	
0 - 10	16 (61.5 %)	22 (84.8%)
11 - 20	6 (23.2%)	2 (7.6%)
21 - 30	3 (11.5%)	1 (3.8%)
31 - 40	1 (3.8%)	1 (3.8%)
	26 (100%)	26 (100%)
	cord blood	
0 - 10	2 (20%)	8 (80%)
11 - 20	5 (50%)	1 (10%)
21 - 30	1 (10%)	1 (10%)
31 - 40	2 (20%)	0 (0%)
	10 (100%)	10 (100%)

After irradiation it can be seen that the proportions of M2-cells are reduced. This is to be expected, since it is known that cell cycle delay is induced by ionizing radiation. Moreover, regarding some individual cases, it is evident that there is no unique proliferation kinetic. In some cases the proportions of M2-cells after irradiation are lower up to 20% as compared with control values, in other cases the proportions of M2-cells are comparable to those in non-irradiated and irradiated samples. In one experiment the proportion of M2-cells is higher after irradiation, as compared with controls, for which fact we have no explanation.

Figure 1



These different proportions of M2-cells in cultures of 48h are of importance when assessing chromosome damage in lymphocytes after exposure to chromosome breaking agents. Accurate information on the aberration rate can only be gained from first metaphases, as shown in Table 2.

Table 2: Dicentrics in 1st and 2nd metaphases

division	scored cells	dic + ace	dic	dic + ace / cell ± SE	dic /cell ± SE	Σ dic /cell ± SE
1st	12.009	4.344	145	0.362 ± 0.005	0.012 ± 0.001	0.374 ± 0.006
2nd	459	57	30	0.124 ± 0.016	0.065 ± 0.012	0.189 ± 0.020
1st + 2nd	12.468	4.401	175	0.354 ± 0.005	0.014 ± 0.001	0.367 ± 0.005

M1-cells were identified by homogenous stained chromosomes. In 12.009 first mataphases, we have found 4.344 dicentrics with associated fragments and 145 dicentrics without fragments (3%). As second metaphases only those were scored where the sister-chromatids showed different staining patterns in all chromosomes of a metaphase. In 459 second metaphases we found 57 dicentrics with associated fragments and 30 dicentrics without fragments. It is remarkable that in our study we have found dicentrics without fragments in the first metaphases and dicentrics with associated fragments in second metaphases.

The dicentrics without fragments in M1-cells may perhaps indicate that some fragments are very small and not detectable. The dicentrics in M2-cells should not have an associated fragment, but it is known that dicentrics can be produced by isochromatid exchanges. The joining of the centric portions of two isochromatid deletions gives what appears to be a chromosome-type dicentric. It is only distinguishable from the true dicentric in case there are two fragments both of which show sister union, i.e. the chromosome-type dicentric can be distinguished from that which arises from chromatid deletions only be the shape of the fragments. Perhaps it may be assumed that the number of dicentrics with associated fragments partially reflects a toxic effect of BUdR. It is well known that treatment with BUdR causes a reduction in the number of metaphases and a slowing of the cell cycle. The dicentrics with associated fragments in second metaphases however seem to be of the chromosome type. This assumption is supported by the fact that we have found the same number of chromatid aberrations in the culture containing BudR as in those without BUdR.

Regarding the dicentric rates in M1- and M2-cells, it can be seen that in M2-cells only half of those dicentric rates are found in M1-cells. From this it can be concluded that in lymphocyte populations with high proportions of M2-cells a dilution of the dicentrics may take place.

For the investigation of a chromosomal age-dependent radiosensitivity, M1-cells only were scored for dicentrics (dic) and ringchromosomes (cr). The aberration yield ranges between 0.297 and 0.498 dic + cr per cell.

The age-dependent radiosensitivity is determined by comparing the mean yield of chromosome aberrations from 4 age-groups comprised of 47 donors (10 newborns, 11 donors 13-16 yrs, 11 donors 25-36 yrs and 15 donors 51-85 yrs). The incidence of dicentric chromosomes is following the Poisson distribution ($p = 0.36$) and the mean yield of chromosome aberrations is comparable in these 4 age groups (Table 3, p=C.19). The radiosensitivity seems not to be associated with the donors' sex (for males p=0.30 and females p=0.20). Therefore, these data cannot support an age-dependent radiosensitivity in man.

Table 3. Mean yields of chromosome aberrations, dic + cr (s.e) per 100 M1-cells, from 4 age-groups

Newborns	13-16 yrs	25-36 yrs	51-85 yrs	All groups
41.0 (6.5)	38.8 (3.3)	36.2 (4.1)	37.4 (5.0)	38.1 (5.0)

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MIKROKERNRATEN NACH IN VIVO BESTRAHLUNG

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Einleitung

Verschiedene Gründe gaben Anlaß, die strahleninduzierten Mikrokernraten bei Lymphozyten des peripheren Blutes von Strahlentherapiepatienten und Kindern, nach einer Röntgenstrahlendurchleuchtung (Herzkatheter-Untersuchungen), zu ermitteln.

- Für die Zwecke des praktischen Strahlenschutzes werden häufig in den Fällen, in denen die Personendosis nicht mit physikalischen Dosimetern gemessen wurde, die Chromosomenaberrationsrate als Maß der Strahlendosis herangezogen. Zur Bestimmung der Chromosomenaberrationsrate müssen einige Hundert bis Tausend Metaphaseplatten mit entsprechend großem personellen Arbeitsaufwand analysiert werden. Die Bestimmung der Mikrokernraten ist dagegen wesentlich einfacher. Da auch die Struktur von Mikrokernen im Vergleich zu der von aberrierten Chromosomen wesentlich einfacher ist, besteht die berechtigte Hoffnung, die Mikrokernanalyse automatisieren zu können. Obwohl in vitro Untersuchungen zeigten, daß eine strenge Korrelation zwischen der Mikrokern- und der Chromsomenaberrationsrate besteht (1) (2), muß die Dosiseffektkurve für Mikrokerne nach in vivo Bestrahlung und ihre Abhängigkeit von der Zeit noch untersucht werden.

- In strahlenbiologischen Untersuchungen wurden die Dosiseffektkurven für Chromosomenaberrationen im Bereich von mindestens 0,5 Gy bis etwa 10 Gy zumindest nach in vitro Bestrahlung sehr intensiv untersucht. Bisher wurde jedoch der Dosisbereich unter 0,5 Gy weitgehend ausgespart. In diesem Zusammenhang erscheint es sinnvoll, diese Lücke durch Untersuchungen der Mikrokernrate, als Maß der Chromosomenschädigung bei Personen mit geringer Strahlenbelastung, wie sie z.B. bei der Herzkatheteruntersuchung auftritt, zu füllen.

Methoden und Ergebnisse

Zur Bestimmung der Mikrokernraten wurden die Leukozyten von den Erythrozyten getrennt und in üblichen Kulturansätzen nach Stimulation mit Phytohämagglutinin und Bebrütung bei 37°C zur Teilung gebracht. Die Kulturdauer wurde so angelegt, daß alle Zellen mindestens eine Teilung vollständig durchlaufen haben. Wie Vorversuche zeigten (1) (2), ist eine Bebrütungszeit von 72 h bezüglich der Mikrokernausbeute und der Bearbeitungszeit optimal. Zur Auszählung wurden die Mikrokerne nach folgenden Kriterien identifiziert.

1. Durchmesser des Mikrokerns höchstens gleich 1/2 Hauptkerndurchmesser

2. Abstand des Mikrokerns zum Hauptkern höchstens 3 mal so groß wie der Hauptkerndurchmesser

3. Gleiche Strukturierung von Mikrokern und Hauptkern und gleicher oder geringerer Grauwert.

4. Pro Hauptkern wurden höchstens 2 Mikrokerne registriert.

Bei 17 Radiotherapiepatienten mit Lungenkarzinomen wurden Blutproben jeweils unmittelbar vor und nach der 1. und 2. Teilbestrahlung entnommen und untersucht. Der zeitliche Abstand zwischen 1. und 2. Bestrahlung lag zwischen einem und drei Tagen. Die Bestrahlung erfolgte in der Regel am Betatron, in einem Fall wurde zusätzlich eine Bestrahlung mit ^{60}Co-Gammastrahlen vorgenommen. Aus der Herddosis, der Größe des Bestrahlungsfeldes, der Dicke des durchstrahlten Feldes und dem Körpergewicht wurde die über dem Körper gemittelte Strahlendosis berechnet. Die Ergebnisse der Untersuchungen sind in Tabelle 1 enthalten.

Bei 16 Kindern, die während der Herzkatheterisierung einer Röntgendurchleuchtung ausgesetzt waren, wurden Blutproben unmittelbar vor und nach der Durchleuchtung sowie 24 Stunden später entnommen. Die Strahlendosen wurden mit auf der Körperoberfläche angebrachten Thermolumineszenzdosimetern gemessen. Aus den Dosen auf der Strahleneintritts- und austrittsseite wurde unter der Annahme eines exponentiellen Dosisabfalls die mittlere Dosis berechnet. Ansonsten wurde, wie bei den Radiotherapiepatienten beschrieben, die über den Körper gemittelte Dosis bestimmt. Die Ergebnisse der Untersuchungen unmittelbar nach der Röntgendurchleuchtung sind in Abb. 1 wiedergegeben, die 24 Stunden später festgestellten Mikrokernraten in Abb. 2. Die eingezeichneten Regressionsgeraden wurden auf die übliche Weise ermittelt.

Diskussion

Die Ergebnisse zeigen, daß selbst bei den geringen Strahlendosen nach Durchleuchtung die Mikrokernrate signifikant erhöht ist. Es ist auffallend, daß die Rate höher ist als aufgrund der Extrapolation von bekannten Dosiseffektkurven zu erwarten wäre. Dies ist nicht der Fall bei den Radiotherapiepatienten. In beiden Gruppen ist jedoch ein schneller Rückgang der Mikrokernraten mit der Zeit zu beobachten.

Literatur

(1) Countryman, P.J., Heddle, J.A.
The production of micronuclei from chromosome aberrations in irradiated cultures of human lymphocytes
Radiat. Res. 41 (1976) 321-332

(2) Grillmaier, R.E., W. Schmidt, H.-K. Stanger, R. Dietz
Studies on radiation induced micronuclei and chromosome aberrations.
Radiat. Environm. Biophys. 19/4 (1981)

Micro Nuclei Rates and Radiation Doses of Radiotherapy Patients

Patient No.	Micro Nuclei Rates				Whole Body Averaged Dose (Gy)	
	Before 1st Irradiation	Immediately after 1st Irr.	Before 2nd Irradiation	Immediately after 2nd Irr.	1st Irrad.	1st + 2nd Irrad.
1	1.2	4.6	2.6	5.8	0.228	0,456
2	1.7	2.1	2.5	7.4	0.193	0.385
3	1.4	3.0	2.0	5.9	0.219	0.219
4	1.9	3.8	2.9	4.8	0.191	0.382
5	1.0	2.2	3.1	-	0.181	0.363
6	2.8	4.6	3.8	4.2	0.203	0.407
7	1.4	2.5	2.0	5.3	0.218	0.438
8	2.1	3.6	2.1	3.1	0.23	0.459
9	1.7	2.1	3.0	3.0	0.296	0.592
10	2.1	4.1	2.0	2.8	0.086	0.172
11	1.0	4.6	-	-	0.212	0.424
12	1.5	2.8	1.0	4.6	0.258	0.516
13	1.8	3.4	1.7	-	0.07	0.14
14	1.6	3.7	2.0	3.9	0.229	0.438
15	1.7	2.8	2.5	5.0	0.266	0.532
16	2.0	2.8	1.9	3.9	0.221	0.442
17	1.8	3.0	2.8	4.0	0.352	0.704
$\bar{x}$	1.7	3.3	2.4	4.6	0.215	0.416
s	0.44	0.86	0.67	1.28	0.066	0.142

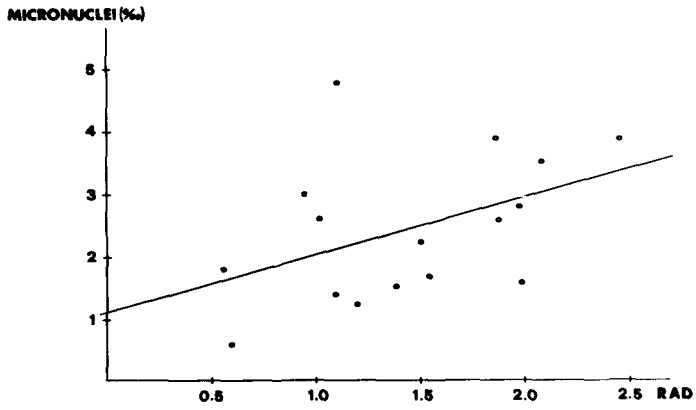

Fig. 1: Micronucleirate (per 1000 cells) immediately after (≤ 10 min.) radiation exposure (fluoroscopy)

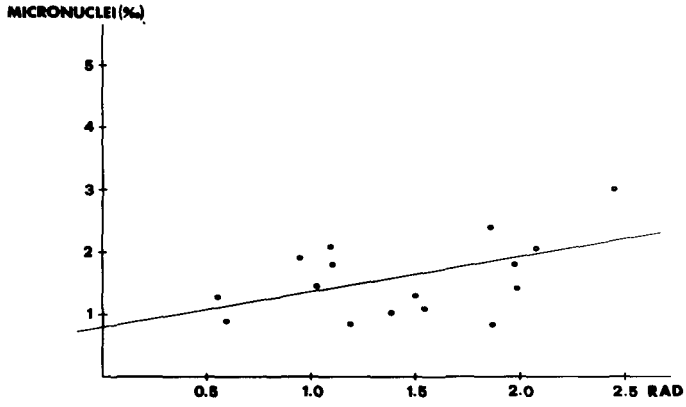

Fig. 2: Micronucleirate 24 hours after radiation exposure

CYTOGENETIC INVESTIGATIONS ON CHINESE HAMSTERS AFTER INTERNAL ALPHA RADIATION EXPOSURE

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Objectives

The previously applied x-ray contrast medium "Thorotrast" (colloidal ThO_2) induced late effects in Thorotrast patients, which are being investigated within a joint research project ("Deutsche Thorotraststudie") including also animal experiments. The aim of this study is to examine that proportion of damage, which is to be attributed to the radioactivity of the incorporated Thorotrast. Another purpose is the determination of the total effective radiation dose during the exposure time under consideration (1).

The subproject "cytogenetic investigations" is dedicated to the study of chromosome aberrations induced by incorporated radionuclides. To date no definite correlation could be established between the chromosome aberration rate of lymphocytes in the peripheral blood of Thorotrast patients and the radiation exposure due to Thorotrast. [However, varying quantities were used as a measure for the radiation dose: exposure time (2, 3), whole body activity (4) and the so-called ^{224}Ra-equivalent (5)]. In the meantime more accurate assessments are available for the dose in lymphocytes of Thorotrastpatients and of Thorotrast-injected Chinese hamsters. They take into account the relevant organ doses and the specific lymphocyte biology (6, 7) and permit with improved precision the investigation of the dose effect relationship for chromosome aberrations after Thorotrast injections.

Specifically, the following aspects are being examined:
- radiation effect as function of dose
- influence of specific activity or dose rate
- comparison of protracted internal and external radiation exposure
- non-radiation effect of the colloids

Methods

Chinese hamsters (Cricetulus griseus), which are cytogenetically well analyzed, were injected with Thorotrast (Table 1, group Th) and exposed for times corresponding to those in Thorotrast patients. In addition to Thorotrast another colloid (Zirconotrast) was used, which shows physico-chemical properties comparable to Thorotrast, but is non-radioactive (Table 1, group Zr-0). Different emission rates could be obtained by enriching Zirconotrast (Table 1, group Zr-1, Zr-10, Zr-100) with Thorium ($^{228/230}Th$). In order to correlate aberration rates and radiation dose for the internal irradiation, animals were exposed to ^{60}Co-gamma whole body irradiation with comparable dose rates and exposure times ("biological calibration").

Chromosomes, being in metaphase stage of mitosis, were prepared from whole-blood lymphocyte cultures using routine techniques (8) however with small amounts of blood per culture. Blood was collected from animals after varying exposure times for each group irradiated,

in order to obtain a dose effect curve for each group (9). Statistical analysis: Curve fitting by maximum likelihood methods. (We would like to thank G. Gerber/Mol, Belg.). Uncertainties have been evaluated assuming Poisson distribution.

Results

The evaluation of the aberration rates gave the following results:

- a definite dose effect relationship for chromosome aberrations in lymphocytes of Chinese hamsters (Figure 1, 2) was established
- the aberration rate at a given dose decreases with increasing colloid activity (Figure 3)
- the aberration yield obtained in these protracted irradiation experiments (internal and external) is a factor of ten to twenty lower than for acute irradiation (10)
- no significant non-radiation effect was observed (Table 2)

Table 1: EXPERIMENTAL AND CONTROL GROUPS

A total of 600 Chinese hamsters was examined during a period of one year. Alpha emission rate one corresponds to an estimated annual lymphocyte dose (in Chinese hamsters) of 0.016 Gy.

	INTERNAL injected solutions			EXTERNAL ^{60}Co-Gamma
relative alpha emission rate	Zirconotrast	Thorotrast	NaCl	
0	Zr-0		Contr.	
1	Zr-1	Th		
10	Zr-10			
100	Zr-100			
				constant dose rate: 0.017 Gy/d

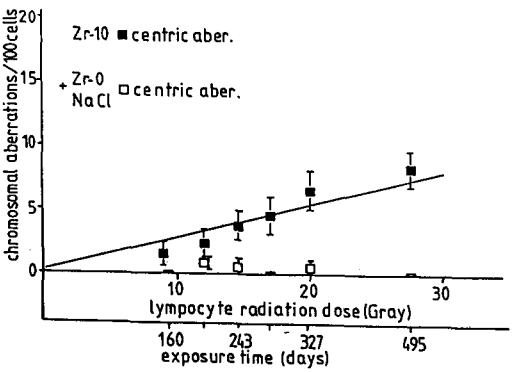

Fig. 1: Dose response curves for the induction of centric chromosome aberrations in lymphocytes of Chinese hamsters for internal alpha irradiation with Zr-10 (■) and for control experiments with Zr-O and NaCl (□).

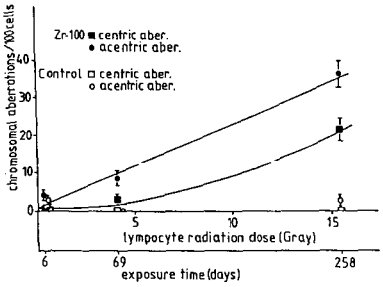

Fig. 2: Dose response curves for the induction of centric and acentric chromosomal aberrations in lymphocytes of Chinese hamsters for internal alpha irradiation with Zr-100 (■, •) and for control experiments (□, o).

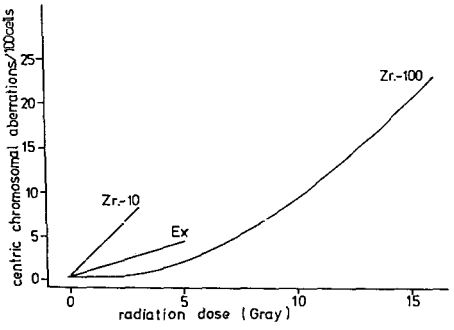

Fig. 3: Comparison of dose effect relationships for the induction of centric aberrations by incorporated colloids with different specific alpha emission-rates (Zr-10, Zr-100) and by ^{60}Co-Gamma irradiation (Ex).

Table 2: CHROMOSOME ABERRATION RATE FOR ANIMALS USED IN THE CONTROL GROUPS

Corresponding group	irradiated	Number of animals	Cells examined	Centric/ 100 cells	Acentric/ 100 cells
Control group for internal exposure	injected solution: NaCl	21	1588	0.31±0.14	1.70±0.33
	injected solution: Zr-O	17	1123	0.62±0.24	1.60±0.38
Control group for external exposure	untreated	23	1578	0.19±0.11	1.65±0.32

Conclusions

1. The fact that the aberration rates for chronic internal irradiation are lower than those for acute one may be due to competing processes: on the one hand the expected increase of aberrations with increasing radiation dose, and on the other hand the decrease of the aberration rates because of the limited lifetime of lymphocytes and the related fading of aberrations due to lymphocyte death.
2. The lower effectiveness of colloids with higher alpha emission rate for aberration induction is possibly related to increased

radiation damage impairing lymphocyte vitality. Inhomogeneity of dose distribution and thus dose waste effects, which increase with specific activity of the colloids, may be another reason for lower effectiveness of the colloid with higher alpha emission rate.

3. Therefore, at present, it is not possible without additional experiments
   a) to derive a RBE for chronic irradiation from the comparison of aberration rates for external and internal irradiation
   b) to estimate cytogenetically the lymphocyte dose from the determinated aberration rates ("biological dosimetry") for Chinese hamsters and ultimately for Thorotrast patients.

Publications

1. Radiobiology of Radium and the actinides in man. Proc. Int. Conf., Lake Geneva, Wisconsin, 11.-16.10.1981, Health Physics 44, Suppl. 1 (1983)
2. BUCKTON K., A. LANGLANDS, G. WOODCOCK "Cytogenetic changes following Thorotrast administration", Int. Radiat. Biol. 12 (1976) 565 - 577
3. TEICHERA-PINTO A., A.E. SILVA "Chromosome radiation induced aberrations in patients injected with Thorium dioxide", Environm. Research 18 (1979) 225 - 230
4. FISCHER P., E. GOLOB, E. KUNZE-MUEHL, T. MUELLNER "Chromosome aberrations in persons with Thorium dioxide burdens", Human Radiat. Cytogen. 7 (1967) 194 - 202
5. KEMMER W., H. MUTH, F. TRANEKJER, U. BORKENHAGEN "Chromosome aberrations caused by Thorotrast", In: Proc. 3rd Int. Meet. Toxicity of Thorotrast, Danish atomic energy commission Copenhagen, Risø Report 294 (1973) 104 - 111
6. STEINSTRÄSSER A. "Biophysical investigations of the dose effect relationship in chromosome aberrations of human lymphocytes caused by Thorotrast deposits. I. Physical aspects", Radiat. Environm. Biophys. 19 (1981) 1 - 15
7. STEINSTRÄSSER A., W. KEMMER "Biophysical investigations of the dose effect relationship in chromosome aberrations of human lymphocytes caused by Thorotrast deposits. II. Biophysical and medical aspects", Radiat. Environm. Biophys. 19 (1981) 17 - 28
8. MOORHEAD J.F., J.J. CONOLLY, W. McFARLAND "Factors affecting the reactivity of human lymphocytes in vitro", Immunology 99 (1967) 23 - 29
9. HEINZE B., H. MUTH, R. GRILLMAIER "Tierexperimentelle Untersuchungen über die Wirkung inkorporierter radioaktiver und nichtradioaktiver Partikel und deren Synergismus", Jahresbericht an den Bundesminister des Innern über die Arbeiten im Jahre 1981 (1982)
10. PRESTON R.J., J.G. BREWEN, K.P. JONES "Radiation induced chromosome aberrations in Chinese hamster leucocytes", Int. J. Radiat. Biol. 21 (1972) 397 - 400

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CHANGES OF THE LYMPHOCYTE POPULATIONS AND THEIR ROLE AS A BIOLOGICAL INDICATOR FOR IONIZING RADIATION

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1 Introduction

It is known for some time (see UNSCEAR Report 1972) that increased doses of ionizing radiation have a negative influence on natural body resistance. In recent years detailed work was done on the effect of ionizing radiation on the immune system, also in man. For these examinations mainly lymphocytes were used, since they frequently proliferate during their development to effector cells. For this reason, their sensitivity to radiation is very high. For many of the radiation-induced effects examined, dose-response curves could be developed.

Within the scope of a research project, supported by the Radiation Protection Programme of the Commission of the European Communities (Contract BIO-C-518-82-D), it is to be clarified to which extent effects of ionizing radiation on the lymphocytes can be used for a "biological dosimetry". It is the aim of the project to find an additional method for the approved and well established, but personnel- and time-consuming method of chromosome analysis. Initially, the relative and absolute number of lymphocytes and their subpopulations in the peripheral blood and the number of immunoglobulins produced by the lymphocytes are the main objective of the examinations.

2 Methods

For differentiating the lymphocytes, a new method developed by "Bio-Rad" is used and modified for our purposes. From heparinized whole blood the mononuclear cells (monocytes and lymphocytes) are isolated via density-gradient separation with Ficoll-Paque. Consecutively, the cells are incubated with so-called immunobeads. Due to rosette formation with the differently coloured beads, the subpopulations of the lymphocytes can be differentiated. The monocytes are identified by phagocytosis of the immunobeads.

At first, the blood of normal controls is examined in order to both optimize the method and obtain comparative values for the absolute and relative number of lymphocyte subpopulations in normal controls. Also the blood of normal controls and mononuclear cells, isolated from this blood, are irradiated in vitro with different doses. After this, the distribution of the lymphocyte subpopulations is evaluated, according to the method described above. The objective of this study is to yield dose-response curves for the change of the proportional and absolute values of the lymphocyte subpopulations.

Another important item is the assessment of the viability of the lymphocytes, the number of the immunoglobulin producing cells

and the amount of immunoglobulin production by the lymphocytes after prior irradiation.

For this purpose, either whole blood, isolated mononuclear cells or isolated lymphocytes are irradiated in vitro. Then the mononuclear cells or lymphocytes which were initially isolated or subsequently separated from whole blood are cultured. The cells are stimulated in culture with "pokeweed mitogen" to proliferate and mature to immunoglobulin producing cells. In some of the tests, cells are repeatedly removed during culturing time in order to determine their vitality by a dye exclusion test with trypan blue. In further experiments, at the end of the culturing time the number of immunoglobulin producing cells is determined by immunofluorescence and the amount of the produced immunoglobulins is found by the ELISA test and nephelometry. In place of the lymphocyte subpopulations or mononuclear cells in their physiological distribution, these can also be cultivated and examined in defined numerical proportions, as described above.

The objective of the work is also in this case to develop dose-response curves for the change of the given parameters, depending on the dose of ionizing radiation applied in each case.

3 Preliminary Results and Discussion

The above described method for differentiating mononuclear cells is a relatively simple process that requires little time. As compared to the currently used method, where B-cells are determined by immunofluorescence and T-cells in an individual process by a rosette test, the technique presented here has the advantage that monocytes and subpopulations of lymphocytes can be counted at the same time. In contrast to the established method for biological dosimetry, the chromosome analysis, our method is less personnel- and time-consuming and requires less experienced personnel. The disadvantage being that the absolute values and the proportional distribution of lymphocyte subpopulations may be quite different in physiology from individual to individual, so that the method might not be as sensitive and accurate as chromosome analysis in the lower dose range. Besides this, the values may vary when influenced by different pharmaceutical drugs.

An extension of the examinations to the blood of radiation therapy patients is intended to clarify to which extent the dose-response curves, after in vitro irradiation, are applying to in vivo conditions. Furthermore, it is to be examined whether the influence of other easily to be assessed parameters may increase the sensitivity and accuracy of the method.

In the method where cells are cultured in order to determine vitality and immunoglobulin production, a larger sensitivity is to be expected for lower dose values, since the radiation effect is increased due to cell cultivation. In this connection it has to be clarified whether a suitable combination of cells from the individual subpopulations or the use of other parameters might limit the biological variability and thus enables a greater accuracy of dose assessment. Here, too, the examinations shall be extended to the blood of radiation therapy patients to find out

whether the dose-response curves after in vitro irradiation can also be applied when irradiated in vivo. It is, however, of disadvantage that on account of the time required for cultivating cells, results are yielded not earlier than after several days, similarly to the analysis of structural chromosome aberrations. On the other hand, the time required as, e.g., for determining the immunoglobulin production at the end of cell culturing is relatively short and may be further reduced by automation.

ETUDE COMPARATIVE EXPERIMENTALE DES EFFETS DES NEUTRONS ET DES RAYONS GAMMA SUR LES CELLULES GERMINALES DURANT LA PERIODE PERINATALE

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INTRODUCTION

Le testicule du rat présente une importante variation de radiosensibilité au cours de son développement foetal et néo-natal. Pour une dose de 1,5 Gray (Gy) de rayons gamma du 60_{Co}, nous avons déterminé la période de plus grande sensibilité.

Elle s'étend du 18ème jour post-coïtum au 2ème jour post-partum (1). Au milieu de cette période radiosensible, à 21 jours de gestation, nous nous proposons d'étudier et de comparer la sensibilité des cellules germinales à différentes doses de neutrons et de rayons gamma. La destruction des cellules germinales est appréciée par la chute pondérale du testicule. En effet, les variations du poids du testicule représentent les différents degrés de destruction sélective des cellules germinales (2). Cette étude est complétée par l'observation histologique des tubes séminifères dont le dépeuplement reflète l'absence des clones cellulaires, issus des cellules germinales. Pour sensibiliser les mesures, les testicules sont étudiés à 26 jours après la naissance afin de laisser se développer des clones cellulaires, mais aussi, avant un repeuplement des tubes atteints par les cellules survivantes (3).

MATERIEL ET METHODES

Les femelles à 21 jours de gestation sont irradiées, soit par des neutrons de 14 MeV à des doses de 0,05 à 1,5 Gy à un débit moyen de 0,025 Gy/mn. , soit avec des rayons gamma du 60_{Co} à des doses de 0,1 à 1,5 Gy au débit de 0,1 Gy/mn. Un testicule est prélevé, pesé puis fixé pour l'histologie. Les tubes séminifères classés (300 à 1200), en coupes bien circulaires, sont classés en tubes normaux, en régénérescence et stériles d'après les critères définis par BEAUMONT (3).

RESULTATS - DISCUSSION

Le pourcentage du poids relatif du testicule des rats irradiés avec les neutrons ou les rayons gamma ne diffère pas pour les différentes doses d'irradiation, sauf à 0,5 Gy (fig.1). Le pourcentage du nombre des tubes séminifères normaux est semblable pour les deux types d'irradiation, sauf à 0,5 Gy où il est beaucoup plus faible après l'action des neutrons (fig.2). Cette particularité de la dose de 0,5 Gy de neutrons est la conséquence de l'augmentation des tubes séminifères en régénérescence et stériles par rapport aux mêmes tubes séminifères après action d'une même dose de rayons gamma. Cependant, cette différence ne se retrouve pas pour les autres doses utilisées.

L'étude histologique révèle les premiers tubes séminifères en régénérescence pour les doses de 0,05 et 0,1 Gy de neutrons alors qu'ils n'apparaissent qu'à partir de 0,2 Gy après rayonnement gamma.

Ensuite, les deux types d'irradiation sont semblables comme le prouve la dose de rayonnement laissant 50% de tubes normaux qui est de 0,38 Gy pour les neutrons et 0,43 Gy pour les rayons gamma.

Classiquement, l'E B R (Efficacité Biologique Relative) des neutrons par rapport aux rayons X ou gamma est voisin de 2 pour ce domaine de doses et d'autres critères. D'autre part, il est généralement admis que les effets des radiations sur les cellules portent plus spécialement sur le matériel génétique. Or, les cellules germinales foetales sont bloquées au stade G1 ou Go du cycle cellulaire pendant toute la période radiosensible (résultats non publiés). Avant et après cette période de blocage du cycle cellulaire, la radiosensibilité des cellules germinales diminue.

L'extrême sensibilité des cellules germinales foetales bloquées en G1 ou Go (0,05 à 0,2 Gy) et la similitude d'effet des neutrons et des rayons gamma permettent de penser que la chromatine de ces cellules est dans un état particulièrement radiosensible et/ou que les systèmes de réparation de l'ADN ne sont pas présents ou bien sont non fonctionnels.

Cependant, chez des mutants de levure, l'EBR est faible lorsque le système de réparation des coupures double brins de l'ADN est non fonctionnel et élevé lorsque celui-ci est fonctionnel (4). De plus, chez le s cellules CHO, le nombre des cassures de l'ADN après irradiation aux rayons X est le même à tous les stades du cycle cellulaire (5). Il semblerait donc que seule l'hypothèse d'un système de réparation de l'ADN, absent ou non fonctionnel, puisse expliquer la grande radiosensibilité des cellules germinales pendant la période périnatale. Dans ce cas, les deux types d'irradiation donneraient sensiblement les mêmes effets.

BIBLIOGRAPHIE

1 - COFFIGNY H. , PASQUIER C. , PERRAULT G. , DUPOUY J.P., in "Late biological effects of ionizing radiation" IAEA, Vienne, 1978, 207-220.

2 - ERICKSON B.H. and MARTIN P.G. Int. J. Radiat. biol. , 1972, 22, 517-524

3 - BEAUMONT H.M. Int. J. Radiat. Biol. 1960, 2, 247-256

4 - FRANKENBERG - SCHWAGER M. , FRANKENBERG D., HARBICH R. , " Proceedings of the 7th ICRR (J.J.BROERSE et coll., eds)," AMSTERDAM, July 3-8 1983, session B, B2-12.

5 - GRAUBMANN S. and DIKOMEY E. Int. J. Radiat. Biol. , 1983, 43, 475-483.

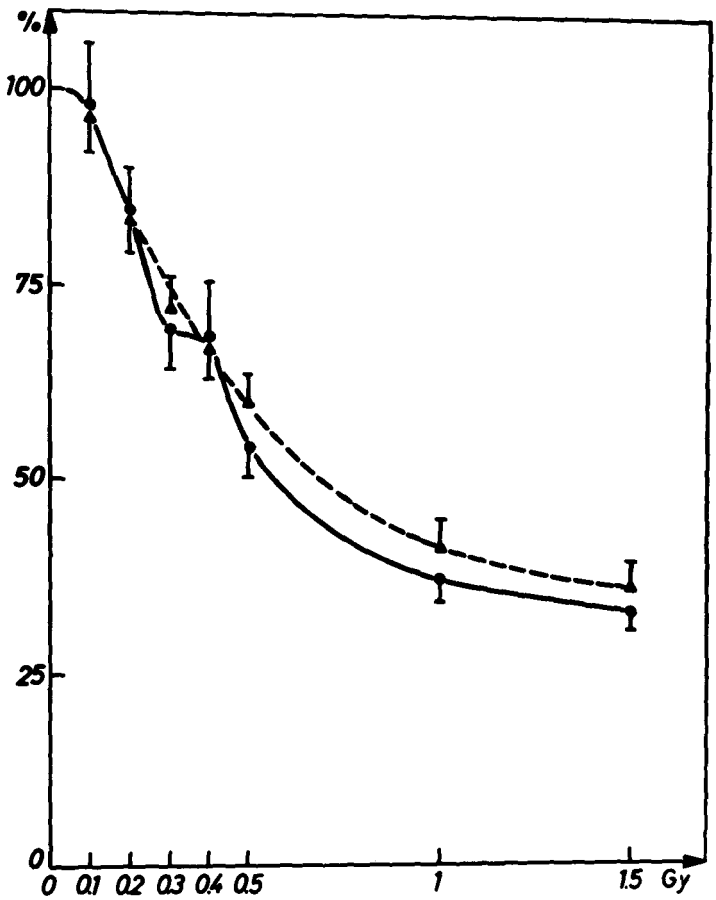

Figure 1 : Pourcentage du poids relatif du testicule des rats irradiés par les neutrons (•———•) ou les rayons gamma (x.....x) par rapport aux témoins.

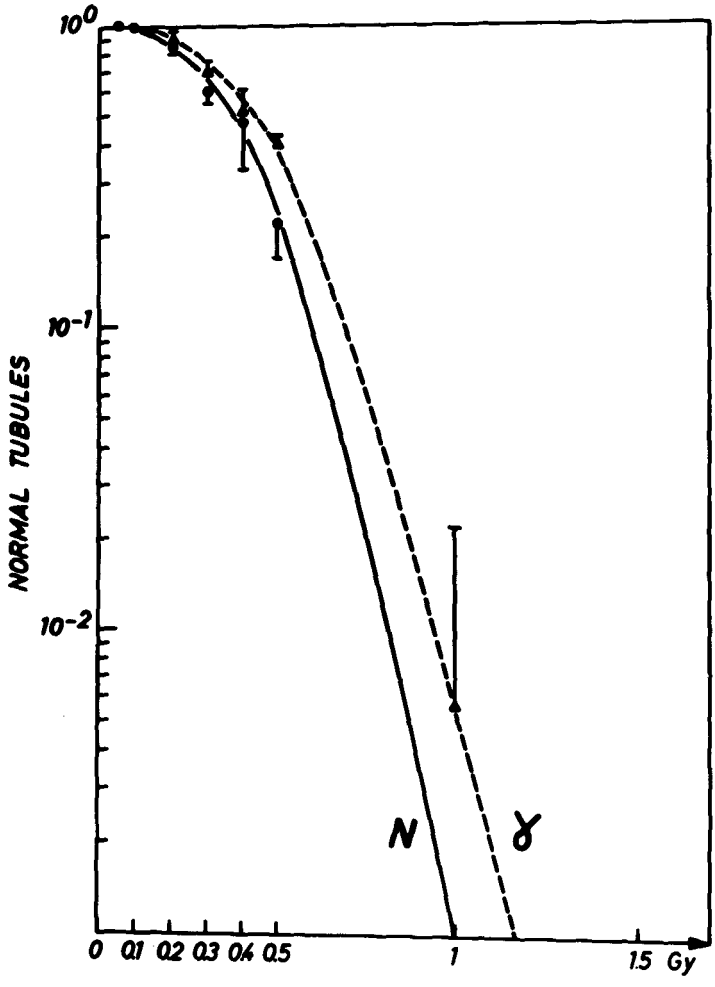

Figure 2 : Proportions des tubes séminifères normaux des animaux irradiés par les neutrons (•———•) ou les rayons gamma (x·····x) par rapport aux témoins.

MECHANISM OF ACTION FOR THE INDUCTION OF CHROMOSOME ABERRATIONS IN HUMAN LYMPHOCYTE CULTURES BY LOW-LET (ß-, X-, -) IRRADIATION

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The yield in chromatid breaks, dicentrics, and sum of all aberrations (except for gaps and fragments) induced in human peripheral lymphocyte cultures by different ß-sources (H-3, S-35, P-32) was investigated and compared with the respective effects of electromagnetic radiation (X- and γ-rays).

From the results for the coefficients α and ß of the linear-quadratic relationship a hypothesis was derived concerning the mechanism of action. The proposed assumption that the DNA is not the only radiation sensitive site within the cell, but that the nuclear membrane seems to play also an important role, is in accordance with our observation of a much stronger suppression of lymphocyte viability especially by acute ß-irradiation than by X-rays. To prove the hypothesis the absorption of ß-electrons in the nuclear membrane, i.e. the slowing-down of the continous ß-spectra, was calculated for H-3 and P-32 and related to the different behaviours of short-range H-3 ($ß \approx 0$) and long-range P-32 ($ß > 0$). In this way an interpretation for the lacking of the quadratic term for H-3- no intertrack interactions- can be given in terms of a strong disturbance of the low energy radiation fields.

For the dependence of the dicentric yield on dose rate a significant decrease of the linear coefficient α which is supposed to be independent on dose rate was found for protracted irradiation. This can be interpreted by different radiation sensivities in resting phases G_0 and cell cycle phases G_1-S-G_2 for acute and chronic irradiation, respectively. In any case, repair and recovery mechanisms are of secondary importance.

With the aid of a comprehensive statistical analysis the distributions of chromosome aberrations were compared with Poisson, Neyman-type A, and lognormal distributions, and the fit was examined with a - -test. The obtained results are in accordance with the above proposed hypothesis and support the specified mechanism of action.

CALIBRATION CURVE RELATING RADIATION DOSE FOR MIXED GAMMA AND THERMAL NEUTRONS TO ABERRATION YIELDS IN CYTOGENETIC DOSIMETY USING PERIPHERICAL BLOOD LYMPHOCYTES

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INTRODUCTION

During the last two decades, dosimetry by means of chromosomic analysis has become a current technique in the assessment of doses in abnormal situations (1). This practice serves as a supplement to physical dosimetry in accidental radiological evaluations.

The dose estimations made have been consistent with both clinical and physical evaluations (2). This is the biological technique with the lowest detection level in accidental overexposures, approximately 0.1 Gy with a low LET. Although it does not have the capacity of estimating doses within the boundaries of occupational doses, due to the accumulation of chromosomic aberrations that occurs as time goes by, it may be a useful technique for the detection of overexposures that were not identified previously or as a supplement of those obtained by means of physical dosimetry.

The frequency of chromosomic aberrations induced in linphocytes of the peripherical blood depends on the dose and on the LET. When dealing with low LET radiations, it also depends on the dose rate.

The development of the nuclear industry has increased the number of problems connected with radiological protection. Considering that physical dosimetry applied to high LET radiations is a complex technique, the application of cytogenetic dosimetry may become an important contribution.

In order to estimate doses in criticality accidents, our laboratory performs dosimeter calibrations by means of chromosomic analysis for mixed gamma radiation fields and for thermal neutrons.

MATERIALS AND METHODS - CULTURAL CONDITIONS

The conditions under which the culture of peripherical blood lynphocytes is performed have already been widely described. The differences in the techniques applied by the various laboratories have been presented as one of the main reasons for the variations observed in the relation between the doses and the frequency of induced chromosomic aberrations.

For this purpose, our laboratory has adopted the recommendations introduced by a programme coordinated by the IAEA (3), whose synthesis indicates that:

1. each laboratory must always attach itself to a strict standard protocole;

2. only those cells within the first mitotic division must be marked;

3. the fastenings must be performed when the curve of the first mitosis reaches its peak;

4. a culture containing 5-bromodeoxiuridine must be processed along with the rest, in order to assess the frequency of cells in the first, second and other divisions, which are eventually used for marking, if necessary;

5. by applying the former criteria, each laboratory must perform its own calibration with blood samples obtained from at least three different donators.

Each point of the calibration curve is estimated as from the analysis of 500 metaphases. Both the fraction of dicentrics and the rings per cell are assessed for each dose.

IRRADIATION CONDITIONS

Irradiations were performed at different positions of the thermal column in a production and research reactor, at doses between 0 and 4 Gy.

The blood was exposed in 10 ml hermetic polyethilene tubes, with three pairs of thermoluminiscent dosimeters (TLD 600 and 700) attached to their external walls.

At the same time, a calibration curve was obtained with gamma radiations from a Co-60 source, by applying the same cultural technique.

RESULTS

It is assumed that the gamma and neutronic components of the mixed field produce additive effects (4).

Microscopical observation does not allow to see the chromosomic aberrations produced by either component.

In order to attain this goal, knowledge must be gained on the gamma/thermal neutrons ratio, so as to assess the separate contribution of each type of radiation in the fraction of chromosomic aberrations observed.

For each dose in the mixed field, the fractions of chromosomic aberration induced by cells must be assessed and the corresponding function is adjusted as per such fractions.

The ratio between the fraction of cell-induced chromosomic aberrations and the gamma dose is that used, as from a previous calibration, in the estimation of doses in accidental overexposures with low LET radiations. Finally, the ratio between the fraction of cell-induced chromosomic aberrations and the thermal neutron doses is assessed for each dose as from the difference between the calibration curves adjusted with mixed field and with gamma radiation.

The calibration curves obtained for each type of radiation are the following:

I. Gamma radiation

Dicentrics plus rings

$$Y = 0.001 + 6.5\ 10^{-6}\ \frac{\text{dicentrics + rings}}{\text{cell} \cdot \text{rad}^2}\ D^2$$

II. Thermal neutrons

Dicentrics plus rings

$$Y = 0.001 + 2.41\ 10^{-3}\ \frac{\text{dicentrics + rings}}{\text{cell} \cdot \text{rad}}\ D$$

CONCLUSIONS

Biological dosimetry performed by means of chromosomic analysis for mixed gamma radiation and thermal neutron fields is a supplementary technique in physical dosimetry. It is needed for estimating doses in criticality accidents.

The gamma/neutron ratio assessed by means of physical dosimetry is needed in every case for estimating the contribution of each component.

BIBLIOGRAPHY

1. Lloyd, D.C.; Purrot, R.J. Chromosome aberration analysis in radiological protection dosimetry. Radiation Protection Dosimetry 1.1., 19-28 (1981).

2. Bender, M.A.; Gooch, P.C. Somatic chromosome aberration induced by human whole-body irradiation: the Recuplex criticality accident. Radiation Research 29, 568-582 (1966).

3. Evaluation of radiation induces chromosomal aberration in human peripheral blood lymphocytes (in vitro). Results of an IAEA Coordinated Programme (1980).

4. Purrot, R.J.; Lloyd, D.C.; Dolphin, G.W. Chromosome aberration dosimetry following a controlled criticality pulse to human peripheral blood lymphocites. 46-50 in AERE-R8521, Nuclear Accident Dosimetry PT. II. AERE, Harwell (1976).

THE GASTROINTESTINAL ABSORPTION OF TRANSURANIUM ELEMENTS IN ANIMALS AND THE IMPLICATIONS FOR MAN

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The most important transuranium radionuclides in terms of environmental pollution and potential hazard to man are the long-lived, alpha-emitting isotopes of neptunium, plutonium and americium. Neptunium-237 has been identified as a major contributor to the dose people could receive due to the eventual leaching of material from radioactive waste disposal sites into water supplies (1). The ingestion of ^{239}Pu and ^{241}Am is currently of concern in connection with the consumption of seafood contaminated by marine discharges from the nuclear fuel reprocessing plant at Sellafield in the north of England (2). No measurements have been made of the gastrointestinal absorption of neptunium, plutonium and americium in man. Estimates of absorption are based on results from animal experiments. The National Radiological Protection Board (NRPB) is recommending values of absorption for use in calculations of doses to members of the public ingesting radioisotopes of these elements at low concentrations in food and water(3). These recommendations consider adults and newborn children as separate groups on the basis of evidence from animal experiments of increased absorption in the immediate postnatal period. In this paper, measurements of the absorption of neptunium, plutonium and americium in adult and newborn animals are presented and the NRPB values of absorption are discussed.

METHODS

Nitrate solutions of ^{237}Np, ^{239}Np, ^{236}Pu, ^{238}Pu, ^{239}Pu and ^{241}Am for administration to animals were prepared by evaporating aliquots of the stock solutions to dryness and redissolving the residues in 0.1M HNO_3. Citrate complexes (pH 6.5) were prepared by mixing a solution of the radionuclide in 0.01M HNO_3 with an equal volume of 70mM citrate. Phytate complexes (pH 6-7) were similarly prepared using 70mM phytate with ^{239}Np and ^{241}Am and 5.6 mM phytate with ^{238}Pu. Bicarbonate solutions (pH 6.5-8) were prepared by adding 1M $NaHCO_3$ to either ^{239}Np or ^{238}Pu in 0.1M HNO_3.

Adult rats and rabbits were anaesthetized during the administration of radionuclide solutions by gastric intubation. The volumes of solution administered were 100-200 µl for rats and mice and 400-500 µl for rabbits. Adult and newborn hamsters were given oral doses of 10-100 µl of solution on to the back of the tongue. Rats were also fed with liver containing ^{239}Np, ^{238}Pu and ^{241}Am as explained below. Animals were kept for 1-3 weeks after actinide administration. The activities of radionuclides in tissue samples were determined by standard methods. Total absorption of radionuclides from the gastrointestinal tract was calculated as described previously (4).

RESULTS

The absorption of ^{239}Np, ^{238}Np and ^{241}Am was measured in adult rats after administration as the bicarbonate, citrate and phytate complexes and after incorporation into rat liver. Citrate and phytate are actinide binding ligands that are common in a wide variety of foodstuffs and bicarbonate is a likely chemical form of the elements in water supplies (5,6). The values of absorption obtained for these chemical forms of the elements are compared in Table 1 with values for the inorganic nitrates. For ^{239}Np, absorption varied from 0.01% after incorporation into liver to about 0.15% after administration as either the citrate

complex or bicarbonate. Phytate increased the absorption of ^{238}Pu to 0.1% compared with the value of 0.02% for the nitrate, as discussed previously by Cooper (5). Citrate caused a smaller increase in ^{238}Pu absorption. For ^{241}Am, no increase in absorption was observed compared with the value of 0.08% for the nitrate; administration as the phytate complex and incorporation into liver reduced absorption by about a factor of two.

A comparison of the absorption by rats of ^{239}Np (5×10^{-7} mg) and ^{237}Np (0.5 mg) administered as the nitrates (Table 1) confirmed observations by Sullivan (7) that absorption is increased at high concentrations. For plutonium, absorption was not affected by the mass administered over the range from 2×10^{-8} mg of ^{236}Pu to 3×10^{-2} mg of ^{239}Pu (Table 1).

Table 1. The gastrointestinal absorption of neptunium, plutonium and americium in adult rats.

Isotope	Chemical form	Dose (mg)	No. of animals	% Absorption ($\bar{X} \pm$ S.E.)
^{239}Np	Nitrate	5 E-7	5	0.03 ± 0.01
	Citrate	5 E-7	5	0.14 ± 0.02
	Phytate	5 E-7	5	0.04 ± 0.01
	Bicarbonate	5 E-7	5	0.15 ± 0.02
	in liver*	2 E-7	6	0.01 ± 0.002
^{237}Np	Nitrate	5 E-1	6	0.26 ± 0.04
^{236}Pu	Nitrate	2 E-8	5	0.01 ± 0.002
^{238}Pu	Nitrate	1 E-5	6	0.02 ± 0.004
	Citrate	3 E-6	6	0.06 ± 0.01
	Phytate	2 E-5	6	0.10 ± 0.02
	Bicarbonate	1 E-5	6	0.04 ± 0.01
	in liver*		6	0.03 ± 0.01
^{239}Pu	Nitrate	3 E-2	6	0.02 ± 0.01
^{241}Am	Nitrate	2 E-4	6	0.08 ± 0.01
	Citrate	2 E-4	6	0.06 ± 0.01
	Phytate	2 E-4	6	0.04 ± 0.01
	in liver*	3 E-5	6	0.03 ± 0.004

*Liver from rats given intravenous injections of the citrate complexes, 2 days previously.

Table 2. Species differences in the gastrointestinal absorption of neptunium, plutonium and americium.

Isotope/Chemical form	% Absorption*		
	Rat	Rabbit	Hamster
^{239}Np bicarbonate	0.15 ± 0.02(5)	0.12 ± 0.02 (4)	0.02 ± 0.002 (6)
^{238}Pu citrate	0.07 ± 0.02(6)	0.02 ± 0.003(4)	0.006 ± 0.001 (6)
^{241}Am citrate	0.08 ± 0.01(6)	0.02 ± 0.004(4)	0.01 ± 0.002(13)

*$\bar{X} \pm$ S.E., numbers of animals in parentheses. Doses of ^{239}Np, ^{238}Pu and ^{241}Am were 5×10^{-7} to 1×10^{-6} mg, 3×10^{-6} to 1×10^{-4} mg and 2×10^{-4} to 8×10^{-4} mg, respectively.

Table 3. The gastrointestinal absorption of neptunium, plutonium and americium in neonatal hamsters.

Isotope/Chemical form	Dose (mg)	No. of animals	Age (days)	% Absorption ($\bar{X}$ ± S.E.)
^{239}Np nitrate	4 E-8	6	2	2.5 ± 0.3
	4 E-8	6	4	1.7 ± 0.3
^{239}Np bicarbonate	5 E-8	10	2	5.5 ± 1.7
	5 E-8	10	4	2.1 ± 0.4
^{239}Pu citrate	3 E-4	5	1	3.5 ± 0.5
	3 E-4	6	4	1.4 ± 0.5
	7 E-4	9	7	0.02 ± 0.004
	3 E-3	6	22	0.007 ± 0.001
	7 E-3	6	30	0.003 ± 0.001
^{241}Am nitrate	2 E-5	5	1	4.5 ± 0.5
	8 E-5	5	4	1.7 ± 0.3
	2 E-4	5	7	0.5 ± 0.04
	3 E-4	6	22	0.006 ± 0.002
	1 E-4	6	30	0.02 ± 0.002

Results obtained for the absorption of ^{239}Np, ^{238}Pu and ^{241}Am in the rat are compared in Table 2 to corresponding values for rabbits and hamsters, showing order of magnitude differences between animal species. The results obtained in the hamster for the absorption of plutonium and americium as the citrate complexes were respectively higher and lower than corresponding values of 0.002% and 0.05% for the nitrates (8).

The absorption of neptunium, plutonium and americium in newborn hamsters is shown in Table 3. The values of 2-6% absorption in 1-2 day old animals represent an increase from adult values of 0.01-0.05% by a factor of about 100. The results for ^{239}Pu and ^{241}Am show a reduction in absorption during suckling to low values after 22 days at the time of weaning.

DISCUSSION

The values used by the International Commission on Radiological Protection (ICRP) in the calculation of Annual Limits on Intakes of radioisotopes of neptunium, plutonium and americium by workers are 1%, 0.01% and 0.05% respectively (9). The corresponding values that the NRPBrecommend for use in the calculation of doses to adult members of the public are 0.1% for neptunium and 0.05% for plutonium and americium (3). The ICRP values apply specifically to occupational exposures to inorganic forms of the elements while the NRPB recommendations apply to the ingestion of low concentrations of the elements in food and water.

The ICRP value of 1% absorption for neptunium was based on measurements on rats using high concentrations of ^{237}Np nitrate (9). Recent experiments with rats and baboons showed that the absorption of neptunium administered as the nitrate was only as high as 1% when milligrame quantities of ^{237}Np were administered and lower values of about 0.1% were obtained using ^{239}Np nitrate (7,10). As shown in Tables 1 and 2, values of 0.01-0.15% absorption were obtained for low concentrations of neptunium in chemical forms which are thought likely to be

encountered in dietary intakes of the element. The NRPB value of 0.1% was recommended on the basis of a review of all available data (11).

The increase for plutonium to 0.05% from the ICRP value of 0.01% takes account of the evidence that absorption may be greater after ingestion of organic complexes of the element in foods. Both phytate and citrate increased absorption in the rat compared with the value for the inorganic nitrate (Table 1) and citrate also increased absorption in the hamster. Results obtained by Sullivan (12) for the absorption by rats and guinea pigs of plutonium incorporated into alfalfa also suggest that incorporation into plant tissue may increase absorption. The recommended value of 0.05% absorption was based on relevant data from all animal species studied (8).

For americium it appears that absorption may be decreased by association with organic molecules in food materials but the few available data do not warrant a change from the ICRP value of 0.05% (8).

Each of these estimates of the absorption of neptunium, plutonium and americium in man is subject to the uncertainties involved in extrapolating from the animal data. The range in results for different chemical forms of the elements (Table 1) and species differences in absorption (Table 2) both lead to uncertainties in the choice of single values for absorption in man.

The extrapolation to absorption in humans is particularly tenuous in the case of the results on increased absorption in the noenate. Nevertheless the high values obtained in animals warrant the consideration of newborn children as a separate group for the purposes of radiological protection. The values of absorption that the NRPB recommend for application to newborn children in the first year of life were based on a review of the available data on hamsters (Table 1), together with results in rats, guinea pigs and pigs (8). The assumptions were made that absorption will be high while milk feeding is maintained and will decrease to adult values over the weaning period. Thus although absorption is likely to be declining rapidly in the immediate post-natal period, an averaged value of 1% was suggested for the absorption of neptunium, plutonium and americium during the first three months of life and an averaged value of 0.5% for the whole of the first year. Adult values of 0.1% for neptunium and 0.05% for plutonium and americium were taken to apply from 9 months of age.

REFERENCES

(1) Hill, M. D. (1979) NRPB-R86, HMSO, London.
(2) Hunt, G. J. (1982) MAFF, Aquatic Monitoring Report No. 8, Directorate of Fisheries Research, Lowestoft.
(3) NRPB-GS3 (1983) HMSO, London.
(4) Harrison, J. D. and Stather, J. W. (1981) Radiat. Res. 88, 47.
(5) Cooper, J. R. and Harrison, J. D. (1982) Health Phys., 43, 912.
(6) Larsen, R. P., Oldham, R. D., Bhattacharyya, M. H., Moretti, E. S. and Austin, D. J. (1981) Radiat. Res. 87, 37.
(7) Sullivan, M. F., Miller, B. M. and Ryan, J. L. (1983) Radiat. Res. 94, 199.
(8) Harrison, J. D. (1983) Radiat. Prot. Dosim. 5, 19.
(9) ICRP 30: Pt. I (1979) Ann. ICRP 2, 3/4; Pt II (1980) Ann. ICRP 4, 3/4.
(10) Metivier, H., Masse, R. and Lafuma, J. (1983) Radioprot. 18, 13.
(11) Harrison, J. D. (1983) NRPB-M94, HMSO, London.
(12) Sullivan, M. F., Garland, T. R., Cataldo, D. A., Wildung, R. E. and Drucker, H. (1981) Health Phys. 38, 215.

THE RETENTION OF PLUTONIUM AND AMERICIUM IN LIVER: AN INTERSPECIES COMPARISON

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All the evidence from both animal and human studies indicates that the liver is a major site of deposition of plutonium and americium in all the species examined. However, the animal studies show that there is a wide species variation in the retention of both elements in the liver. The magnitude of the liver uptake, which probably depends critically, especially for plutonium, on the chemical form in which the actinide enters the systemic circulation, ranges from about 4 to about 60% of the resorbed material in all species including man (I-II).

A review of the animal data for the retention of plutonium and americium in liver suggests that the various species examined fall into one or other of two main classes. First species showing a relatively rapid, usually at least biphosic pattern of loss from the liver, this group includes mice (1), rats (2), tupaia (tree shrew) (3), macaque monkey (4) and baboons (5). In this group 80 to 100% of the liver burden is lost whith half-times ranging from about 4 to 200 days. In the second group which includes dogs (6), hamsters (7) deermice, grass-hopper (8) mice and probably pigs (9) the loss of plutonium or americium from liver appears to follow a monoexponential clearance with half-times measured in years. The equations for the retention of plutonium in the liver of eight animal species are listed in Table I.

The rat data suggest that, at least in this species, the first, rapid component of liver clearance may not always be present. The bi-exponential clearance was observed after intravenous injection of plutonium citrate (2) and appears to be typical of the clearance of this compound. The monoexponential was seen following intravenous injection of plutonium nitrate into over 100 rats (10); a similar clearance pattern was found after intramuscular injection of plutonium chloride (11). These observations suggest that the plutonium species which enters the blood stream may be an important determinant of liver uptake and retention. The reasons for these species differences in liver retention cannot yet be explained in physiological or biochemical terms. However, it is suggested that the retention may be controlled mainly by the ability of the particular species to excrete heavy metals into the bile.

The critical question from a radiological protection viewpoint is the retention time of plutonium and americium in human liver. The data from the human subjects injected with plutonium citrate (12) do not provide any clear indication of the retention time.

McInroy et al. (13) published data on the content of fall-out plutonium in human liver, which had been derived from analysis of autopsy material collected from different geographical areas of the continental United States during the period 1959 to 1976. Since the level of fall-out plutonium in air and diet has been at a relatively low, and decreasing, level since the late 1960's it appeared possible that an analysis of these data for each year from 1968 to 1976

might permit some estimate of the retention time of plutonium in human liver. The data for the plutonium content of the whole liver were grouped according to the year of death and a grand mean and standard error was calculated for each year group. The data were then examined and any values deviating by more than 2 standard deviations from the grand mean were excluded as probably being unrepresentative of their year group. A "revised" mean and standard deviation was then calculated for each year. As a result of this analysis 27 data points out of 597 were excluded. The revised mean values were then plotted against time on a log-linear plot and the line of best fit was calculated by the method of least squares. The data shown in Fig. 1 approximate to the regression function:

$$L\ (pg) = 36 \exp - 0.0313 \pm 0.0089\ t$$

where "t" is given in years (1960 = 0). The correlation coefficient, R = 0.801 at 7 degrees of freedom, is significant at the 1% level. The half-time of retention is 22.1 years with 95% confidence limits of 17.2 and 30.9 years.

This analysis of the annual *mean* values for fall-out plutonium in human liver suggests that, measured against a background of a continous low-level plutonium intake, there is a slow elimination of the element with a half-time of about 20 years. The uncertainty on the measurement is rather greater than the 95% confidence limits suggest since for the year groups the coefficient of variation about the mean lies between 49% and 83%.

The 20 year half-life for plutonium in liver is half that currently assumed by ICRP (14). However, it is in broad agreement with faecal excretion data (12) and also with the observations of americium in human liver (15).

References

1. Andreozzi,U., Clemente,G.F., Ingrao,G. and Santori,G., Health Phys. 44, Suppl. 1, 505-511, 1983
2. Hollins,J.G. and Storr,M.C., Radiat. Res. 61, 468-477, 1975
3. Seidel,A., Darai,G., Flügel,R., Hofmann,W. and Sontag,W., Health Phys. 43, 239-246, 1982
4. Durbin,P.W., Jeung,N. and Schmidt,C.T., Proceedings Seventh. International Congress Radiation Research, Amsterdam, July 3-8, 1983, Martin Nijhoff, Amsterdam E5-03
5. Metivier,H., Masse,R., Nenot,J.C., Nolibe,D. and Lafuma,J. in Diagnosis and Treatment of Incorporated Radionuclides IAEA, Vienna 1976, 107-117
6. Stover,B.J., Atherton,D.R. and Buster,D.S., Health Phys. 20, 369-374, 1971
7. Sütterlin,U., Doctoral Dissertation, University Karlsruhe, 1982
8. Taylor,G.N., Jones,C.W., Gardner,P.A., Lloyd,R.D., Mays,C.W. and Charrier,K.E., Radiat. Res. 86, 115-122, 1981
9. Sullivan,M.F. and Gorham,L.S., Health Phys. 44, Suppl. 1, 411-417, 1983
10. Taylor,D.M. - Unpublished
11. Scott,K.G., Axelrod,D.J., Fisher,H., Crowley,J.F. and Hamilton, J.G., J. Biol. Chem. 176, 283-293, 1948

12. Langham,W.H., Bassett,S.H., Harris,P.S. and Carter,R.E., Health Phys. 38, 1031-1060, 1980
13. McInroy,J.F., Campbel,E.E., Moss,W.D., Tietjen,G.L., Eutsler, B.C. and Boyd,H.A., Health Phys. 37, 1-136, 1979
14. International Commission on Radiological Protection, Publication 30, 1980, Publication 19, 1972
15. Wrenn,M.E., Rosen,J.C. and Cohen,N. in "Assessment of Radioactive Organ and Body Burden", IAEA, Vienna 1972, 595-621

Table 1 The Retention of Plutonium in the Liver of Various Animal Species

Species	Retention Equation	Reference
Mouse	L1 = 34exp -0.693t/17 *	(1)
	L2 = 7exp -0.693t/75	
	L3 = 3exp -0.693t/354	
Rat	L1 = 12exp -0.693t/6	(2)
	L2 = 3exp -0.693t/115	
	L1 = 4exp -0.693t/128	(10)
Tupaia	L1 = 17exp -0.693t/164	(3)
Macaque	L1 = 15exp -0.693t/25	(4)
	L2 = 46exp -0.693t/170	
Dog (Beagle)	L1 = 32exp -0.693t/4100	(6)
Hamster (Chinese)	L1 = 38exp -0.693t/ ∞	(7)
Deer Mouse	L1 = 34exp -0.693t/ ∞	(8)
Grasshopper Mouse	L1 = 21exp -0.693t/ ∞	(8)

* Days

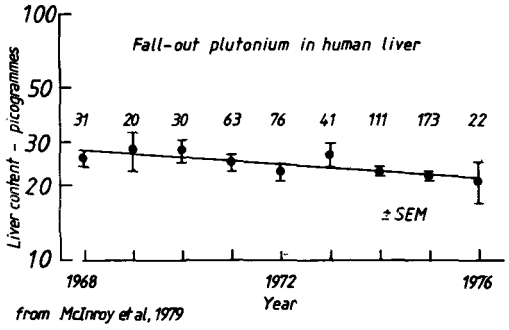

Figure 1: The retention of "fall-out" plutonium in human liver calculated from the autopsy data of McInroy et al. (13). The numbers above the error bars are the numbers of observations.

ABSORPTION GASTROINTESTINALE DU NEPTUNIUM CHEZ LE SINGE, INFLUENCE DU REGIME ALIMENTAIRE

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Une attention croissante a été récemment portée au neptunium 237, émetteur α à vie longue, en relation avec sa contribution majeure dans la gestion à long terme des déchets radioactifs (1).

Dans les études d'impact radiologique associé au rejet de ces déchets les doses potentielles délivrées par le neptunium ont été calculées en assumant que 1% de cet élément est absorbé au niveau gastrointestinal. Cette valeur du coefficient de transfert (f_1) est retenue depuis 1980 par la CIPR dans sa publication 30 (2). Toutefois, les données prises en compte lors de l'établissement de ce coefficient résultaient d'expériences utilisant de grandes quantités pondérales de neptunium (3). Depuis, en utilisant de plus faibles masses ingérées il a été montré que la quantité transferrée est environ 10 fois plus faible (4-5).

L'influence du régime alimentaire sur la valeur de f_1 ayant été démontrée pour le plutonium nous avons voulu vérifier s'il en est de même pour le neptunium chez un primate proche de l'homme, le babouin.

MATERIELS ET METHODES

Les singes utilisés, mâles et femelles, sont des babouins, papio papio, pesant entre 6 et 12 kg. Les solutions de neptunium 237 ou 239 sont purifiés par une technique mettant en oeuvre les étapes suivantes : réduction du Np au degré d'oxydation IV en milieu HNO_3, 3M, extraction sélective du Np IV par une solution de trilaurylamine à 15% dans le dodécane, réextraction nitrique par addition d'acide α bromocaprique à la solution de nitrate de trilaurylamine (6), oxydation du Np IV en Np V par addition de $Na\,NO_2$ dans l'éluat. Les masses utilisées sont obtenues soit à partir de l'isotope 239 seul, soit à partir des mélanges 239-237. L'isotope 239 servant alors de marqueur isotopique. La technique d'ingestion et de détermination de la charge en Np a été décrite par ailleurs (5).

Le régime alimentaire normal (N) d'un singe comprend chaque jour 120 g d'aliments deshydratés (UAR, France) plus 1 orange, 1 pomme et 1 banane. Le régime A, comprend uniquement des aliments deshydratés appauvris en vitamines et acide alcools. Le régime B, dérivé du régime normal, est composé d'aliments deshydratés composés pour 50% de poudre de lait. Le régime C, ne comprend que des fruits (2 oranges, 2 bananes et 2 pommes), le régime D enfin, consiste en l'ingestion de pommes de terre riches en phytates (7) cuites durant 15' à l'autoclave. L'eau est dans tous les cas à volonté. Tous ces régimes sont donnés 8 jours avant l'ingestion et jusqu'au sacrifice.

RESULTATS ET DISCUSSION

La quantité absorbée après ingestion gastrique de neptunium (f_1) est calculée à partir de la somme des charges des organes prélevés ajoutée à la valeur de l'excrétion urinaire. Les seules charges significatives prises en compte au niveau des calculs des coefficients de transfert sont les charges du squelette, du foie et des reins. Nous avons porté dans le tableau les valeurs du coefficient f_1 (colonne 4) et celles de la charge organique (colonne 5).

Les résultats montrent qu'à faible masse ingérée (de l'ordre du ng/kg) le coefficient de transfert est chez le singe de 1.10^{-3}, alors qu'il est supérieur

Régime	Poids	Masse ingérée (µg/kg)	Pourcent transféré (f_1)	Charge organique (en %)
(1)	(2)	(3)	(4)	(5)
N	6,5	769	0,783	0,335
N	11,6	431	1,619	0,730
			M : 1,201	M : 0,533
N	7,85	0,00140	0,0948	0,0362
N	8,6	0,00128	0,107	0,0377
			M : 0,101	M : 0,0370
A	5,2	0,00262	0,0126	0,00877
A	6,2	0,00220	0,0842	0,0192
			M : 0,0484	M : 0,0140
B	5,4	0,000774	0,112	0,00646
B	5,15	0,000796	0,0119	0,00480
			M : 0,0620	M : 0,00563
C	3,5	0,00509	0,969	0,0565
C	4,1	0,00435	0,441	0,0143
			M : 0,705	M : 0,0354
D	6,1	0,000877	0,350	0,250
D	5,9	0,000907	0,296	0,160
			M : 0,323	M : 0,205

Tableau 1 : Influence de la masse et du régime alimentaire sur le coefficient de transfert f_1 et la charge organique après ingestion de neptunium par le singe.

à 1.10^{-2} lorsque la masse ingérée est proche de 0,5 mg. Ces valeurs tenant compte de l'excrétion urinaire ne peuvent être à priori que surestimées, si l'on redoute une contamination par les fécès. Toutefois nous pensons que celle-ci doit être faible puisque nous avons montré par ailleurs (5) que les rapports os/urine obtenus chez le rat et ches le singe sont après ingestion, d'une part identiques, et d'autre part légèrement supérieurs à ceux obtenus chez le rat après injection intramusculaire de neptunium, où une telle suspicion de contamination ne peut être avancée (8). Quelle que soit la masse ingérée, environ 60% de la quantité transférée sont excrétés dans les urines, 40% restant au niveau des organes et plus particulièrement du squelette.

Le régime A, appauvri en acide alcool abaisse de façon peu significative la valeur de f_1 (col 4). A l'inverse le régime C très riche en acide-alcools complexants augmente de près d'un facteur 7 la valeur de f_1. Toutefois cette augmentation ne se solde pas par une charge organique accrue, la forme complexée migrante se trouvant éliminée par la voie urinaire (col 5). Le régime lacté, B, abaisse le transfert intestinal et la charge organique de près d'un facteur 7. Le régime D, composé uniquement de pommes de terre cuites, favorise le passage intestinal du neptunium (col 4) sous une forme favorisant le dépôt organique de près d'un facteur 5 (col 5) ; 2.10^{-3} de la masse ingérée.

En conclusion, l'ingestion de neptunium chez un primate proche de l'homme conduit également à un facteur de transfert au niveau intestinal qui varie avec la masse. Les valeurs obtenues avec les plus faibles masses de neptunium montrent qu'un coefficient de transfert digestif de 1.10^{-3} semble plus approprié aux risques envisagés lors de la gestion des déchets.

L'influence du régime alimentaire, que nous avons voulu excessive dans cette expérimentation ne semble apporter que quelques modifications. L'ingestion unique de fruits augmente le transfert sans toutefois augmenter le dépôt qui contribue principalement à la dose délivrée si l'on retient l'os, comme organe critique. A l'inverse l'ingestion de pommes de terre augmente réellement le dépôt, à un niveau qui reste toutefois proche de 1.10^{-3}, mais un tel régime qui n'a qu'un intérêt expérimental est inenvisageable durablement chez l'homme. Les autres régimes enfin, abaissent la valeur de f_1.

REFERENCES

1. R.C. THOMPSON, Radiat. Res., 1982, 90, 1-32
2. CIPR, ICRP Publication 30, part 2, Oxford, Pergamon Press, 1980.
3. M.F. SULLIVAN, Health Phys., 1980, 38, 159-171.
4. M.F. SULLIVAN, B.M. MILLER, J.L. RYAN, Radiat. Res., 1983, 94, 199-209.
5. H. METIVIER, R. MASSE, J. LAFUMA, Radioprotection, 1983, 18, 13-17.
6. C. MADIC, G. KOEHLY, Nucl. Technol., 1978, 41, 323-340.
7. J.R. COOPER, J.D. HARRISON, Health Phys., 1982, 43, 912.
8. M. MORIN, J.C. NENOT, J. LAFUMA, Health Phys., 1973, 24, 311-315.

Le texte complet sera soumis à la fin d'une expérimentation plus complète à la Revue Radiation Research pour publication.

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COMPARISON OF THE SUBCELLULAR BEHAVIOUR OF TRANSURANIUM ELEMENTS IN RAT AND CHINESE HAMSTER LIVER

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One of the most striking aspects of the comparative behaviour of transuranium elements is the distinct species variation in their retention times in mammalian liver (1,2). The aim of our studies is to explain these species variations in order to provide a better basis for the understanding of the comparative behaviour of transuranium elements in animals and man. Of primary importance is a knowledge of the subcellular component to which the nuclides are initially bound and the question whether it is the same for all animal species. For this purpose, we have started with rat and Chinese hamster as examples of species showing rapid and slow elimination of transuranium elements from liver, respectively.

The animals were injected with Pu-239 citrate, some (Fig.1)being treated with 750mg/kg Triton WR 1339 four days prior to sacrifice (3). A mitochondrial-lysosomal fraction (ML) was prepared from liver (3) (4) (5) and further analyzed by sucrose or metrizamide gradients or by free flow electrophoresis (6). Electrophoresis was performed with a VaP 11 (Bender+Hobein,Munich), field strength 140V/cm, current 180mA, buffer (.25M sucrose, .01M triethanolamine, .01M acetic acid, ph 7.4) flow rate 4ml/h for each of the 90 samples, sample flow rate 4ml/h, temperature 6C. All other details are described elsewhere (7) together with the conditions for the assay of the marker enzymes (for lysosomes: acid phosphatase, AP; arylsulfatase,AS; N-acetyl-β-glucuronidase, N-Ac and cathepsin D, Cath; for mitochondria:glutamate dehydrogenase,GDH; for endoplasmic reticulum: glucose-6-phosphatase, G-6-P; for pericellular membranes:alkaline phosphodiesterase,AlPD). Data from density gradients are presented as frequency histograms, where ΔQ = fractional amount of constituent in fractions and $\Delta\rho$ = density increment from fraction to fraction.

Results and Discussion

In Fig.1 and 2 we present the profiles for ^{239}Pu and for the lysosomal enzymes in sucrose and metrizamide gradients for days 4 or 10 after injection, when the nuclide content of the cytosol is less ∿10% of total liver burden in both species (7). In all cases a close correlation of the radioactivity and the lysosomal enzyme profiles exists. The profiles for the other marker enzymes are presented in a previous report (7), which showed, that the Pu-239-profiles from Triton WR 1339 treated animals in sucrose gradients and of untreated controls in metrizamide are clearly separated from mitochondria and endoplasmic reticulum, but the relationship to pericellular membranes remains equivocal for Chinese hamsters. Using Percoll gradients (7), and especially free flow electrophoresis (Fig.3), we could demonstrate that only a very small fraction of Pu-239 is bound to pericellular membranes in Chinese hamster liver at early times after injection.

The results obtained so far enabled us to identify secondary lysosomes as primary target organelles for Pu-239 (and Am-241 (7)) in

rat as well as in Chinese hamster liver. This implies that the species differences in nuclide elimination cannot be due to binding to different cell organelles with different biological fates.

There is no evidence for redistribution of transuranium elements in rat liver with time (7) and it seems probable that they are eliminated into the bile by a normal, relatively rapid excretion of lysosomal material (8,9). In Chinese hamster liver, Pu-239 remains bound to organelles with a diameter ≥400nm (5).The particles to which Pu-239 is bound after ≥70 days consist electrophoretically of two groups (Fig.4),they are less dense than normal lysosomes and their Pu-239 content can no longer be set free by addition of Triton X-100 (7).From their enzyme content they are neither typical lysosomes nor do they correspond to other organelles which have been considered (Fig.4 and (7)). At present we can only speculate that they may be old lysosomes, i.e. residual bodies.

Further studies should show how far our findings can be generalized. Most probably lysosomes are also the primary binding site in mouse and Syrian hamster (10).The results of extensive biochemical studies with beagle liver (11,12) do not conflict with the assumption of initial lysosomal binding, though they do not prove it unequivocally. In analogy to our long-term experiments with Chinese hamster, Pu-239 also became bound to structures other than lysosomes in beagle liver after 90 days (12).

In conclusion, the species differences in the biological half-life of transuranium elements probably reflect differences at the physiological and biochemical level, perhaps with regard to the composition and biological fate of lysosomes or to the mechanisms of biliary metal transport. Another explanation could be reabsorption from the bile duct system (not from the gastrointestinal tract)in species with slow elimination but no evidence exists for such a process. Further the type of liver cells in which Pu-239 is bound must be defined carefully,before further speculations are possible.

References

1. D.M.Taylor,In:Proceedings of the 6th international IRPA congress Berlin, 1984, in press
2. A.Seidel, G.Darai, R.Flügel, W.Hofmann, W.Sontag, Health Phys.43, 239 (1982) (This paper contains further references).
3. R.Gruner, A.Seidel, R.Winter, Radiat.Res.85, 367 (1981)
4. J.K.Beckman, K.Owens, W.B.Weglicki, Lipids 16, 796 (1981)
5. M.C.Balani, M.Lehman, U.Sütterlin, A.Seidel, Int.J.Radiat.Biol. 43, 559 (1983)
6. K.Hannig, H.-G.Heidrich,In:Meth.Enzymology XXXI (S.Fleischer, L.Packer edits.), Academic Press, N.Y., 746 (1974)
7. U.Sütterlin, Kernforschungszentrum Karlsruhe -Report KFK-3385(1982)
8. C.de Duve, R.Wattiaux, Rev.Physiol.28, 435 (1966)
9. N.F.LaRusso, L.J.Kost, J.A.Carter, S.S.Barham, Hepatology 2, 209 (1982)
10. R.Winter, A.Seidel, Radiat.Res.89, 113 (1982)
11. F.W.Bruenger, B.J.Stover, W.Stevens, Health Phys.21, 679 (1971)
12. F.W.Bruenger, W.Stevens, D.R.Atherton, D.S.Bates,In:The Health Effects of Plutonium and Radium (W.S.S.Jee edit.), J.W.Press, Salt Lake City, 199 (1976)

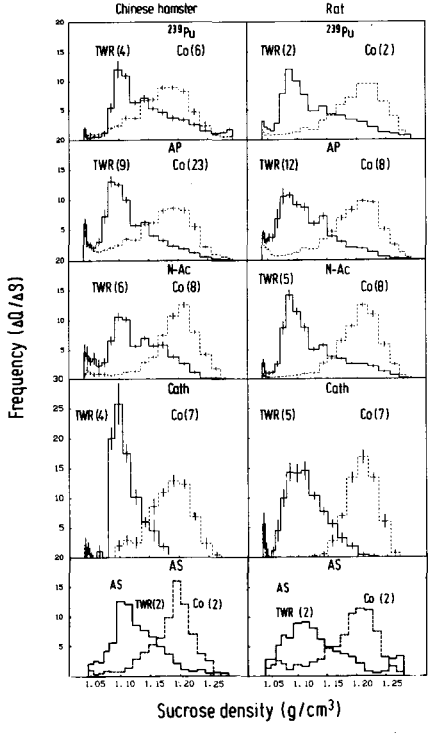

Fig.1: Distribution of Pu-239 and lysosomal marker enzymes after centrifugation in a linear sucrose gradient of the liver ML-fraction (further explanations see text). Arithmetic means, vertical bars are standard errors of means, number of experiments in brackets.

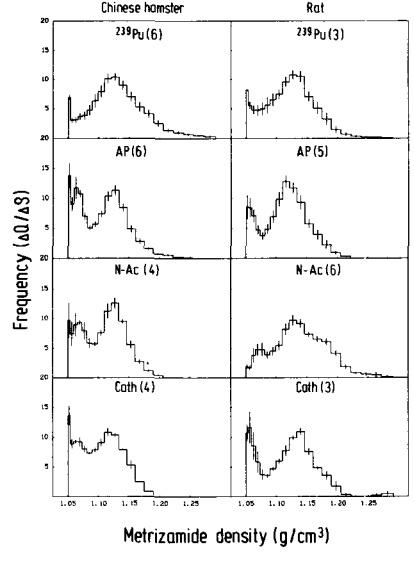

Fig.2: Distribution of Pu-239 and lysosomal marker enzymes after centrifugation in a linear metrizamide gradient of the liver ML-fraction (see Fig.1).

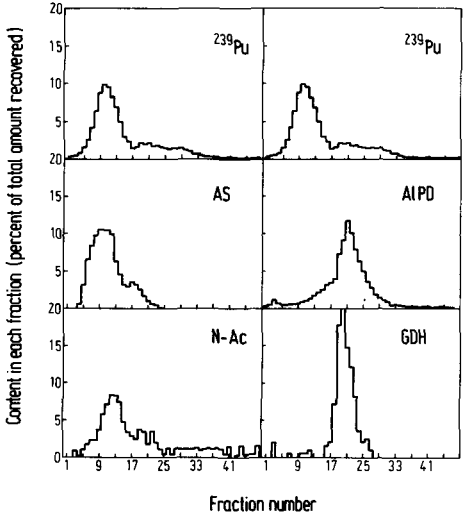

Fig.3: Profiles of Pu-239 and marker enzymes after free flow electrophoresis of the liver ML-fraction of Chinese hamster liver (Cathode on left side, further explanations see text) at day five after Pu-239 injection. Representative for three experiments.

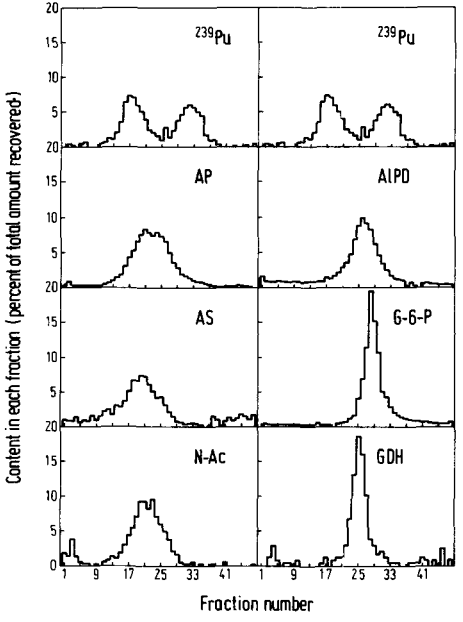

Fig.4: Profiles for Pu-239 and marker enzymes after free flow electrophoresis of the ML-fraction of Chinese hamster liver. Representative for five experiments between day 200 and 300 after Pu-injection.

REMOVAL OF DEPOSITED ^{238}Pu, ^{239}Pu AND ^{241}Am BY PROLONGED ORAL INTAKE OF DTPA

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Oral treatment with chelating agents would be the most convenient method for the removal of incorporated actinides from the body, especially in cases where protracted administration is indicated. There has been, however, a general assumption that DTPA, the present agent of choice, would be ineffective when given orally because of the low rate of intestinal absorption.

In a previous paper it was shown that after a prolonged addition of Zn-DTPA to drinking water substantial fractions of Pu-239 were removed from the rat skeleton and liver (Taylor D.M. and Volf V., Health Phys. 38, 147-158, 1980). The aim of the present investigation was to collect further experimental data to support the hypothesis that prolonged administration of quite small amounts of DTPA represents a true alternative to the hitherto accepted schedule of repeated DTPA injections.

In the first experiment on female Heiligenberg rats (Tab.1) the effects of various concentrations of Zn-DTPA in drinking water on the retention of Pu-238 and Pu-239 were compared with the effect of Ca-DTPA injected in human equivalent doses (1g Ca-DTPA in a 70kg man). After Ca-DTPA injections the contents of both isotopes in the bone and liver were reduced by about 40% and 70%, respectively. A similar effect was achieved with Pu-238 and Pu-239 by drinking 10^{-4} and 10^{-3} molar Zn-DTPA, respectively. Thus, the effect of oral Zn-DTPA was substantially more pronounced with Pu-238 than with Pu-239, both in the bone and liver. This suggests that with low chelate concentrations in the tissues following oral administration of DTPA the mass of Pu becomes a critical factor: For equal activities the mass of Pu-238 is about 280 times less than that of Pu-239.

The total DTPA intake in rats drinking Zn-DTPA was between 1 and 30 times higher than in those injected with Ca-DTPA yet it proved equally effective in reducing the contents of Pu-238 in the bone and liver, respectively. The effect in the liver of oral DTPA can be easily understood when assuming that only about 3% of ingested DTPA is absorbed from the intestine. In the bone, however, it is obviously important that a low level of DTPA is maintained for longer periods of time, thus preventing the redeposition of the small quantities of Pu-238 released. The higher mass of released Pu-239 can be bound only if about 30 times higher local DTPA concentrations are achieved.

In man, chelation therapy is indicated after incorporation of substantially lower amounts of Pu-239 than are those usually administered in animal experiments (e.g., the activity injected in the above experiment was 7μCi per 70kg body weight). Thus, animal studies with the low mass Pu-238 seem to be a suitable model for human accidental incorporation of Pu-239.

In the second experiment (Tab.2), Zn-DTPA was added to drinking water so that the intake equalled approximately 300μmoles per kg body weight per day, i.e. 10 x the human equivalent dose. Male Sprague-

Dawley rats were injected with Pu-238 and Am-241 citrate, 4d or 30d before the beginning of treatment which was continued until 105d,when the animals were sacrificed. The two actinides were administered simultaneously in order to compare their response to treatment under identical conditions. The changes in the overall retention of Am-241 in vivo were followed by repeated whole body counting.

At 15 weeks postinjection the whole body retention of Am-241 as well as the content of Pu-239 and Am-241 in the skeleton were reduced to approximately 60% and 40% by the treatment with Zn-DTPA beginning late or early after injection of the actinides, respectively. In the soft tissues, both Zn-DTPA treatments reduced the Pu-238 content to below the detection limit, and that of Am-241 to about 10% and 50% in the liver and kidneys, respectively.

The latter results indicate that substantial fractions of Pu-238 and Am-241 can be removed even by delayed administration of DTPA in drinking water, except for a small fraction of Am-241 which remains fixed in the kidney, even if DTPA treatment begins only 4 d after Am-241 injection.

It is at present not clear what happens to the Zn-DTPA introduced into the gastrointestinal tract. The acid stomach juice may lower the stability of the chelate, while during the neutralization process in the small intestine the previously freed DTPA may bind Zn as well as Ca, Fe and other metals, from food and/or from the gut mucosa. All this could result in a greater oral Zn-DTPA toxicity than that expected from the injection studies on the relative toxicity of Ca- and Zn-DTPA. However, as shown previously (see above), there was no evidence of impaired intestinal DNA synthesis or iron utilization in rats exposed to up to $3x10^{-2}$ molar Zn-DTPA in drinking water for 21d. The total amount of Zn-DTPA given in this toxicity study was about twice the highest amount administered in the present experiments.

As seen in Table 3, initial uptake of Pu-238 and Am-241 in the distal half of the femur was twice as high as that in the proximal half. The subsequent release of the two actinides from the distal half was more pronounced than that from the proximal one. Thus, the distal: proximal content ratio decreased with time, especially after treatment; this decrease appeared earlier with Am-241 than with Pu-238. The latter suggests slight differences in binding of the two actinides in the long bones.

In conclusion, the results obtained indicate a surprisingly good effect on the retention of Pu-238 and Am-241 of protracted treatment using small amounts of Zn-DTPA in drinking water, even when treatment was started as late as one month after actinide injection. The total amounts of Zn-DTPA used in the present study were equal to only one half or less of those shown previously to be non-toxic.

TABLE 1. RETENTION OF ACTINIDES IN FEMALE HEILIGENBERG RATS AS INFLUENCED BY DTPA

Treatment			Actinide content (% of injected amount; arithm.means + S.E.)			
			Skeleton		Liver	
Substance	Concn. or amount	Total amount (mmol / kg)	Pu-238	Pu-239	Pu-238	Pu-239
Controls 4d	–	–	63.9 ± 4.1	65.8 ± 2.3	19.2 ± 1.7	20.9 ± 1.0
Controls 28d	–	–	54.1 ± 1.3	61.7 ± 1.9	4.1 ± 0.2	4.6 ± 0.4
Zn-DTPA (Drinking-water)	$1x10^{-4}$ M	0.09	40.8* ± 1.3	59.5 ± 2.6	2.9* ± 0.2	4.5 ± 0.6
	$3x10^{-4}$ M	0.27	35.2* ± 1.3	56.6 ± 2.9	2.2* ± 0.1	3.7 ± 0.3
	$1x10^{-3}$ M	0.90	32.6* ± 0.9	54.2* ± 3.0	1.9* ± 0.2	3.4* ± 0.3
	$3x10^{-3}$ M	2.70	30.5* ± 1.5	49.5* ± 2.8	1.8* ± 0.1	1.9* ± 0.1
	$1x10^{-2}$ M	9.00	30.9* ± 1.8	39.4* ± 2.9	1.2* ± 0.1	1.1* ± 0.1
	$3x10^{-2}$ M	27.00	31.4* ± 1.4	27.7* ± 2.6	1.0* ± 0.1	0.6 ± 0.1
Ca-DTPA (s.c.in-jection)	30 µmol/kg	0.27	34.2* ± 1.2	37.7* ± 2.9	1.4* ± 0.1	1.2* ± 0.1

Treatment started 4d after i.v.injection of Pu-citrate (0.1µCi/kg) and continued 3 x per week for 3 weeks. Rats (5-10 per group) sacrificed 4 weeks post Pu.

* Statistically significant difference between control and treated group ($p < 0.05$) (t-test).

TABLE 2. RETENTION OF ACTINIDES IN MALE SPRAGUE DAWLEY RATS AFTER DRINKING Zn-DTPA

Group (Time post injection)		Actinide content (% of injected amount; arithmetic means ± S.E.)					
		Skeleton		Liver		Kidneys	
		Pu	Am	Pu	Am	Pu	Am
Controls	4d	51.9 ± 1.1	39.4 ± 1.0	13.5 ± 0.7	33.0 ± 1.7	1.9 ± 0.1	2.3 ± 0.1
Controls	105d	51.6 ± 1.0	36.9 ± 1.0	0.7 ± 0.2	1.3 ± 0.5	0.2 ± 0.03	1.0 ± 0.1
Late DTPA	105d	32.9 ± 1.6	23.5 ± 1.0	< 0.2	0.1 ± 0.01	< 0.03	0.6 ± 0.1
Early DTPA	105d	21.9 ± 0.6	14.3 ± 0.4	< 0.2	0.1 ± 0.01	< 0.03	0.5 ± 0.1

Zn-DTPA was added to drinking water (concentration: 3×10^{-3} molar) from day 4 (early DTPA) or day 30 (late DTPA) up to day 105 after i.v.injection of 0.5μCi/kg of Pu-238 and Am-241 citrate.

TABLE 3. EFFECT OF ORAL Zn-DTPA ON THE RETENTION OF ACTINIDES IN THE FEMUR OF SPRAGUE-DAWLEY RATS

Group (Time post injection)		Actinide content (% of injected amount; arithmetic means + S.E.)					
		Distal half		Proximal half		Distal/Proximal Ratio	
		Pu	Am	Pu	Am	Pu	Am
Controls	4d	1.75±0.06	1.28±0.03	0.90±0.05	0.67±0.04	1.98±0.15	1.93±0.11
Controls	105d	1.61±0.04	1.09±0.04	0.96±0.03	0.76±0.02	1.68±0.06	1.43±0.05
Late DTPA	105d	0.99±0.04	0.62±0.03	0.66±0.04	0.55±0.03	1.50±0.03	1.14±0.07
Early DTPA	105d	0.58±0.03	0.38±0.01	0.51±0.02	0.34±0.01	1.16±0.10	1.13±0.05

For explanations see Table 2.

DEVELOPMENT OF A STOCHASTIC LUNG MODEL - EXPERIMENTAL INVESTIGATION OF ENHANCED DEPOSITION AT BRONCHIAL AIRWAY BRANCHING SITES AND MONTE CARLO MODELLING OF RANDOM PARTICLE WALKS

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INTRODUCTION

Aerosol deposition calculations are commonly based on simplified models of the anatomical structure of the human lung. These lung models consist in most cases of a symmetric arrangement of straight cylindrical tubes with specified diameters, lengths, and number of airways. Morphometric studies have revealed, however, that a realistic airway system is highly asymmetric with randomly varying linear dimensions and branching angles. This paper presents preliminary results of our joint effort to develop a more realistic model for aerosol deposition in human lungs by Monte Carlo simulation techniques in close connection with detailed measurements of regional particle deposition in bifurcating airways.

AEROSOL DEPOSITION IN BIFURCATION UNITS

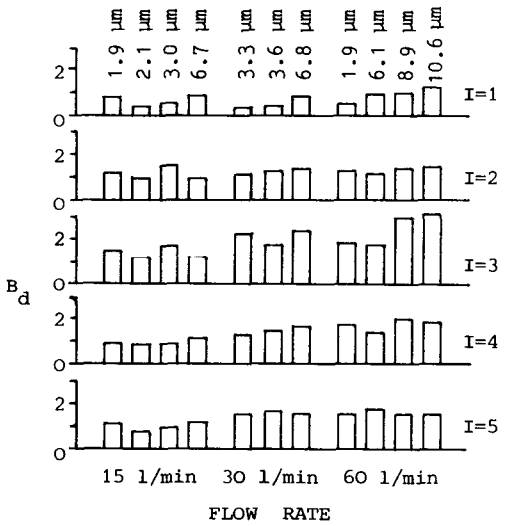
Methods

A combined larynx-TB model system, aerosol dispersion regulating apparatus, and laboratory protocols used to quantitate aerosol deposition have been discussed in detail elsewhere (1); for brevity, salient features only will be presented here. Six generation, hollow, single-pathway TB tree models were constructed from Silastic^{R} E RTV silicone rubber following the dimensions of a symmetric, dichotomously branching morphology of human conducting airways (2). Three Silastic^{R} larynx casts were made from original replica casts moulded from autopsy specimens of human larynges (3). The interior configurations of the three different casts were flow-related to constant inspiratory rates of 15, 30 and 60 l/min. The larynx casts insured that airflow patterns in downstream TB models were as physiologically realistic as possible to simulate in vivo conditions (laryngeal jet and flow instabilities created at the glottis).

Monodisperse ammonium fluorescein aerosols were produced with a spinning-top instrument. Aerosol mass median aerodynamic diameters varied from 1.9 to 10.6 μm, with geometric standard deviations between 1.11 and 1.18. Aerosol dispersion was precisely regulated with an arrangement of rotameters and vacuum pumps; airflow was divided symmetrically at each airway bifurcation. TB models were dissected into well-defined segments to quantitate deposited aerosol mass. Pieces were immersed in measured volumes of aqueous 0.1 N ammonium solution to dissolve deposited particles; aliquots of each wash were analyzed with a spectrophotofluorimeter.

Results and Dosimetry Implications

Measured localized deposits, or "hot spots", at airway branching sites are compiled in Fig. 1. Measured bifurcation doses showed two recurrent patterns. First of all, B_{α} values were minimum at the trachea-main bronchi junctions, and secondly, values were maximum entering airway generation 3. The I=1, 2, etc., terminology is



Bifurcation dose, B_d, is defined as:

B_d = (aerosol mass deposited within bifurcation zone/aerosol mass deposited within sister airways) (surface area of sister airways/surface area of bifurcation zone).

$B_d > 1$ indicates that a bifurcation zone has received a greater dose than if the total aerosol mass deposited in sister airways was uniformly distributed

Fig. 1 Distribution of deposited aerosol mass

in accordance with the Weibel (2) definition. Importantly, dose distributions within bifurcation zones per se were heterogeneous. Inspection of TB model interiors prior to dissection and washing revealed enhanced aerosol mass concentrations at carinal ridges (or airstream dividers) within each bifurcation zone.

Histological investigations (4, 5) have detected concentrated neoplastic and preneoplastic lesions at bifurcation sites. Therefore, the data of Fig. 1 and the very limited submicron particle findings (6, 7) suggest that TB sites, where bronchogenic carcinomas originate, may be directly related to initial sites of enhanced particle deposition. Moreover, it has been suggested (8) that mucociliary clearance rates may be slowest at airway branching sites. Epithelial cells located there could have increased exposure to inhaled aerosols of health effects concern. Together with preferential particle deposition at bifurcations originally, this latter factor may give additional support to a hypothesis that bronchogenic cancers are induced within bifurcation zones.

Sufficient experimental evidence exists that "hot particles" in the lungs are less carcinogenic than uniformly distributed alpha-emitting radionuclides. The main reason for this finding is increased cell killing by multiple alpha particle traversals in the cellular cluster around the "hot particle". In the case of "hot spots" at bronchial branching sites the number of emitted alpha particles is significantly smaller than for "hot particles" making multiple cellular hits a very unlikely event. Thus, enhanced deposition increases only the number of cells traversed, and, consequently, the probability for cancer induction.

STOCHASTIC LUNG MODELLING

Anatomical Lung Model

Current anatomical lung models do not reflect the variability of the structural components of the human lung which leads to experimentally observed random variations of particle deposition (9). To our knowledge, the most detailed morphometric

data are provided by Raabe et al. (10). In their extensive investigation, each airway segment is assigned a unique identification number which allows statistical sampling of airway parameters to construct probability distributions for each airway generation. From all data files, only the files DS 1, 8 and 10 have been selected for statistical analysis, because a) these sets contain the largest amount of data, and b) the cut-off criterion is the same in all three files.

For mean airway diameters, a "saturation value" of 0.8 mm is reached in generation 13 (Weibel morphology). Further statistical analysis of the data base suggests, however, that this finding illustrates the experimental cut-off rather than it is a realistic anatomical effect. It is interesting to note that Yeh and Schum (11) adopted a smaller saturation value of about 0.44 mm, while Weibel (2) preserved a decrease in diameter, although the curve flattens out in the peripheral airways. In Fig. 2, the coefficients of variation, i.e. the standard deviation σ divided by the mean μ, are plotted vs. the generation number (Weibel morphology) for the three files. No values were included for the first five generations, since the number of data in these generations was too small to yield statistically significant values. Up to about generation 13 a steady increase of the coefficient of variation can be observed. The following slight decrease is most probably due to the already discussed cut-off effect. For comparison, Yu et al. (12) assumed a linear relationship between coefficient of variation and generation number, starting with 0.12 for the trachea and ending with 0.24 for generation 23. Although a rigorous statistical analysis could not confirm in all cases the originally anticipated lognormal distributions, sampling from lognormal distributions will hardly be the main source of error.

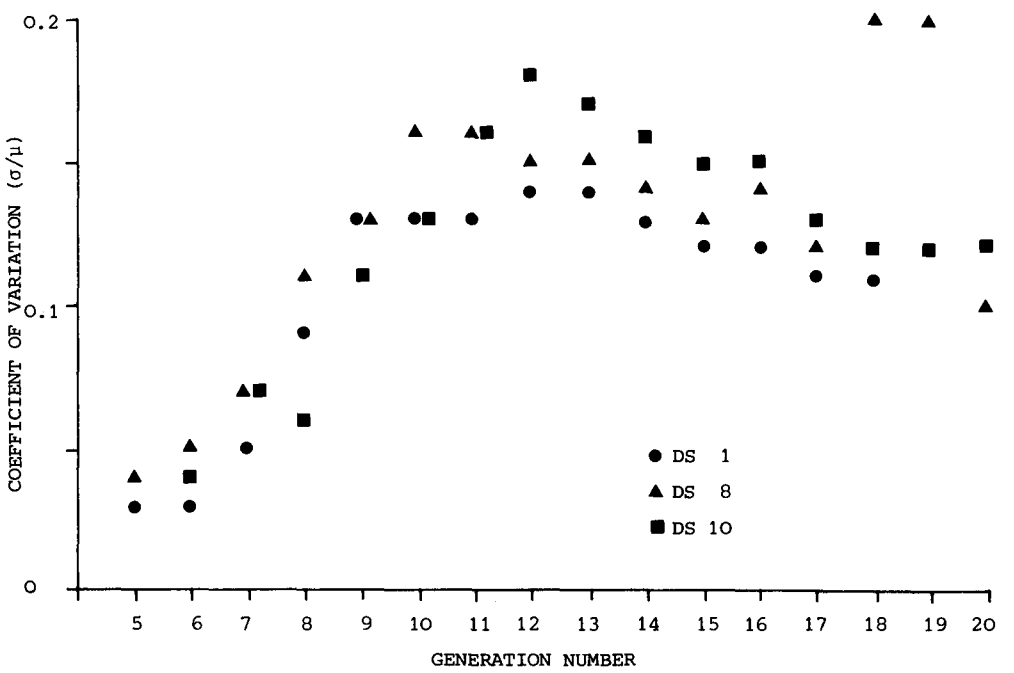


Fig. 2 Coefficients of variations for airway diameters from generations 5 through 20

A complete analysis of the morphometric data yielded further interesting results:

(a) The average ratio of the parent cross-sections to the cross-sections of both daughters is 0.82 for generations 5-16, indicating that parent and daughter dimensions are correlated.
(b) The frequency distribution of the diameter ratio of minor to major daughters, illustrating the asymmetry of branching, shows that high asymmetries are statistically related to large parent diameters.
(c) The most likely distribution of airway lengths within a defined generation is again lognormal (within the above limitations) with coefficients of variations similar to the ones found for diameters.
(d) Studies of correlation between diameters and lengths of the same generations in the same lobes revealed that the length of a tube belongs always with the highest probability to the same equiprobability class as its diameter.

Monte Carlo Random Pathway Model

The most effective method to handle complex asymmetric and statistically varying systems is stochastic modelling. For the modelling of random walks of aerosol particles during inhalation and exhalation in a random airway structure, the Monte Carlo code IDEAL was developed. For the simulation procedure, all geometrical data are selected randomly from the prespecified distributions. During the flight of the particle, time is continuously recorded and the selected dimensions are corrected for the change of the geometry during the breathing cycle by a sine function. The decision whether a particle continues its path in the major or minor daughter tube is also made by random selection. In the case of a deposition event no actual deposition of the particle is simulated; only the statistical weight attributed to the particle is decreased. The deposition probabilities for each airway generation and the exhaled fractions as a function of time are recorded at the end of each simulation.

REFERENCES

(1) Martonen, T.B. (1983). J. Aerosol Sci. 14, 11.
(2) Weibel, E.R. (1963). Morphometry of the Human Lung. Springer Verlag, New York.
(3) Schlesinger, R.B., Bohning, D.E., Chan, T.L. and Lippmann, M. (1977). J. Aerosol Sci. 8, 429.
(4) Kotin, P. and Falk, H.L. (1959). Cancer 12, 147.
(5) Auerbach, O., Stout, A.P., Hammond, E.C. and Garfinkel, L. (1961). New Engl. J. Med. 265, 253.
(6) Schlesinger, R.B., Cohen, V.R. and Lippmann, M. (1974). In: Experimental Lung Cancer: Carcinogenesis and Bioassays. Springer Verlag, New York.
(7) Schlesinger, R.B. and Lippmann, M. (1978). Environ. Res. 15, 424.
(8) Macklin, C.C. (1956). J. Thorac. Surg. 31, 238.
(9) Heyder, J., Gebhart, J., Roth, C., Stahlhofen, W. et al. (1978). J. Aerosol Sci. 9, 147.
(10) Raabe, O.G., Yeh, H.C. and Schum, G.M. (1976). Tracheobronchial Geometry: Human, Dog, Rat, Hamster. Lovelace Foundation Report LF-53.
(11) Yeh, H.C. and Schum, G.M. (1980). Bull. Math. Biol. 42, 461.
(12) Yu, C.P., Nicolaides, P. and Soong, T.T. (1979). Am. Ind. Hyg. Assoc. J. 40, 999.

COMPARATIVE DISTRIBUTION OF ^{238}U, ^{234}U, AND $^{239+240}Pu$ IN THE SAME SETS OF HUMAN TISSUES OF GENERAL POPULATION

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INTRODUCTION

The purpose of these studies is sufficient understanding of the metabolism of the different actinides in both man and experimental animals. Toward this end, we have determined the concentrations of "naturally" occurring uranium, thorium and plutonium in tissues of control beagle dogs from our research colony (1), and the concentrations of plutonium and thorium in human tissues from two states of the United States (2). Determining both plutonium and thorium in the same human tissue set allowed comparison of the distribution pattern of these two actinides, which have very similar chemical and biological properties. The present work describes the distribution of uranium and plutonium in human tissues. Because measurements of both elements are made on the same set of human tissues, effects of environmental, physiological, and dietary intake should be the same for both elements. The present work will also suggest the distribution patterns of the two actinides.

EXPERIMENTAL

Collection of Tissues

Ten sets of tissues were obtained at autopsy from persons who died suddenly from two states of the United States (Pennsylvania and Colorado). All ten tissue sets included lung, liver, kidney, vertebrae and ribs, but in some cases, we also obtained lymph nodes and spleen. For each subject, the tissue samples were placed in individual plastic bags and packaged in separate labeled containers. The specimens were frozen after autopsy and transported to our laboratory packed in dry ice.

The date of death, age at death, sex, residential history, smoking and drug history, past medical history and occupational data were obtained for each individual. The cause of death was determined by either gross inspection or toxicological examination. All the organs were examined for degeneration or impairment, and for evidence of tumors or carcinomas.

Radiochemical Determinations of Uranium and Plutonium

Uranium and plutonium were determined using the radiochemical procedures of Singh et al. (3,4). The weighed amounts of soft tissues were spiked with 1-2 dpm of ^{232}U and ^{242}Pu tracers. The tissues were wet ashed with HNO_3 followed by a mixture of HNO_3 and H_2SO_4, with occasional additions of a few drops of HNO_3 and H_2O_2. Uranium and plutonium were coprecipitated with iron as hydroxides, and extracted into a 20% tri-lauryl amine (TLA) solution in xylene from 10 M HCl. Plutonium was first back-extracted by reducing to the trivalent state with 0.05 M NH_4I solution in 8 M HCl. Uranium was then back-extracted by shaking with an equal volume of 0.1 M HCl. Uranium and plutonium were electrodeposited separately onto platinum discs and counted alpha-spectrometrically using a surface barrrier silicon diode and a multi-channel analyzer.

Weighed amounts of bone tissue were spiked with ^{232}U and ^{242}Pu tracers followed by slow heating on a hot plate. The bone was dry ashed in a muffle furnace at 550°C, followed by wet ashing with HNO_3, with occasional additions of HNO_3 and H_2O_2, until the evolution of brown fumes ceased. Uranium was reduced to the tetravalent state by addition of 200 mg $SnCl_2$ and 25 ml HI. Uranium and plutonium were coprecipitated as oxalate by 10% oxalic acid. The precipitate was heated in a muffle furnace at 550°C overnight and dissolved in 10 M HCl. Uranium and plutonium were extracted, back-extracted, electrodeposited and counted alpha-spectrometrically as described for soft tissues.

RESULTS

The mean concentrations of plutonium and uranium are given in Figure 1. Among soft tissues, the concentrations of $^{239,240}Pu$ were highest in liver, ranging from 0.14 to 2.41 pCi/kg (mean: 1.06 ± 0.26 pCi/kg wet weight), followed by lung with a range of 0.06 to 1.79 pCi/kg (mean: 0.43 ± 0.22 pCi/kg). Concentrations in kidney ranged from 0.01 to 0.09 pCi/kg (mean: 0.05 ± 0.01 pCi/kg). In lymph nodes, the concentrations ranged from 0.04 to 3.54 pCi/kg (mean: 1.02 ± 0.56 pCi/kg); however, these values are as high as the analytical errors associated with the measurement.

The concentrations of ^{238}Pu in all soft tissues were below the detection limit except in liver, where the concentration ranged from 0.02 to 0.10 pCi/kg (mean: 0.05 ± 0.01 pCi/kg wet weight).

The concentration of $^{239,240}Pu$ in vertebrae ranged from -0.12 to 0.34 pCi/kg (mean: 0.18 ± 0.04 pCi/kg). Concentrations ranged from 0.03 to 0.41 pCi/kg (mean: 0.21 ± 0.06 pCi/kg) in ribs, and from 0.002 to 1.46 pCi/kg (mean: 0.46 ± 0.26 pCi/kg) in sternum. The concentrations of ^{238}Pu were below the detection limit in these bones.

The concentrations of ^{238}U were highest in lymph nodes, ranging from 0.51 to 19.88 pCi/kg (mean: 5.2 ± 3.0 pCi/kg). In lung, concentrations ranged from 0.27 to 2.28 pCi/kg (mean: 1.18 ± 0.26 pCi/kg); in kidney they ranged from 0.12 to 3.40 pCi/kg (mean: 0.68 ± 0.32 pCi/kg). Liver contained the lowest concentrations, ranging from 0.02 to 0.76 pCi/kg (mean: 0.15 ± 0.08 pCi/kg).

The concentrations of ^{234}U followed the same pattern. Lymph nodes contained the highest concentration, ranging from 0.99 to 19.2 pCi/kg (mean: 7.6 ± 2.7 pCi/kg), followed by lung with a range of 0.19 to 3.16 pCi/kg (mean: 1.36 ± 0.39 pCi/kg). Concentrations in kidney ranged from 0.10 to 3.89 pCi/kg (mean: 0.91 ± 0.36 pCi/kg). Liver contained the lowest concentrations, ranging from 0.08 pCi/kg to 0.94 pCi/kg (mean: 0.24 ± 0.09 pCi/kg).

The concentrations of ^{238}U were similar in vertebrae and ribs, ranging from 0.09 to 4.66 pCi kg (mean: 0.85 ± 0.43 pCi/kg), and from 0.14 to 2.81 pCi/kg (mean: 0.87 ± 0.40 pCi/kg), respectively. The concentrations of ^{234}U were also similar in vertebrae and ribs with ranges of 0.17 to 5.75 pCi/kg (mean: 1.08 ± 0.54 pCi/kg), and of 0.26 to 4.47 pCi/kg (mean: 1.17 ± 0.58 pCi/kg), respectively. The sternum contained slightly higher concentrations of ^{238}U and ^{234}U with ranges of 0.25 to 3.91 pCi/kg (mean: 1.63 ± 0.62 pCi/kg) and 0.27 to 6.54 pCi/kg (mean: 2.34 ± 1.08 pCi/kg), respectively.

The concentrations of ^{235}U in all tissues were below the limit of detection (<0.01 pCi).

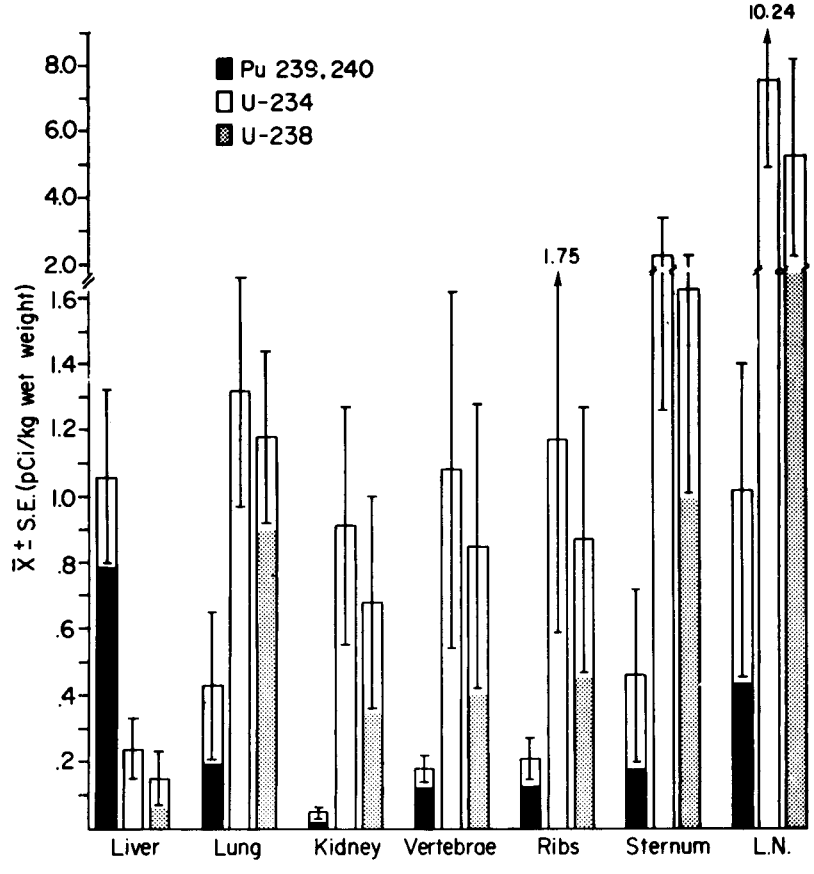

Figure 1. Comparisons of the concentrations (pCi/kg wet weight) of $^{239+240}Pu$, ^{234}U, and ^{238}U in the same sets of human tissues from the general population of the United States.

DISCUSSION

A comparison of uranium and plutonium concentrations are given in Figure 1. There are certain obvious differences between the concentrations of uranium and plutonium in different organs. The large differences between the concentrations of uranium isotopes (^{238}U and ^{234}U) and $^{239,240}Pu$ in lung may simply be due to the higher concentrations of uranium isotopes in the air as compared to plutonium isotopes.

The other obvious difference is between the concentrations of uranium and plutonium in the liver. The concentration of ^{239,240} Pu was highest in the liver, yet the concentration of uranium was lowest in the liver. This may be due to the fact that inhaled plutonium, when cleared from the lungs and circulated in the bloodstream, hydrolyzes to form colloidal particles which are taken up by reticuloendothelial (RE) cells in the liver. Uranium is believed to occur in the body as UO_2^{++}, which does not undergo hydrolysis; it accumulates mostly in bone and clears through the kidney. Hydrolysis occurs more readily in elements with higher valence and small ionic size; therefore, plutonium, which is mostly in the tetravalent state and has an ionic size of 0.93 Å, hydrolyzes more readily than uranium, which exists in the hexavalent state but behaves as divalent because it is present as UO_2^{++} and has a larger ionic size than Pu(IV).

The higher concentrations of ^{238}U and ^{234}U as compared to $^{239,240}Pu$ in bones may possibly be due to a larger intake of uranium through the food chain and inhalation. Only a very small amount of plutonium is inhaled, and a negligible amount is ingested through the food chain.

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REFERENCES

(1) Singh, N.P., Zimmerman, C., Taylor, G.N., and Wrenn, M.E. Background concentrations of actinides in control beagle dose. Radiation Research 91: 296 (1982).

(2) Singh, N.P., Ibrahim, S.A., and Wrenn, M.E. Plutonium concentrations in human tissues: Comparison to thorium. Health Physics 44: 213-220 (1983).

(3) Singh, N.P., Zimmerman, C.J., and Wrenn, M.E. Simultaneous determinations of alpha-emitting isotopes of uranium and plutonium in soft tissues. Mikrochimica Acta 3: 61-70 (1983).

(4) Singh, N.P., Zimmerman, C., Lewis L.L. and Wrenn, M.E. Simultaneous determination of alpha emitting isotopes of uranium and plutonium in bone. J. Radioanal. Chem. (in press).

A MODEL FOR THE AGE-DEPENDENT SKELETAL RETENTION OF PLUTONIUM*

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INTRODUCTION

The metabolic model for plutonium given in Publication 30 of the ICRP[1] assumes that 90% of Pu reaching blood is divided equally between the skeleton and liver, with the remainder going to other tissues and early excreta. Plutonium deposited in the skeleton is assumed to be apportioned uniformly on cortical and trabecular surfaces, where it is retained with a biological half-time of 100 years. For the liver, a biological half-time of 40 years is assigned.

This model was designed principally for use in estimating long-term dose commitments and was not intended to be an accurate mathematical description of the actual metabolic and physiologic processes involved in the retention and translocation of Pu in the body, even for an average adult. In this paper we describe a model intended to more accurately depict these processes, and we examine the implications of our model concerning the dose as a function of age to radiosensitive tissues of the skeleton. To describe the model concisely it is convenient to view the body as subdivided into the liver, the skeletal compartments shown in Fig. 1, and all remaining tissue.

UPTAKE AND TRANSLOCATION OF PU BY THE SKELETON

Experimental data for beagles together with limited human autopsy data indicate that the fraction of circulating Pu depositing in the skeleton decreases from about 0.7 in newborns to about 0.5 in adults.[2-6] Plutonium is deposited initially on bone surfaces (pathways K and L in Fig. 1), with higher deposition being at sites with red marrow and lower deposition at sites with yellow marrow.[7] In the adult, nearly all of the red marrow is in trabecular bone,[8] and deposition on trabecular bone appears to be greater than on cortical bone. In children, some or all of the marrow in cortical bone is active,[8] and a fairly uniform distribution on cortical and trabecular bones is expected. Thus it is assumed that Pu is divided evenly between cortical and trabecular surfaces in children; in adults, 60% of the initial deposit in the skeleton is assumed to be on trabecular surfaces and 40% on cortical surfaces (cf. Ref. 5).

Bone surfaces labeled with Pu may remain unchanged, or they may be buried by formation of new bone (pathways A and C) or resorbed by osteoclasts (pathways B and D).[9] The rate of removal from surfaces by burial or resorption depends on the age of the individual and on the bone surface type (trabecular or cortical).[9] To a large extent, bone addition and resorption occur on opposite sides of a bone or bone trabecula, so that the bone may be pictured as continually drifting in a given direction.[10] Bone drift apparently occurs at all ages but may not be the predominant type of bone remodeling in mature bone. Rather, resorption and addition may often occur at the same location in mature bone; first an area of bone is excavated by osteoclasts, and then the same area is refilled with osteoid, which is later mineralized.[11] It is assumed that all bone remodeling in nonadults involves bone drift, and the removal rate of Pu from bone surfaces is estimated as the sum of the resorption rate and the formation rate (since these processes are assumed not to overlap). For the adult, an intermediate scenario involving both types of bone remodeling is assumed; the burial rate in bone is assumed to be one-half the formation rate, and the removal rate from surfaces is estimated as

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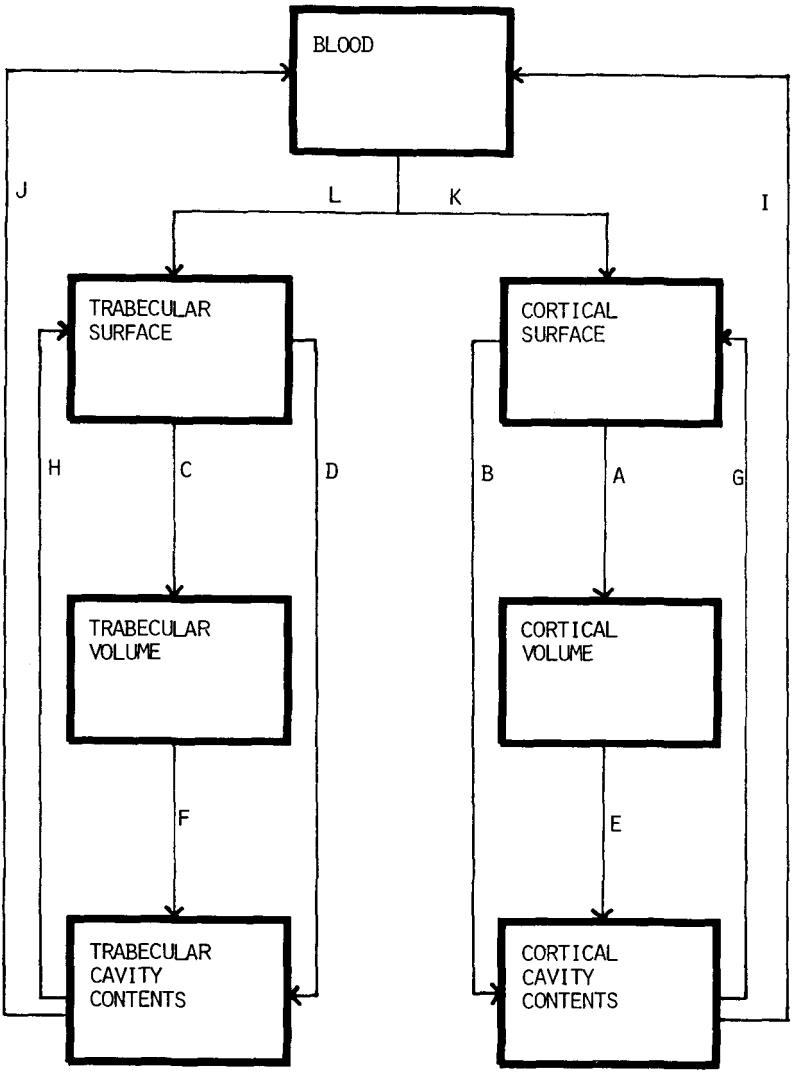

Fig. 1. Compartments and pathways in the model of the skeleton.

the resorption rate plus one-half the formation rate. The age-dependent formation rate is estimated for cortical bone from tetracycline data and for trabecular bone from strontium removal rates from vertebrae.[11-13] Resorption and formation rates are assumed to be equal; this may result in an overestimate of the half-time of Pu on bone surfaces in children.

Plutonium resorbed by osteoclasts may be released and concentrated by macrophages in bone cavities, particularly in marrow.[9] The length of time that Pu remains in these macrophages is not known. Experiments with beagles suggest that several half-times have elapsed after two years.[9] For the present calculations we have arbitrarily assumed a half-time of 0.25 years.

Pu buried in bone volume may eventually become distributed in volume as the bone section is remodeled.[9] If the resorption rate for trabecular bone is k per year, then buried Pu may begin to be resorbed about 1/k years or more after exposure, depending on the fraction of remodeling attributable to bone drift. Here account must be taken of the fact that k varies with age. Because of the slow turnover time for cortical bone at most ages, much of the Pu buried in cortical bone of adolescents and adults may never be recycled.

Experimental data suggest that both local (pathways G and H) and systemic (pathways I and J) redeposition onto bone surfaces occur, although we suspect that systemic redeposition may predominate in the highly vascularized bone of children. For lack of information, we have attempted to minimize errors in the present calculations by assuming that all Pu released from marrow is channeled through blood.

PU IN NON-SKELETAL TISSUES AND EXCRETA

Experimental data indicate that the fraction of Pu depositing in liver increases with age, from about 0.1 in very young animals to about 0.3 in adults.[3-4,6] It is known from animal studies[14-16] that Pu may leave the liver both in blood and in bile, that soluble Pu is taken up by hepatocytes but is later transferred to reticuloendothelial (RE) cells, and that Pu may reside for years in the RE system. It is also suggested by autopsies of persons occupationally exposed that Pu may reside for many years in the human liver. No differences with age in removal of Pu from liver are discernable from animal studies or from autopsy data for non-occupationally exposed humans. Our model assumes that, at all ages, about 4% of Pu going to liver is removed in feces with a relatively short half-time (a few days) and the remainder is removed to blood with a half-time of 10 years. These values were chosen because they lead to reasonable agreement with human data.

The biological half-time of Pu in soft tissues other than liver appears to be on the order of a few hundred days.[6] In this model we have assumed a half-time of 500 days; estimated doses to sensitive skeletal tissues are fairly insensitive to this value within reasonable limits of the half-time.

Plutonium released to blood may be excreted or may be recycled to the skeleton, liver, and remaining tissue. The effective half-time of Pu in blood is about one day, although an accurate description of retention in blood over a few weeks or months requires several exponential terms.[6] The amount excreted per day at equilibrium is estimated as 8.4% of the integrated activity in blood, on the basis of data for humans. Plutonium reaching blood after release from skeleton, liver, or remaining tissue is assumed to be redeposited or excreted in the same percentages as the original deposition in blood. Processes of retention and translocation outside the skeleton and liver are assumed to be independent of age.

IMPLICATIONS FOR SKELETAL DOSIMETRY

Integrated doses over 50 years were calculated for various age groups, assuming that one unit of activity of Pu-239 was injected into the bloodstream. It is estimated using this model that about three-fourths of the alpha energy released in the skeleton over a period of 50 years is deposited in non-sensitive tissues, for all ages at injection. Moreover, for an adult, the ICRP model leads to an estimated dose commitment to active marrow that is about 2 times higher than that

estimated using the present model. According to our model, there is little age dependence in 50-year dose commitments to sensitive skeletal tissues, mainly because of the large amount of recycling that occurs over 50 years among the skeleton, liver, and other tissues. This example refers only to an initial unit injection and does not consider the potentially large difference with age in the amount of Pu-239 that may reach the bloodstream in a more typical exposure situation, particularly through the ingestion pathway.

REFERENCES

[1] Limits for Intakes of Radionuclides by Workers, International Commission on Radiological Protection, Publication 30, Pergamon Press, 1979.

[2] Lloyd, R. D., Atherton, D. R., McFarland, S. S., Mays, C. W., Stevens, W., Williams, J. L., and Taylor, G. N., Health Phys. 30, 1976, 47-52.

[3] Lloyd, R. D., McFarland, S. S., Atherton, D. R., and Mays, C. W., Radiat. Res. 75, 1978, 633-641.

[4] Lloyd, R. D., McFarland, S. S., Atherton, D. R., Bruenger, F. W., Taylor, G. N., and Mays, C. W., Health Phys. 35, 1978, 211-215.

[5] Larsen, R. P., Oldham, R. D., and Toohey, R. E., Macrodistribution of plutonium in the human skeleton, in Actinides in Man and Animals, Proceedings of the Snowbird Actinide Workshop, ed. M. E. Wrenn, 1979, 191-196.

[6] Durbin, P. W., Plutonium in man: a new look at the old data, in Radiobiology of Plutonium, ed. B. J. Stover and W. S. S. Jee, J. W. Press, Salt Lake City, 1972, 469-530.

[7] Wronski, T. J., Smith, J. M., and Jee, W. S. S., Radiat. Res. 83, 1980, 74-89.

[8] Cristy, M., Phys. Med. Biol. 26, 1981, 389-400.

[9] Jee, W. S. S., Health Phys. 22, 1972, 583-595.

[10] Enlow, D. H., Principles of Bone Remodelling, Charles C. Thomas, Publishers, Springfield, Ill., 1963.

[11] Frost, H. M., Calc. Tiss. Res. 3, 1969, 211-237.

[12] Klusek, C. S., USDOE Rep. EML-363, 1979, I-125 to I-138.

[13] Leggett, R. W., Eckerman, K. F., and Williams, L. R., Health Phys. 43, 1982, 307-322.

[14] Taylor, G. N., Jee, W. S. S., Dockum, N. L., Hromyk, E., and Brewster, L., Translocation of Pu-239 in Beagle Livers, Univ. of Utah Report COO-119-234, 1966, 70-84.

[15] Stevens, W., Bruenger, F. W., Atherton, D. R., and Stover, B. J., Subcellular Distribution of ^{241}Am(III) and ^{239}Pu(IV) in Livers Studied Serially, Univ. of Utah Report COO-119-244, 1971, 159-174.

[16] Boocock, G., Danpure, C. J., Popplewell, D. S., and Taylor, D. M., Radiat. Res. 42, 1970, 381-396.

RE-EVALUATION OF RADIATION EXPOSURE DUE TO ADMINISTRATION OF RADIOACTIVE CALCIUM-47

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INTRODUCTION

The radioactive calcium isotope Ca-47 is used in medical research and diagnostics to estimate the Ca-turnover in the skeleton and to measure its absorption in the gastro-intestinal-tract.
The investigation reported here was performed to estimate the radiation dose resulting from an injection of Ca-47, because the existing rather sophisticated models describing the metabolism of alkaline earth elements (e.g. ICRP 1972, Johnson 1981) do not take into account the age-dependence of the dose absorbed. They give no information on its variability due to age, sex, body weight, diseases etc. and are mainly designed for nuclides of longer halflifes than Ca-47 (4.5 days).

METHODS

Measurements were performed to follow the retention functions of the injected activity (150 kBq Ca-47) in the total body, in the blood and in the wrist (Malluche 1978, Roth 1980):

- by means of a whole body counter at 4-18 times between 0.08 and 32 days after injection (55 healthy adults, 45 adult patients and 30 children with different diseases);
- by means of 7-15 blood-samples during the time-interval of 10 minutes to 6 days after injection (7 healthy adults, 45 adult patients and 8 children with different diseases);
- by means of a specially designed probe (shielded 3"*3" NaJ(Tl)-detector) at 19-25 times from time of injection to 10 days thereafter (5 healthy adults, 4 patients and 5 children with different diseases).

The values of the absorbed dose were calculated mainly in accordance to the principles described by the ICRP (1979). The calculation is based on the relation:

$$D_T = 1.6 * 10^{-10} * \sum_S U_S * SEE(T,S) \quad (Sv)$$

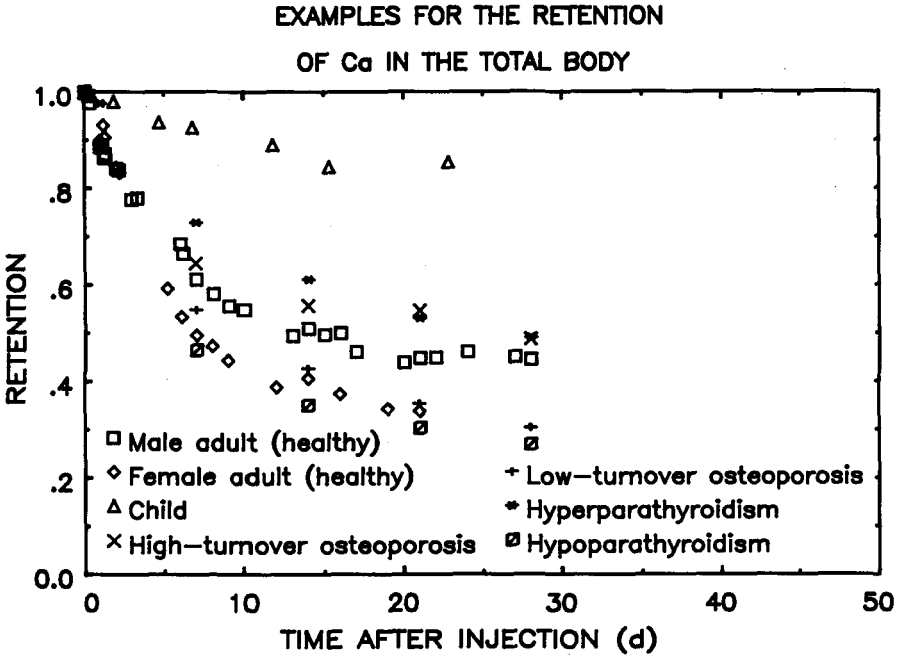
where D_T is the radiation dose absorbed in the tissue T, U_S is the number of nuclear decays in the tissue S and SEE(T,S) is the so-called specific effective energy (MeV/g), which is proportional

to the dose in T per decay in S. Values of SEE(T,S) were calculated for 6 different ages according to a method described previously (Henrichs 1982).

U_S is computed by the integration of the retention function of the radionuclide in the tissue S. In the case of Ca-47 the source tissues S of interest are the bone-surfaces, soft-tissues and the blood. The bone retention was extracted from the long-term component of the total body retention; the uptake-phase was calculated from the measured activities in the wrist, taking into account the activity in blood and soft-tissue seen simultaneously. The remaining number of nuclear decays was assumed to be homogeneously distributed. On the basis of these calculations, values of the dose to 29 tissues and the effective dose equivalent were computed, taking into account the contribution of the radioactive Ca-47-daughter Sc-47.

RESULTS AND DISCUSSION

Typical total body retention functions are shown in figure 1. Values of retention are corrected for radioactive decay of Ca-47 to the time of administration. The data show a dependence on age similar to bone formation rates as obtained from bone histology by Frost (1969). There is also a tendency to lower retention values in females than in males beyond the age of 30 years. In certain clinical situations long term retention may deviate from normal behaviour by a factor of 2. Total body retention of healthy subjects in this series is somewhat less than that given by ICRP (1972).



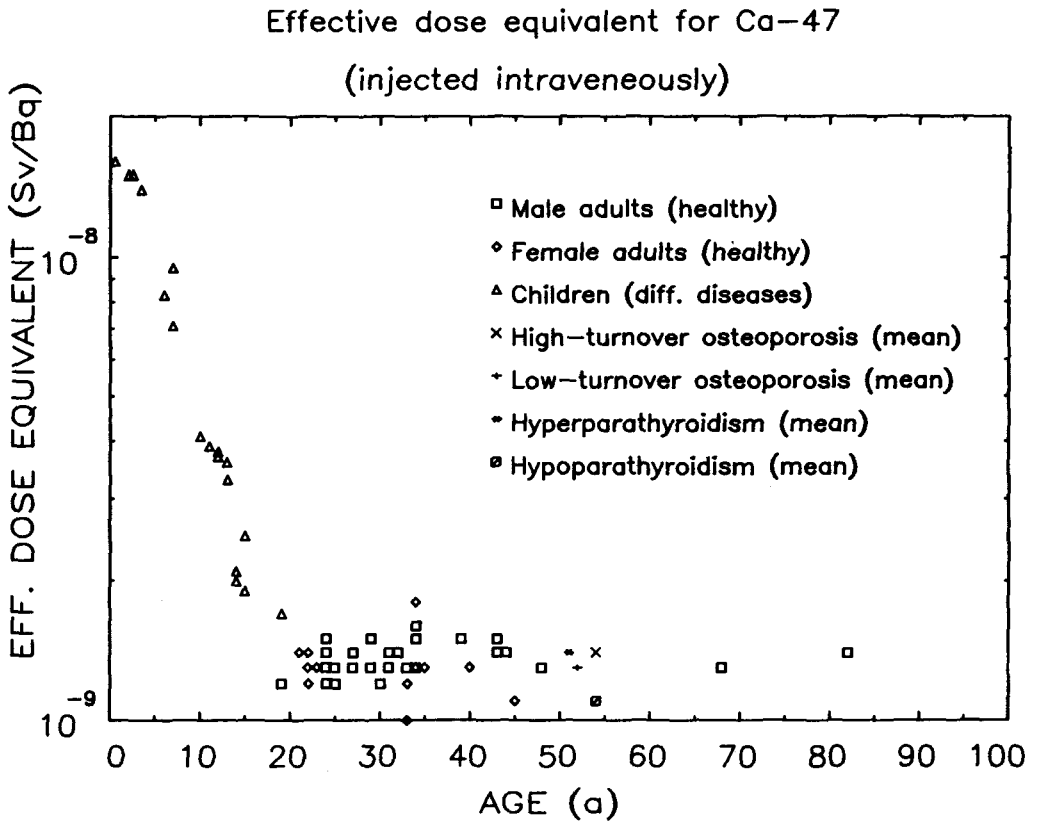

Effective dose equivalent for Ca–47
(injected intraveneously)
EFF. DOSE EQUIVALENT (Sv/Bq)
10^-8
10^-9
0
10
20
30
40
50
60
70
80
90
100
AGE (a)
Male adults (healthy)
Female adults (healthy)
Children (diff. diseases)
High–turnover osteoporosis (mean)
Low–turnover osteoporosis (mean)
Hyperparathyroidism (mean)
Hypoparathyroidism (mean)

Figure 2 summarizes the results of the dose calculations. For ages from 0.5 to 20 years the effective dose equivalent decreases with increasing age by one order of magnitude. The mean value for adults is $1.3*10^{-9}$ Sv/Bq (90%-confidence-interval: $1.2-1.4*10^{-9}$ Sv/Bq).

The statistically significant differences between the retentions of female and male, healthy and diseased adults do not result in important differences of the corresponding dose values; this is due to the dominating influence of the short physical halftime and to the smoothing effect of the calculation of the effective dose equivalent as a weighted mean of doses to numerous tissues.
Considering the longer physical halflife of Ca-45 or other alkaline earth radioisotopes an increased influence of the factors mentioned above on the absorbed dose is to be expected.

REFERENCES

Frost H.M.: Tetracycline-based histological analysis of bone remodelling.
Calcified Tissue Research, vol. 3, 1969, p. 211

Henrichs K., Kaul A.: Age dependent values of specific absorbed fractions and specific effective energies for the dosimetry of internal emitters.
Radiation protection dosimetry, vol. 3, 1982, p. 71-73

ICRP Publication 20: Alkaline earth metabolism in adult man.
Pergamon Press, Oxford 1972

ICRP Publication 30: Limits for intakes of radionuclides by workers.
Pergamon Press, Oxford, 1979

Johnson J.R., Myers R.C.: Alkaline earth metabolism: a model useful in calculating organ burdens, excretion rates and committed effective dose equivalent conversion factors.
Radiation Protection Dosimetry, vol. 1, 1981, p. 87-95

Malluche H.H., Werner E., Ritz E.: Intestinal Absorption of Calcium and whole-body calcium retention in incipient and advanced renal failure.
Mineral Electrolyte Metabolism, vol. 1, 1978, p. 263-270

Roth P., Werner E.: Vergleich verschiedener Methoden zur Bestimmung der intestinalen Kalziumabsorption des Menschen.
Medizinische Physik (editor: U. Rosenow), HÜthig Verlag, Heidelberg, 1980.

BIOKINETICS OF INTERNAL EMITTERS AND ANNUAL LIMITS ON INTAKE*

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1 Objectives

The Radiation Protection Ordinance of the Federal Republic of Germany (Strahlenschutzverordnung 1976) in its present form is essentially based on ICRP Publication 2 (1960). Additional knowledge acquired since that time in regard to radiation effects and biokinetics of radionuclides has been included in ICRP Publication 30 (1979). These recommendations were passed by the Council of the European Communities in the form of an EC-Guideline (EG-Richtlinie 1980). This guideline represents the basis for the Federal Ministry of the Interior, BMI, for a revision of the Radiation Protection Ordinance in the Federal Republic of Germany. For occupationally exposed persons the annual limit of 0.05 Sv effective dose equivalent will be adopted, but the annual dose limit for non-stochastic effects of 0.5 or 0.3 Sv be substituted by organ dose limits of 0.5 Sv (extremities and skin), 0.05 Sv (gonads and uterus) and 0.15 Sv (other organs), the latter being referred to as BMI dose limitation system.

Secondary limits ALI (annual limit on intake) and DAC (derived air concentration) being relevant for practical radiation protection, essentially depend on the biokinetic data of the radionuclide compound in addition to the physical properties of the radionuclide and the dose limitation system. Since it is difficult for the reader of ICRP Publication 2 (1960), sometimes also of ICRP Publication 30 (1979), to trace the selection of biokinetic data used therein, a research project was initiated within the scope of which own proposals of biokinetic data for 32 elements were derived from available literature and compared with those contained in ICRP Publication 30 (1979) for the purpose of examining the reproducibility of the values used therein and the sensitivity of the ALI in regard to variations of biokinetic data.

2 Methods

Initially, data available from literature on uptake to blood, inhalation class, distribution and retention were compiled and tabulated. They were scored as to their suitability for absorbed dose calculations. From this, own proposals for biokinetic data were derived and compared with those in ICRP Publication 30 (1979). For both sets of data the ALI was calculated and evaluated for 263 radionuclides.

3 Results

3.1 Biokinetic data

In regard to their conformity, the results may be assigned to 3 classes as shown in Table 1 for uptake to blood f_1, inhalation class, fractional distribution F and biological half-life T(biol).

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*) Financially supported by the Bundesminister des Innern of the Federal Republic of Germany

Table 1 – Comparison of own proposed biokinetic data for 32 elements with data used in ICRP Publication 30 (1979)

Quantity	Identical	Slightly modified (factor < 2)	Significantly modified (factor ≥ 2)
Uptake to blood f_1	77 %	5 %	18 %
Inhalation class	76 %		24 %
Fractional distribution F	22 %	22 %	56 %
Biological half-life T(biol)	30 %	35 %	35 %

Accordingly, our proposed values for fractional distribution and half-lives differ in about 75 % of cases either slightly (factor < 2) or significantly (factor ≥ 2) from those values in ICRP Publication 30 (1979); for uptake to blood f_1 and inhalation class this proportion is approximately 25 %.

3.2 Annual limit on intake (ALI)

Results from calculations of the ALI can be described by the ratio

$$R = \frac{\text{ALI (own proposed biokinetic data)}}{\text{ALI (ICRP biokinetic data)}}$$

In Table 2 the relative frequencies of R for values lower and greater than 1 are summarized.

Table 2 – Relative frequencies of R for values lower and greater than 1

Range of R values	Frequency of R values in dose limitation systems according to ICRP / EG		BMI	
	Oral Intake	Inhalation	Oral Intake	Inhalation
$0.99 \leq R \leq 1.01$	62 %	51 %	69 %	58 %
$R > 1.01$	16 %	30 %	16 %	27 %
$R < 0.99$	22 %	19 %	15 %	15 %

Table 3 shows the cumulative distribution of R values within given ranges.

Table 3 – Cumulative distribution of R values

Range of R around 1, characterized by factors as given below	Frequency of R values in dose limitation systems according to ICRP / EG		BMI	
	Oral Intake	Inhalation	Oral Intake	Inhalation
< 1.5	83 %	82 %	81 %	78 %
< 2	89 %	87 %	85 %	84 %
< 3	91 %	91 %	90 %	89 %
≥ 3	9 %	9 %	10 %	11 %

From 263 radionuclides examined the following values of the geometric mean of R were derived (see Table 4).

Table 4 – Geometric mean of R values in both dose limitation systems

Route of intake	Geometric mean of R values in dose limitation systems according to ICRP / EG	BMI
Oral intake	1.08	1.09
Inhalation	1.25	1.28

Accordingly, the ALI values calculated on the basis of biokinetic data from ICRP Publication 30 (1979) on the average are more restrictive than those calculated on the basis of our own proposed biokinetic data. One of the reasons for this result may be seen in the frequently quite conservative ICRP estimates of the long-term retention component which, in many cases, could not be substantiated by us from the available literature.

In spite of – from an overall view – relatively few changes of ALI when substituting ICRP biokinetic data by own proposed data, there are some chemical forms of radioisotopes of the following elements for which the ALI values, calculated from these two sets of data, differ by a factor of 3 or more:

Co, Hg, In, Ni, Np, Se, Te, Zr

As a more detailed analysis of these data shows, essentially the following reasons apply for the described differences in the two sets of ALI values:

- the compounds with $R > 3$ are particularly those for which the above mentioned, quite conservative long-term retention values used in ICRP Publication 30 (1979) cannot be easily substantiated from the available literature;
- for compounds with $R \leqslant 1/3$ our proposed biokinetic data are frequently more differentiated with higher organ uptakes than those given in ICRP Publication 30 (1979).

4 Discussion

These investigations not only confirm the relative insensitivity of the ALI values that may be expected for the concept of effective dose equivalent in respect to modified biokinetic data. Although in the BMI dose limitation system the effective dose equivalent limit is the determining factor in only a few cases (in approximately 10 % compared to approximately 75 % in the ICRP/EG dose limitation system for the 263 radionuclides considered here), the ALI in both dose limitation systems reacts comparatively insensitive to modifying the biokinetic data, particularly for changes of uptake to blood.

One explanation may be that for oral intake the GI-tract frequently represents the limiting organ, so that the biokinetic data for the remaining organs are practically without relevance, and thus a change of the fractional absorption from 10^{-4} to 10^{-1} with a corresponding portion of the administered activity in the GI-tract of 99,9 % or 90 %, respectively, only results in a 10 % change of ALI, as long as the GI-tract remains the limiting organ.

5 Conclusions

The biokinetic and dosimetric models used in ICRP Publication 30 (1979) are so complex that the effect of modified biokinetic data on the ALI values can hardly be predicted and thus in most cases require a detailed calculation. At the Institute for Radiation Hygiene of the Federal Health Office, the required programs and data files, which were developed and implemented within the scope of the above mentioned research project, are available.

In view of the variability and the scarce knowledge on biokinetic data this relative insensitivity of ALI values, even within the BMI dose limiation system, in respect to modifying the biokinetic data used is quite remarkable. On account of this relative insensitivity together with a now given possibility for rapid calculation of ALI values also for modified biokinetic data, future discussions and problems in this field may be dealt with more easily.

6 References

EG-Richtlinie
Richtlinie des Rates vom 15. Juli 1980 zur Änderung der Richtlinien, mit denen die Grundnormen für den Gesundheitsschutz der Bevölkerung und der Arbeitskräfte gegen die Gefahren ionisierender Strahlen festgelegt wurden
80/836/Euratom
Amtsblatt der Europäischen Gemeinschaften
L 246, 23. Jahrgang, vom 17. September 1980

ICRP Publication 2
Recommendations of the International Commission on Radiological Protection
Report of Committee II on Permissible Dose for Internal Radiation (1959)
Pergamon Press, Oxford (1960)

ICRP Publication 30
Limits for Intakes of Radionuclides by Workers
Pergamon Press, Oxford (1979)

Strahlenschutzverordnung
Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung – StrlSchV) vom 13. Oktober 1976 (BGBl. I, 2905; berichtigt 1977, BGBl. I, 184 u. 269), zuletzt geändert durch die Erste Verordnung zur Änderung der Strahlenschutzverordnung vom 22. Mai 1981 (BGBl. I, 445)

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BIOKINETICS OF INTERNAL EMITTERS AND ABSORBED DOSE TO THE HUMAN FETUS

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1 Objectives

In the medical use of radiopharmaceuticals as well as in the unintentional intake of radionuclides, e.g. by emission from nuclear power plants, incorporation-caused radiation exposure of the fetus as compared to that of the adult may be of particular interest. Therefore, as a research project, biokinetic data of selected radionuclide compounds were extracted from literature references on investigations in man and animal with the purpose to provide the basis for comparative assessments of fetal and adult radiation dose from incorporated radionuclides.

2 Methods

A comparative assessment of internal radiation exposure of fetal and adult tissues requires that the factors determining absorbed dose be known. In case of a tissue dose being mainly determined by the activity in this tissue, the assessment is based on the value of the proportion of fetal (f) and adult (a) tissue doses as follows:

$$\frac{D_f}{D_a} = \frac{(A_{o,f}/m_f) \cdot T_f \cdot \phi_f}{(A_{o,a}/m_a) \cdot T_a \cdot \phi_a} \qquad (1)$$

D	mean absorbed dose in respective tissue (Gy or rd)
A_o	maximum tissue activity (Bq or μCi)
m	mass of tissue (kg or g)
T	effective half-life (s or h)
ϕ	absorbed fraction

Accordingly, the following proportions should be considered for fetal and adult tissues: activity concentrations, effective half-lives and mean absorbed fractions.

There are only some radionuclides for which biokinetic data (activity concentrations and effective half-lives) measured in *human* fetuses and newborns are available that are suitable for a comparative assessment of fetal and adult radiation exposure. These are iodine 131 and iron 59 from medical use as well as strontium 90, caesium 137 and again iodine 131 from fallout due to nuclear weapons' tests. Literature references on the above radionuclides were evaluated as to activity concentrations and half-lives in total body or organs in order to perform absorbed dose calculations.

The evaluation of literature references on *animal* studies was made for activity concentrations in fetuses and mother animals as follows: summary of methods used, compilation of measured results from authors, evaluation of measurement results for ratios of activity concentrations in fetal and adult tissues, discussion of activity concentration ratios, conclusions and list of references evaluated.

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*) Financially supported by the Bundesminister des Innern of the Federal Republic of Germany

Uniform total body distribution			
Ratio of activity concentrations in fetus and mother or reference man:			
Humans:			~1
Animals:	0.3 – 0.6		
Effective half-lives			
Human fetus:		no data	
Assumption:		50 d (equal to maternal)	
Newborn:		15 d	
Adult:	male:	110 d	
	female:	80 d	
	pregnant:	50 d	
Ratio of half-lives in fetus and mother:			~1
Ratio of total body doses to fetus and mother:			~1
Ratio of total body doses to fetus and reference man:			~0.5

Table 1: **Data and assumptions for estimating the total body dose to fetus and mother or reference man from caesium 137**

Organ with highest radiation exposure			Liver
Activity concentrations in the liver			
Fetus:	11th week:	0.319 %/g	
	22th week:	0.150 %/g	
	mean value:	0.230 %/g	
Adult:		0.0067 %/g (12%/1800g)	
Ratio of activity concentrations in fetus and adult			34
Effective half-lives, corrected for growth			
Fetus:	(0.12 x 20 d) + (0.88 x 2.4 d)	= 4.5 d	
	storage erythropoesis		
Adult:	$T_{eff} = T_{phys}$	= 45 d	
Ratio of half-lives in fetus and adult			0.1
Liver doses per kBq (μCi)			
Fetus:	11th week:	61 mGy (165 mrd)	
	22th week:	39 mGy (104 mrd)	
Adult:		33 mGy (89 mrd)	
Ratio of liver doses in fetus and adult			1.2 – 1.9

Table 2: **Data and assumptions for estimating the liver dose to fetus and adult from iron 59**

Organ with highest radiation exposure			thyroid
Ratio of activity concentrations in fetus and mother (8 values)			1.7
Effective half-lives, corrected for growth			
Fetus:	17th week:	1.9 d	
	birth:	4.6 d	
Mother:		7.5 d	
Ratio of effective half-lives in fetus and mother:			0.25 – 0.60 (17th week – term)
Absorbed fraction of emitted β-energy			
Fetus:	⩽1		
Mother:	1		
Ratio of thyroid doses			0.43 – 1.0

Table 3: **Data and assumptions for estimating the thyroid dose to fetus and mother from iodine 131**

Organ with highest radiation exposure			bone
Ratio of activity concentrations in fetus and adult (from fallout-measurements during 1957–1966)			
– based on pCi/g Ca			~2 – 4
– based on pCi/g bone (wet)			~0.5 – 1
Effective half-lives			
Fetus:	6 months	21 d	
	9 months	97 d	
Infant:	1 year	350 d	
	2 years	600 d	
Adult:	several thousand days (power function)		
Ratio of half-lives in fetus and adult (conservative assumption: adult half-life ≈ 1000 d)			0.02 – 0.1
Absorbed fraction of energy emitted			
Fetus:	6 months	65 %	
Infant:	1 year	80 %	
Adult:		93 %	
Dose commitment per kBq(μCi)			
Fetus:	for intake at 6 months	0.22 Gy (0.6 rd)	
	for intake at 9 months	0.67 Gy (1.8 rd)	
Adult:		2.2 Gy (6.0 rd)	
Ratio of bone doses to fetus and adult:			0.1 – 0.3 (6 months – 9 months)

Table 4: **Data and assumptions for estimating the bone dose to fetus and adult from strontium 90**

3 Results

The biokinetic data from measurements in *humans* and the assumptions used for estimating the fetal and adult radiation exposure after intake of caesium 137, iron 59, iodine 131 and strontium 90 are summarized in Tables 1 – 4.

Based on *animal* experiments, the ratios of activity concentrations in fetal and adult tissues were derived for antimony, cerium, cobalt (chloride and vitamin B 12), manganese, nickel, niobium, technetium (pertechnetate, albumin, iron complex, polyphosphate, pyrophosphate, sulfur colloid, sulfur colloid albumin particles) and zinc (chloride, citrate, sulfate), whereas fetal retention data are practically non-existent. These ratios are of considerable variability, in some cases up to three orders of magnitude. This is not only due to different experimental conditions and metabolic characteristics of different species, but also due to the authors' selection of measuring times not necessarily representing the maximum fetal and adult activity concentrations.

4 Discussion

Ideally, for a comparative assessment of fetal and adult radiation dose, the activity concentrations versus time in the tissues of fetal and adult animals should be known. As an approximation, these can be replaced by the maximum activity concentrations and mean half-lives. For the total body of mother or mother animals this maximum is equal to the mean activity concentration immediately after administration and can be calculated as ratio of the administered activity and body weight. For the fetal total body, the measurements performed at selected times may not include the maximum activity concentration, even if measurements are performed at different times after administration. For organs, the interpretation of ratios of activity concentrations in fetal and adult tissues is even more difficult due to the different retention functions and the delay during diaplacental transfer, since the measurements refer only to selected times and the retention function, especially in relation to its maximum, is generally not known.

Furthermore, for a comparative assessment of the mean total body or organ dose to the human fetus, the fetal growth must be considered which, at an assumed constant total body or organ activity, leads to a decrease of the mean activity concentration. With a fetal weight of approximately 100 g at the 16th week of pregnancy and of about 3.300 g at the 40th week, this implies a purely growth related decrease of the mean activity concentration with a mean half-life of about 5 weeks. This decrease was included in the above calculations.

5 Conclusions

For those radionuclides where results from investigations in man are available (caesium 137, iron 59, iodine 131, strontium 90), a significantly higher radiation exposure of the fetus in comparison to that of the adult is not to be expected: the ratio of estimated absorbed doses to the fetus and adult is approximately in the range of 0.1 – 1.9.

Due to the scarcity and variability of relevant data on animal experiments from literature references, it appears necessary to perform studies on fetal and adult biokinetics that are particularly designed for comparative dose calculations. This concerns mainly niobium, nickel and vitamin B 12, but also technetium and zinc.

Acknowledgements

Thanks are due to Mrs. R. LeMar for assistance in translation and Mrs. I. Zeitlberger for typesetting the manuscript.

METABOLISM OF INGESTED ORGANICALLY BOUND TRITIUM IN ORGANIC MILK CONSTITUENTS AND IN TISSUES OF TWO SPECIES OF RUMINANTS

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Introduction

The important routine releases of tritium to the environment by the nuclear industry have constituted a major reason for the Commission of the European Economic Community to initiate and coordinate a research programme on the radio-ecological behaviour of tritium. In our laboratory, organically bound tritium (OBT) was fed to experimental animals in order to investigate the incorporation and turnover of OBT into organic material of animal tissues and secretions.

Milk formation was used as a model for tritium metabolism studies because incorporation and turnover of tritium in organic milk constituents (milk protein, milk fat, lactose) may be considered representative for tritium behaviour in other tissues. Moreover, milk can be sampled for a long time, the main organic compounds can be separated easily and their tritium content determined. Two species of ruminant animals (cow and miniature goat) were chosen for providing the milk samples.

Material and Methods

Grass was grown under a plastic cover and sprayed regularly with tritiated water (THO). It was cut when full grown and dried to hay. After evacuation of the remaining THO by drying in a vacuum stove at 37°C, the hay was fed to two Frisian cows for 28 days and to several miniature goats for about 160-170 days. All animals were equipped with rumen fistules which insured accurate daily dosing of the same quantity of OBT. They were kept on a dietary regime of good hay without supplements.

Milk was sampled daily in all animals and milk fat, casein, lactose and milk water were separated by conventional methods. Casein was combusted to water in a Packard Sample Oxydizer and the tritium content of the combustion water determined. All other constituents were pipetted directly into the scintillation cocktail and the tritium activity was determined by liquid scintillation counting. More details of the radiochemical procedures adopted are described elsewhere.

Results and Discussion

Figures 1 and 2 show the tritium incorporation in organic milk constituents and in milk water of the cow and goat during and following the administration of OBT. The curves are similar in many respects which is not surprising because cow and goat are both ruminant animals and they are closely related physiologically. However, some differences also exist. For example, the cow incorporates about 2.5 times as much tritium in fat as in casein whereas the tritium content in casein of goat's milk is somewhat higher than it is in milk fat.

The differences and similarities are summarized in Table 1 which shows tritium content in milk constituents of cow and goat for the same daily intake of 1 mCi of OBT. When the figures for casein, milk fat and lactose are considered, it can be seen in the last column of Table 1 that the cow incorporates about three times the amount of tritium into milk fat as does the goat. The differences in tritium content of casein and lactose between cow and goat can be explained by differences in intake of feed. Under our experimental conditions, the cows

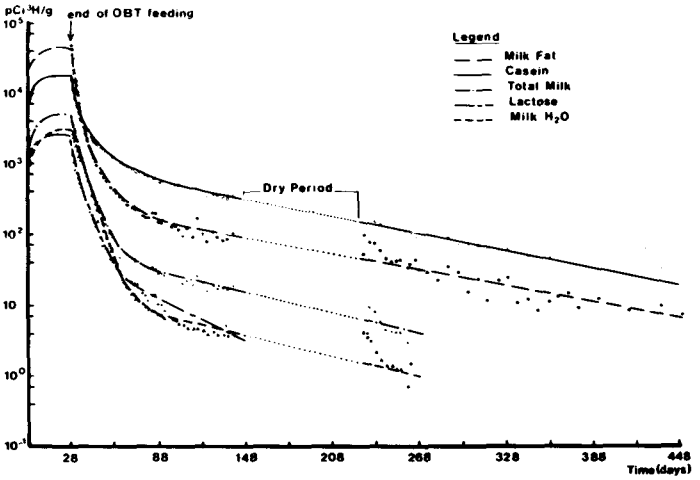

Fig.1 Tritium activity in milk fat, casein, lactose, total milk and milk water during and following ingestion of OBT by a lactating cow for 28 days.

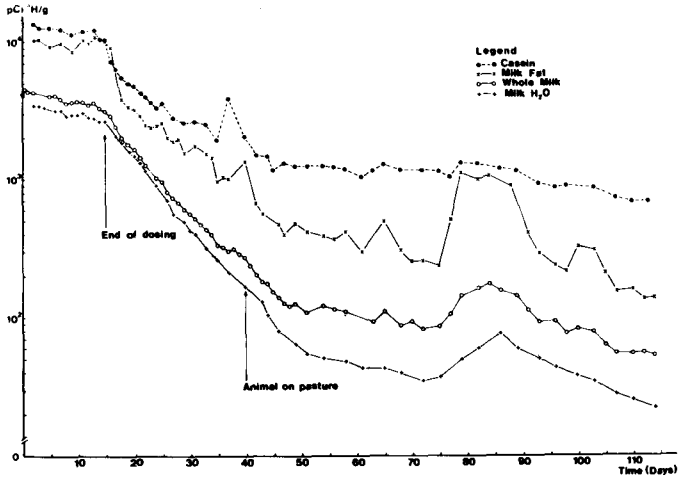

Fig.2 Tritium activity in some milk constituents and in whole milk of the goat, during and following daily ingestion of OBT for 162 days.

ingested about 21 times as much as did the goats. In other words, dilution with organically bound stable H appears to explain satisfactorily the differences in tritium incorporation in casein and lactose between cow and goat. But this is not the case for milk fat. The reason for the much lower incorporation of tritium into milk fat by the goat remains unknown. The higher ^{3}H content of milk water in the goat probably results from the slower turnover of body water in this animal as compared to the cow.

	^3H Activity (pCi/g) at 1 mCi/day Intake of OBT		
	Cow	Goat	Ratio $\frac{\text{Goat}}{\text{Cow}}$
Casein	54250	1240917	22.87
Milk fat	137978	1013730	7.35
Lactose	7998	199754	24.98
Milk H_2O	9299	311276	33.47

Table1 : Tritium activity in some organic milk constituents and in milk water in two ruminant species after continuous ingestion of 1 mCi of organically bound tritium (OBT).

Regression analysis of the curves describing the decrease of tritium activity in organic milk constituents and in milk water (Fig. 1 and 2) has shown that these curves can be resolved into three components with rapid, intermediate and slow half-lives (Table 2). Although equilibrium conditions for the pool(s) of slow turnover were centainly not obtained in the cow, the results for cow and goat show reasonably good agreement. There is a rapid decrease initially, representing uptake of organic precursors for milk synthesis from the animal's feed, followed by a much slower decrease later. The half-life values of the slow components for casein and milk water show higher values in the goat but this may be due to the much longer period of administration of OBT to goats.

	$T^1/_2$ of Component (days)					
	Cow			Goat		
Milk Constituent	1^{st}	2^{nd}	3^{rd}	1^{st}	2^{nd}	3^{rd}
Milk Fat	1.6	7.6	88.0	2.4	9.0	78.9
Casein	1.5	10.2	82.0	1.5	7.9	133.0
Lactose	0 7	5.1	34.1	–	–	–
Milk H_2O	5.0	67.7	–	6.2	233.1	–
Whole Milk	1.7	5.5	68.0	3.2	7.0	188.0

Table 2: Biological half-life values of different components in some milk constituents of cow and goat after daily administration of OBT.

One of the goats was sacrificed one year after receiving hay containing organically bound tritium. Milk was sampled for about 30 days. The ^3H content was determined in organic milk constituents, in milk water and also in several depots of body fat. The tritium activity per gram of body fat from omentum and from adipose tissue around the kidney was significantly different, and it was 5-10 times higher than that in milk fat.

BIOKINETICS AND DIAPLACENTAL TRANSFER OF J-131 IN RATS FED WITH EITHER STANDARD DIET OR IODINE DEFICIENT DIET

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INTRODUCTION

It was the aim of these studies to investigate the diaplacental transfer and distribution of iodine in the maternal and fetal organs of rats at different days of gestation, because there exist only limited quantitative information. The exact biokinetic data will serve for estimation of the hazard to the fetus from the release of these radionuclides by nuclear energy use and from its application in medicine (1,2,3,4). In particular we wanted to find out if there is a difference in uptake of J-131 into the maternal and fetal thyroid of female rats normally fed and rats fed with an iodine deficient diet (5).

METHODS

On day 17, 18, 19 and 20 of gestation 0.74 MBq of J-131 were injected intravenously into female rats. For the iodine deficient group the maternal animals were put on a diet a few days after birth that contains 41 ng iodine per gram food (standard diet 320 ng/g). The pregnant animals were sacrificed in groups of 5 animals each 0.5, 1, 2, 4, 8, 24 hours after administration of the radionuclide. From each female rat 5 fetuses were assayed for radioactivity in toto and additional 5 were dissected. From the dams and their fetuses blood and amniotic fluid were obtained and thyroid, stomach, liver and kidneys were dissected. After weighting, the radioactivity in the tissues was determined in a scintillation well-counter and the activity concentration was expressed in percent of the injected radioactivity per gram tissue.

RESULTS

Standard diet:

The fetal thyroid gland can first be dissected on day 17 of gestation. On this day its J-131 activity concentration is only 0.14% of the administered activity into the mothers blood circulation per gram tissue 8 hr after injection and is ever lower than the concentration in the fetal stomach. In the time intervall between 8 and 24 hr post injection the uptake of J-131 into the fetal thyroid increases by a factor of 4. This results are in agreement with autoradiographic studies that have demonstrated the starting function of the thyroid in tats on day 17 of gestation (6). The J-131 accumulation by the fetal gland raises very rapidly between day 17 and 20 of pregnancy (from 0.14%/g to 193%/g). The highest increase of activity concentration is found between day 17 and 18 of gestation and becomes less during the following days (7). Only on day 20 the J-131 concentration in the fetal thyroid gland equals the value of the analoqous maternal organ (193%/g and 198%/g). The J-131 activity of the fetal thyroid is only 0.44% of the whole fetal body radioactivity on day 17 raising to 35% until day 19. This value

decreases between day 19 and 20, as the fetal thyroid remains fairly constant in weight whereas the whole fetal body gains about 60% on weight.

But not only the fetal thyroid accumulates iodine the fetal stomach too shows a remarkable uptake of the nuclide (7). The activity concentration in the fetal stomach is higher than the concentration in the fetal blood during the whole investigation period (day 17-20).

Table 1: J-131-Activity Concentration in the Maternal Thyroid 2 and 8 hr p.i. (% inj. activity/g, mean ± S.D. from n=10)

Day of Gestation	2 hr p.i.		8 hr p.i.	
	standard diet	iodine deficient diet	standard diet	iodine deficient diet
17	133 ± 12	1136 ± 44	400 ± 26	1971 ± 132
18	193 ± 18	793 ± 71	436 ± 31	1690 ± 181
19	251 ± 27	854 ± 93	471 ± 28	1489 ± 162
20	105 ± 8	808 ± 67	198 ± 17	1476 ± 139

Iodine deficient diet:

A comparison between the J-131 concentration in maternal and fetal organs of the iodine deficient animal group and the organs of animals fed with standard diet shows significant differences. The activity concentration in the maternal thyroid of animals with iodine deficiency is much higher and the uptake is more rapid than into the thyroid gland of normally fed maternal animals (Fig. 1).

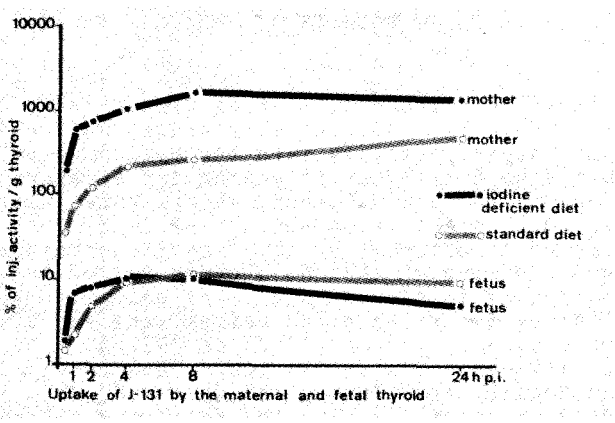

Fig. 1: Uptake of J-131 into the maternal and fetal thyroid of animals fed with iodine deficient diet (black circles) or with standard diet (white circles) on day 18 of gestation.

The results in Tab. 1 indicate that on all days of gestation investigated the J-131 accumulation by the maternal thyroid of the iodine deficient animals is increased at least 3 times compared with the animals fed with standard diet, the activity concentration in blood and all other maternal organs however beeing less.

For the fetal thyroid the accumulation of iodine after iodine deficient diet of the mothers is more rapid too, resulting in a higher activity concentration 2 hours after administration than in fetal thyroids of animals fed with standard diet. The concentrations 8 hours post injection however remain always lower in the fetal thyroids of the iodine deficient group (Fig. 1). Blood and all other fetal tissues of this animal group show a lower activity concentration 2 and 8 hours post injection. These results are represented in Fig. 2. The 2 upper graphs show the J-131 activity concentration in the whole fetal body for the two different fed animal groups on different days of gestation 2 and 8 hours post injection. The whole body activity concentration for the iodine deficient group (black bars) is always less than for the animal group after standard diet (white bars) 8 hours p.i..
The two lower graphs reflect the activity concentrations in the fetal thyroid gland. They show the higher 2-hr values for the iodine deficient group as well as the resulting lower concentrations for this animal group 8 hours after administration of the radionuclide.

The high J-131 uptake by the maternal thyroid after iodine deficient diet results in a reduced maternal blood level of iodine, leading to a decreased diaplacental transfer of the nuclide and therefore to a lower activity accumulation in the fetal thyroid.

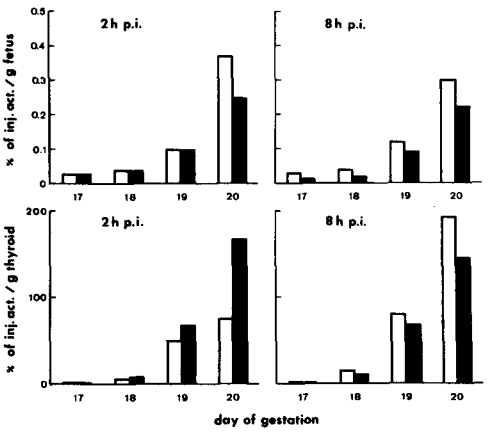

Fig. 2: Activity concentration (% inj. activity/g) in the whole fetus (upper graphs) and in the fetal thyroid (lower graphs) 2 and 8 hr p.i. for animals fed with iodine deficient diet (black bars) or standard diet (white bars)

CONCLUSION

The activity concentration of J-131 in the fetal thyroid is highly dependent on the stage of gestation. On day 17 the fetal thyroid starts its function with a slight accumulation of J-131, that rises by a factor of 1380 until day 20. In the fetal stomach an accumulation of J-131 could be determined too.

A comparison between the J-131 concentration in the maternal and fetal thyroid of iodine deficient females with that of animals fed with a standard diet showed a much higher and a much more rapid uptake of the nuclide into the thyroids of the iodine deficient dams. The uptake in the thyroid of their fetuses is also more rapid resulting in higher 2-hr values, the 8-and 24-hr activity concentrations however remain less than in the fetal thyroid of normally fed animals on all days of gestation.

The high J-131 uptake into the maternal thyroid of iodine deficient dams leads to a decreased diaplacental transfer of the nuclide. The resulting lower activity concentration in the fetal gland means a reduction of the absorbed dose for the fetal thyroid.

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REFERENCES

1. Book, St.A., M. Goldmann. Thyroidal Radionuclide Exposure of the Fetus. Health Phys. 29, 874-877 (1975)
2. Dyer, N.C., and A.B. Brill. Maternal-fetal Transport of Iron and Jodine in Human Subjects. Adv. Exper. Med. Biol. 27, 351-366 (1972)
3. Evans, T.C., R.M. Kretzschmar, R.E. Hodges, and Ch.W. Song (1967) Radioiodine Uptake Studies of the Human Fetal Thyroid. J. Nucl. Med. 8, 157-165 (1967)
4. Exss, R., and B. Graewe. Congenital Athyroidism in the Newborn Infant from Intra-Uterine Radioiodine Action. Biol. Neonate 24, 289-291, (1974)
5. Fish, W.A., H. Carlin, F.C. Hickey. The Thyroidal Uptake, Retention and Iodine Pool in Rats under-Controlled Iodine Diet. Endocrinology 51, 282-292, (1952)
6. Gray, B., and V.A. Galton (1974). The Transplacental Passage of Thyroxine and Foetal Thyroid Function in the Rat. Acta Endocrinologica 75, 725-733
7. Senekowitsch, R., B.U. Kaczmar, H. Kriegel, S. Möllenstädt. Diaplacental Transfer and Distribution of J-131 in Fetal Organs at Different Stages of Gestation. Nuclear Medicine and Biology; Proceedings of 3rd World Congress of Nuclear Medicine and Biology. Pergamon Press, Paris, 2685-2688, (1982)

EQUATION FOR THE URINARY EXCRETION OF AMERICIUM FOLLOWING A SINGLE UPTAKE

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INTRODUCTION

The equation for the urinary excretion of a radionuclide following a single uptake into the body fluid is useful not only for assessing intake of the radionuclide after its accidental entry, but also for determining the Derived Investigation Level (DIL) of the radionuclide in the routine individual monitoring. For americium there is not an established equation such as the one for plutonium given by Langham. In the present study, the equation for the urinary excretion of americium following a single uptake was derived from the various reported data on urinary excretion of intravenously injected americium (citrate) in various animals. Validity of the equation derived was tested by means of the data on urinary excretion of Am-241 observed in a male following its accidental inhalation in JAERI. The observed urinary excretion of Am-241 in the male agreed fairly well with the calculated ones obtained under the assumption that the Am-241 deposited in the lungs was retained there as a class W material (1) and the Am-241 absorbed from the lungs was excreted in the urine following the derived equation. The urinary excretion of Am-241 calculated under the assumption that the Am-241 absorbed from the lungs was excreted in the urine following the equation derived from the metabolic model given in ICRP Publ. 30 (1) (Fu=0.3) deviated markedly from the observed ones.

DERIVATION OF THE EQUATION FOR THE URINARY EXCRETION OF AMERICIUM FOLLOWING A SINGLE UPTAKE FROM THE ANIMAL DATA

The experimental data on the urinary excretion of americium following its intravenous injection as citrate to various mammals were gathered from the various reference sources (2,3,4,5,6), and shown in Fig.1. Difference between the data of monkey and those of beagle is fairly large. However, it is surprising that the data of a small rodent, rat, are intermediate between those of monkey and beagle. To derive the desired equation, all of the data gathered were used. The derived equation was given as follows (see also the curve in Fig.1) :

$$Yu(t)=0.10\exp(-2.0t)+0.0030\exp(-0.26t)+0.0013\exp(-0.025t)$$

where t is days after injection, and Yu(t) is the fractional daily urinary excretion of americium t days after an intravenous injection of unit amount of americium citrate to the animals.

It was tested in the following sections whether the equation derived above was valid or not as an equation for expressing the human urinary excretion of americium following a single uptake.

DATA ON THE URINARY EXCRETION OF Am-241 IN MAN OBSERVED IN ITS ACCIDENTAL INHALATION IN JAERI

In 1977, a healthy male (34 yr old, 66 kg) inhaled mists of the mixture solution of Pm-147 and Am-241 chloride, when opening the bottle containing the old solution of about 40 mCi Pm-147 chloride and about 1 mCi Am-241 chloride. Vomiting of the mists from the bottle was considered to have occurred by pressure of the degradation gas produced in the bottle. Measurement of the amount of Am-241 in the lungs with a chest counter having a NaI (Tl) detector of 24 cm diameter by 1.3 cm thick and determination of the amount of Am-241 in the urine and feces by chemical analysis were made for about half a year following the inhalation. The results of chest counting showed that the fraction of deposited Am-241 remaining long in the lungs (= long-remaining initial lung burden) was 0.57 nCi, and the results of chemical analysis showed that the initial fecal excretion of Am-241 in the first 5 days after inhalation was 2.0 nCi. The removal half-time of the long-remaining initial lung burden from the lungs was longer than that of a class W material in the dosimetric model of ICRP (1). However,this seems to have been caused by the inclusion of the counts in other organs such as liver and skeleton. In the present study, as described below, the Am-241 in the lungs, the chemical form of which was chloride, was assumed to have behaved there as a class W material. The data on the urinary excretion of Am-241 observed in the accident are shown, as open circles, in Fig.2. The method of chemical analysis used for determination of the amount of Am-241 in excreta appeared elsewhere (7).

TEST OF THE VALIDITY OF THE DERIVED EQUATION FOR MAN BY THE HUMAN DATA OBSERVED IN JAERI

If it is assumed that the Am-241 inhaled in the accident described in the third section was deposited in and cleared from the lungs following the dosimetric model for the respiratory system by ICRP (1), in which all of the americium compounds are assigned to inhalation class W, the intake of Am-241 can be assessed from either the initial fecal excretion or the long-remaining initial lung burden. However, this is only when the AMAD of the aerosol is known. In the present accident, measurement of the AMAD of the mist was not made. Therefore, the intake of the Am-241 was estimated by the use of both the initial fecal excretion and the long-remaining initial lung burden, using the prerequisite that the estimates from both the values are the same. This procedure gave the intake of 4.3 nCi of the Am-241 aerosol having the AMAD of 1.3 μm. These values were used in the following test as the intake and AMAD of the aerosol inhaled.

In order to test validity of the equation derived in the second section for man, the urinary excretion of the Am-241 inhaled was calculated assuming that the Am-241 transferred to the body fluid from the lungs was excreted in the urine following that equation. The results are shown in Fig.2 as the curve A. Although the values thus calculated are somewhat smaller than the observed ones, these values agree fairly well with each other. The curve B in Fig.2 shows the values calculated under the assumption that the Am-241 absorbed from the lungs was excreted in the urine following the equation derived from the metabolic model given in ICRP Publ.30 (1), where Fu was assumed as 0.3. It is obvious from the figure that these values deviate markedly from the observed ones. It is thus seen that the equation derived here from the animal data is satisfactorily applicable to the human case, apparently better than that derived from

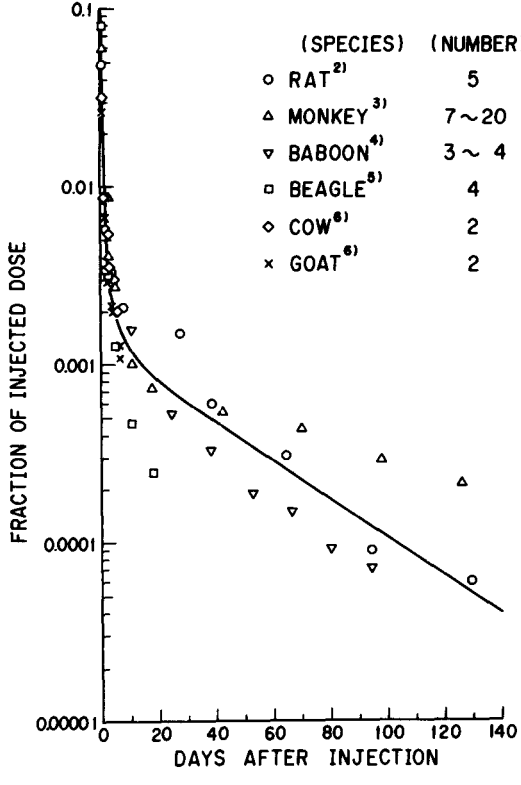

Fig.1. Daily urinary excretion of Am-241 in various animals following an intravenous injection of Am-241 citrate.

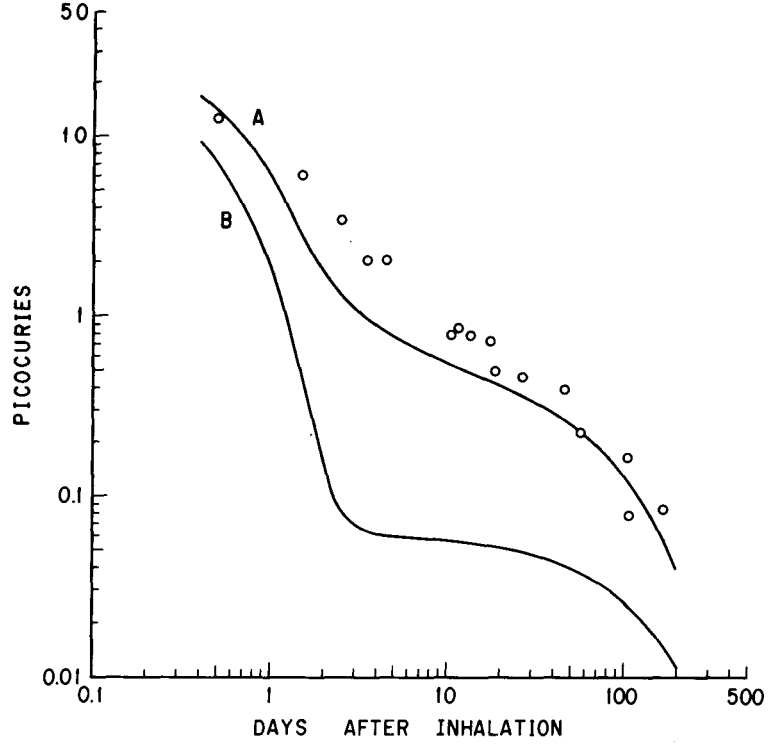

Fig.2. Daily urinary excretion of Am-241 in man following an inhalation of Am-241 chloride. For the curves A and B, see the text.

the metabolic model given in ICRP Publ.30 (1). The somewhat larger observed values than the calculated ones might have occurred owing to the larger absorption of Am-241 than that of the class W material, because the chemical form of the Am-241 inhaled was soluble americium chloride. Zalikin et al. reported (8) the large absorption of Am-241, administered intraperitoneally as chloride, from the rat lungs.

Parkinson et al. derived (9) the equation for the urinary excretion of curium, a chemically analogous element to americium, following a single uptake, from the data on the urinary excretion of curium absorbed from the wound site in man. Validity of the present equation for man might also be supported by the fact that the values calculated from their equation for curium are similar to the present ones in the tested period.

CONCLUSION

The results above show that the equation for the urinary excretion of americium derived here from the animal data can satisfactorily simulate the urinary excretion of americium in man.

However, it must be kept in mind that the present test was carried out under many presumptions. Namely, the amount of intake and the rate of absorption from the lungs of inhaled Am-241 were obtained assuming the model which is inherently given for derivation of the annual limit on intake or the derived air concentration of radionuclides. It is hoped that further test to confirm the validity of the equation derived here for man will be performed.

REFERENCES

1) International Commission on Radiological Protection: "ICRP Publication 30, Part 1, Limits for Intakes of Radionuclides by Workers", Pergamon Press, Oxford(1979).
2) D.M. Taylor, F.D. Sowby and N.F. Kember: Phys. Med. Biol. 6, 73 (1961).
3) P.W. Durbin: "Handbook of Experimental Pharmacology, vol.36: Uranium·Plutonium·Transplutonic Elements" ed. H.C. Hodge et al., Springer-Verlag, Berlin, 739(1972).
4) N. Cohen and M.E. Wrenn: Radiat. Res. 55, 129(1973).
5) R.D. Lloyd, C.W. Mays, G.N. Taylor and D.R. Atherton: Health Phys. 18, 149(1970).
6) W.W. Sutton, R.G. Patzer, A.A. Mullen, P.B. Hahn and G.D. Potter: NVO-192 (Vol.1), 19(1978).
7) Manual of Individual Monitoring for Internal Exposure, JAERI-memo 2198, 23(1966)(in Japanese).
8) G.A. Zalikin, Yu.I. Moskalev and I.K. Petrovich: Radiobiologiya 8, 65(1968), AEC-tr-6950, 107.
9) W.W. Parkinson Jr., L.C. Henley and C.W. Nestor Jr.: Health Phys. 39, 977(1980).

EFFECT OF AGE ON THE METABOLISM OF SOME IMPORTANT RADIONUCLIDES IN THE RAT

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INTRODUCTION

Most of the information upon which the ICRP values concerning radioactivity contamination such as ALI and MPC are based is derived from observations or experiments in which high concentration of radionuclide was administered to adults in a single dose. In applying these values to the general public, we consider that problems involved in wide distribution of age as well as the diversity of chemical form and concentration of radionuclide is very important, since the general public consists of people of wide range of age and they intake radionuclide incorporated into food material.

Age is one of the most important factors which influence the metabolism of radionuclide and the consequent internal radiation dose. It is assumed that young growing animals as well as human beings, particularly those in suckling stage, may incorporate a larger amount of various radionuclides than adults because of their rapid growth, active bone accretion process and better intestinal absorption. It is also assumed that young growing animals have a shorter biological half life of radionuclide than adults. Because of this reason, metabolic features of many radioculcides have been reported in relation to age of the animal, but quantitative information is still not adequately provided. In the present study, the whole-body retention after oral administration was observed in rats of various ages for twelve radionuclides important from radiotoxicological point of view. And then, on the basis of these observations, the age dependency of the committed radiation dose after ingestion of radionuclides has been discussed.

MATERIALS AND METHODS

Wistar strain rats bred and supplied by our institute were used as the experimental animal. The rats were fed standard diet (Funabashi Farm, Japan) and water ad libitum and were kept in an animal room controlled at 21 C and an alternating 12 hr light-dark cycle. Litter size was adjusted to contain 10 pups on the day of parturition and they were nursed by their mother on wood shavings in an aluminum box. Sucklings were weaned at 21 days of age and they as well as adult rats were housed as a group of 5 animals in a wire mesh bottomed cage.

Radionuclide used in the experiment are shown in Table 1. Radionuclides solution was diluted to a proper concentration with distilled water and orally administered to rats as a single dose via a stomach tube. The rats were not fasted for the administration. Immediately after administration the initial body activity was determined by in vivo counting with an Armac counter and whole-body retention was assayed periodically thereafter. To quantitate

Table 1. Radionuclides used in the experiment

Nuclide	Chemical form	Specific activity	Purchase
Cr-51	Na_2CrO_4	102 mCi/mg Cr	NEN
Mn-54	$MnCl_2$	Carrier free	NEN
Fe-59	$FeCl_3$	5 mCi/mg Fe	NEN
Co-60	$CoCl_2$	160 mCi/mg Co	RCC
Zn-65	$ZnCl_2$	6 mCi/mg Zn	NEN
Sr-85	$SrCl_2$	0.3 mCi/mg Sr	NEN
Ru-106	$RuCl_2$	Carrier free	RCC
Ag-110m	$AgNO_3$	5 mCi/mg Ag	NEN
Cd-115m	$CdCl_2$	0.5 mCi/mg Cd	RCC
Sb-125	Sb chloride	Carrier free	RCC
Ce-144	$Ce(NO_3)_3$	0.2 mCi/mg Ce	NEN
Hg-203	$HgCl_3$	1 mCi/mg Hg	NEN

contamination of the experimental animals with radionuclide arising from contact with excreta and through the dam's milk, some animals in each cage were not given radionuclide and remained with dosed animals. To correct for cross contamination, counts of these rats not given radionuclide were subtracted from those of dosed rats.

RESULTS AND DISCUSSION

Whole-body retention of various radionuclides after oral dosing to 5 day-old suckling and 150 day-old adult rats is shown in Fig. 1. The retention pattern in sucklings showed a great difference from that in adults for all radionuclides studied. From the difference of retention pattern between sucklings and adults, the radionuclides can be divided into three groups, although the grouping is not very clear cut.

Radionuclides of manganese, iron, zinc and strontium belong to first group. The radionuclides orally given to sucklings decrease very slowly and linearly on the semilogarithmical paper. This fact suggests that sucklings absorb nearly 100 % of orally dosed these nuclides and the endogenous excretion rate is very low. On the other hand, radionuclide given to adult rats decreased very rapidly during first few days after administration followed with very slow decrease. Fraction rapidly excreted represents the non-absorbed fraction of radionuclide from gastrointestinal tract.

Radionuclide of chromium, ruthenium, silver and cerium belong to second group. After oral administration to adult rats almost all of these radionuclides were excreted in feces very rapidly. This fact suggests that intestinal absorption is very low. Sucklings showed a very unique retention pattern, i.e., intestinal hold up of radionuclide. Whole-body retention immediately after oral dosing to suckling rats was very high and the excretion rate was small. As sucklings approached weaning the decreasing rate of whole-body retention of these radionuclides became more abrupt, and after weaning the retention achieved a plateaus and the excretion rate came to very low again.

Radionuclides of cobalt, cadmium, antimony and mercury belong to third group. Whole-body retention of these nuclides in sucklings

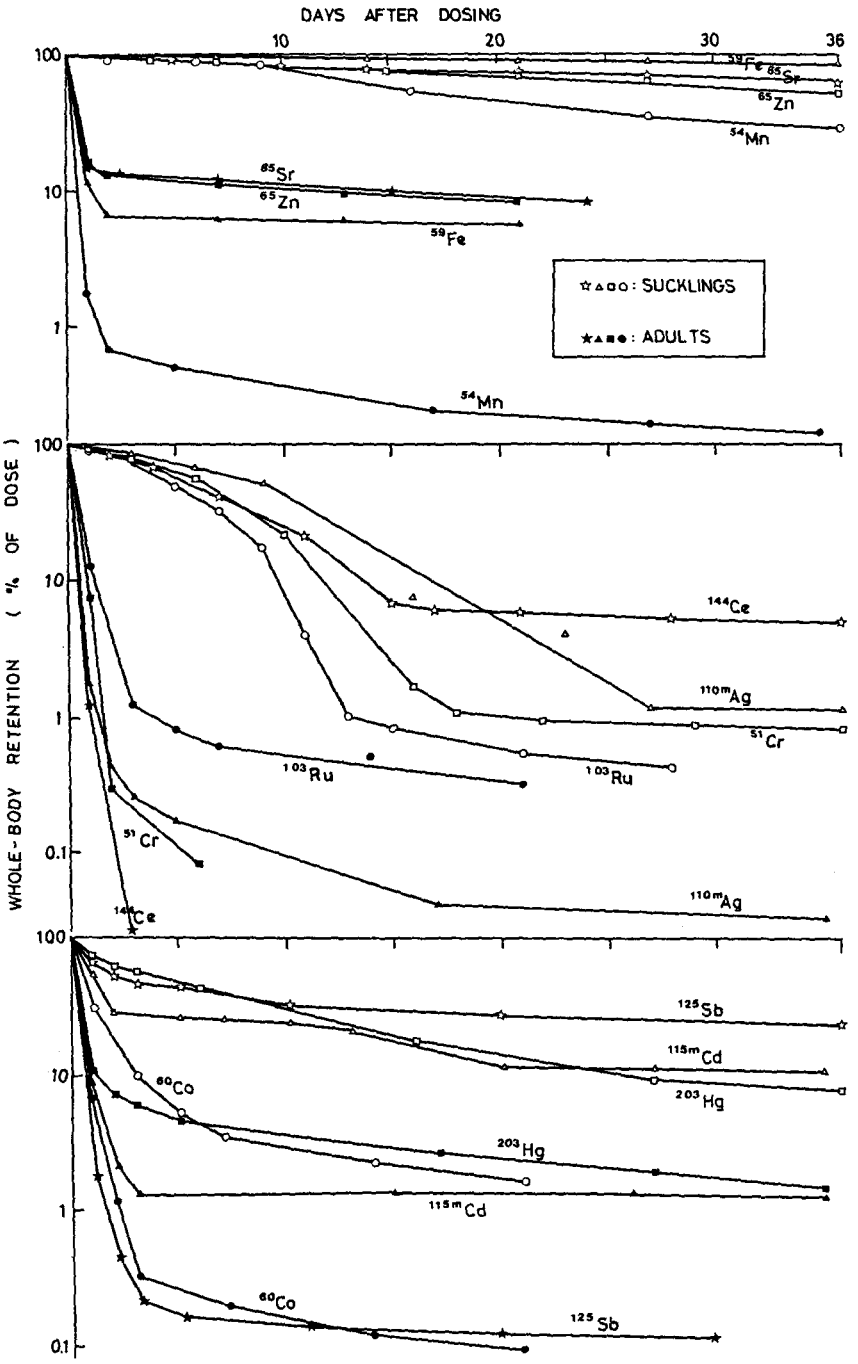

Fig. 1. Whole-body retention of radionuclides after oral dosing

Table 2. Whole-body retention of radionuclides after oral administration to suckling and adult rats

Nuclide	Whole-body retention (% of dose)* Sucklings		Adults	ICRP Publ.30 f_1 (% of dose)
Cr-51	70	(1)	0.08	10
Mn-54	70	(50)	0.5	10
Fe-59	100	(100)	6	10
Co-60	5	(2)	0.2	5
Zn-65	80	(70)	10	50
Sr-85	80	(70)	10	30
Ru-103	80	(0.8)	0.5	5
Ag-110m	80	(5)	0.2	5
Cd-115m	25	(10)	1	5
Sb-125	40	(30)	0.2	1
Ce-144	70	(2)	0.01	0.03
Hg-203	50	(10)	4	2

* : on fifth day after dosing.
Values in the parenthesis are whole-body retention on 20th day after dosing.

is rather similar to that in adults, although initial decrease which represents non-absorbed fraction is much larger for adults compared with that for sucklings. Intestinal hold up seen in the second group can not be observed in the third group.

From the result of the present experiment, whole-body retention of radionuclides on 5th day after dosing for suckling and adult rats is summarized in Table 2. In case of adult rats the value of whole-body retention on 5th day after dosing can be considered roughly equal to the f_1 value. But in case of sucklings, the intestinal hold up was observed for some radionuclides, and in order to avoid this hold up whole-body retention on 20th day after dosing, namely at the time when retention curve reached a so called plateau, is shown in the parenthesis in Table 2. From physiological point of view, values in the parenthesis can be considered more similar to the f_1 value for suckling. These values reveal that the whole-body retention is much higher for sucklings than adults. Moreover, ICRP f_1 value can be considered to be rather conservative for adults but not conservative for sucklings, although difficulty to compare the present value obtained from animal experiment with f_1 value determined for the human directly remains a important problem to be solved.

It can be concluded from the result of the present experiment that intestinal absorption and body retention of radionuclides in younger animals especially sucklings are very high compared with that in adult animals. This fact should be taken into account in the assessment of internal dose of public especially newborn and sucklings due to these radionuclides. As to the gastrointestinal absorption of radionuclide in neonate, Harrison has suggested an increased uptake value for actinides (Radiol. Pro. Bull., No.41, 1981). We would like to propose that Harrison's idea should be extended to other elements which were considered to be very poorly absorbable elements such as cerium.

AGE DEPENDENT METABOLISM AND DOSIMETRY OF BONE SEEKING RADIONUCLIDES

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Age dependent metabolic and dosimetric models and parameters are required to calculate doses from intakes of radionuclides that occur in the environment, either naturally or from man's past, present or future activities. One class of radionuclides that require special attention in age dependent models are the so-called bone seekers, as they tend to be retained in bone for periods of time long compared to the time it takes for bone metabolism and bone structure to change significantly.

The ICRP has recently published (1) a review of recommended metabolic and dosimetric models and parameters for use in internal dosimetry calculations for occupational exposures. That review recommended the metabolic model developed by a Task Group of ICRP (2) for use with alkaline earth elements and used the concepts of bone compartmentalization from this model for doing dosimetry calculations for bone seeking radionuclides other than the alkaline earths (about one half of the elements). The compartmentalization considers the bone to be composed of three source organs (bone surfaces, cortical bone and cancellous bone) and two target organs (bone surfaces and red bone marrow). This poster describes work being done at the Chalk River Nuclear Laboratories in extending the alkaline earth model of bone metabolism to other bone seeking radionuclides, and in extrapolating this model to include ages from birth to adulthood, so that the same dosimetric model can be used for all bone seeking radionuclides and for all age groups.

The alkaline earth metabolic model (2) has been reformulated by Johnson and Myers (3) to make it a first order compartment model (see Figure 1) for which solutions can be easily obtained for any boundary (exposure) conditions by numerical methods. In addition, progress on extending this model to include the effects of age on alkaline earth model parameters and on dosimetry calculations have been reported (4) and Table 1 gives the current estimates of the integrated activities in the bone source organs, and committed doses (5, 6) to the bone target organs for the one year old infant and for the adult (Reference Man) for the important alkaline earth radionuclides. These calculations are performed by first deriving age dependent functions for the rate constants (λ_i, α_i , β_i and σ_i of Fig. 1) for each alkaline earth element (as well as age dependent lung and GI tract models) and then using these with age dependent factors ($S_{ij}(A)$, or S factors) that give the dose to the target organ (j) per unit of integrated activity in the source organ (i) to calculate age dependent committed doses. These committed doses have been shown (6) to be an adequate approximation to either the average committed dose to end of life from a single intake or the average equilibrium dose rate from chronic intake for population exposures. More exact calculations can also be done. That is, the results given in Table 1 are obtained from calculations of the form

$$D50_j = \sum_i S_{ij}(A_o) \int_0^{50\ y} R_i(A_o,t)\ dt$$

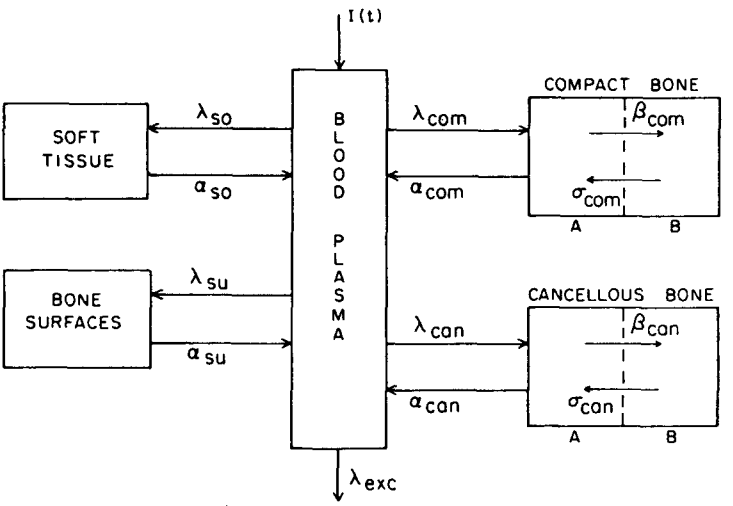

FIGURE 1 First order compartment model used to describe alkaline earth metabolism and bone remodelling in man.

where $S_{ij}(A_o)$ is the S factor for the age (A_o) of intake, and $R_o(A_o,t)$ is the retention function for the age at intake as a function of time after a pulse intake. This formulation assumes that bone metabolism and structure is constant and equal to that at the age of intake, which overestimates the committed dose to the one year old infant. A more exact expression for the dose to an age A from any intake pattern is

$$D_j(A) = \int_0^A \dot{D}_j(a)\ da$$

$\dot{D}_j(a)$ is the dose rate to target organ j at age a, given by

$$\dot{D}_j(a) = \sum_i S_{ij}(a) \int_o^a I(a')\ R_i(a',\ a-a')\ da'$$

where I(a') is the intake rate at age a' and $R_i(a',a-a')$ is the retention in source organ i at age a' and time a-a' since a pulse intake. Space limitations do not allow the inclusion of results obtained with these equations in this paper, but they will be presented in the poster.

Solutions to the above equations can easily be obtained by numerical methods once functions for $S_{ij}(a)$ and R(a,t) are

TABLE 1 The Q50's are the integrated activities on bone surfaces (BS) and in cancellous (Can) and cortical (Cor) mineral bone. The D50's are the committed dose equivalents to the target organs, bone surfaces (BS) and red bone marrow (RBM). Class refers to the ICRP inhalation class (D or W) or ingestion (I). F1 is the fraction of activity that enters the GI tract that is absorbed in blood. Values in parentheses are powers of 10 (i.e. (-9) = x 10^{-9}).

Isotope	Class	F1	INFANT					ADULT				
			Q50(d)			D50 (Sv Bq^{-1})		Q50(d)			D50 (Sv Bq^{-1})	
			BS	Can	Cor	BS	RBM	BS	Can	Cor	BS	RBM
Calcium-41	W	0.3	2.9	8.8(1)	9.0(1)	9.8(-9)	5.8(-9)	2.7	1.5(2)	4.5(2)	3.1(-9)	1.2(-9)
	I	0.3	3.9	1.20(2)	1.20(2)	1.3(-8)	7.9(-9)	3.7	2.01(2)	6.2(2)	4.2(-9)	1.7(-9)
Strontium-85	D	0.3	4.2(-1)	9.8	8.5	1.0(-8)	8.0(-9)	6.4(-1)	2.7	4.0	9.4(-10)	2.3(-10)
	I	0.3	2.9(-1)	6.7	5.8	6.8(-9)	5.5(-9)	4.4(-1)	1.9	2.8	6.6(-10)	5.7(-10)
Strontium-89	D	0.3	4.0(-1)	8.1	7.2	1.8(-7)	1.3(-1)	6.3(-1)	2.2	3.3	8.0(-9)	5.1(-9)
	I	0.3	2.8(-1)	5.6	4.9	1.2(-7)	9.1(-8)	4.3(-1)	1.5	2.3	5.4(-9)	3.5(-9)
Strontium-90	D	0.3	8.3(-1)	7.0(1)	4.8(1)	2.7(-6)	2.2(-6)	8.1(-1)	6.5(1)	1.6(2)	5.4(-7)	2.4(-7)
	I	0.3	5.6(-1)	4.7(1)	3.3(1)	1.9(-6)	1.5(-6)	5.5(-1)	4.4(1)	1.1(2)	3.7(-7)	1.6(-7)
Barium-140	D	0.1	1.0(-1)	4.4(-1)	4.9(-1)	2.0(-8)	1.3(-8)	1.1(-1)	1.7(-1)	2.5(-1)	1.5(-9)	8.7(-10)
	I	0.1	2.5(-2)	1.0(-1)	1.2(-1)	4.8(-9)	3.1(-9)	2.7(-2)	4.0(-2)	5.9(-2)	4.0(-10)	3.8(-10)
Radium-223	W	0.2	1.0(-1)	9.9(-2)	1.1(-1)	1.5(-5)	1.3(-6)	1.0(-1)	1.4(-2)	2.5(-2)	1.6(-6)	1.3(-7)
	I	0.2	1.7(-1)	1.6(-1)	1.8(-1)	2.4(-5)	2.0(-6)	1.8(-1)	2.6(-2)	4.5(-2)	2.8(-6)	2.3(-7)
Radium-224	W	0.2	5.4(-2)	2.5(-2)	2.8(-2)	7.4(-6)	6.1(-7)	5.4(-2)	3.6(-3)	6.3(-3)	8.3(-7)	6.7(-8)
	I	0.2	1.0(-1)	4.7(-2)	5.3(-2)	1.4(-5)	1.1(-6)	1.1(-1)	7.1(-3)	1.3(-2)	1.7(-6)	1.3(-7)
Radium-225	W	0.2	1.1(-1)	1.3(-1)	1.5(-1)	9.8(-6)	8.7(-7)	1.1(-1)	2.0(-2)	3.4(-2)	8.8(-7)	7.1(-8)
	I	0.2	1.8(-1)	2.1(-1)	2.3(-1)	1.2(-5)	1.1(-6)	2.0(-1)	3.4(-2)	5.9(-2)	1.1(-6)	9.1(-8)
Radium-226	W	0.2	2.4(-1)	2.4	2.5	2.3(-5)	2.5(-6)	2.4(-1)	4.7	1.1(1)	6.3(-6)	5.2(-7)
	I	0.2	2.6(-1)	2.6	2.8	2.5(-3)	2.7(-6)	2.6(-1)	5.1	1.2(1)	6.8(-6)	5.7(-7)
Radium-228	W	0.2	2.4(-1)	2.2	2.3	1.2(-5)	1.3(-6)	2.4(-1)	2.2	3.1	5.1(-6)	5.1(-7)
	I	0.2	2.6(-1)	2.4	2.5	1.1(-5)	1.2(-6)	2.6(-1)	2.4	3.4	5.4(-6)	5.3(-7)

available, and I(a) has been specified. The $S_{ij}(a)$ can be calculated from knowledge of the change in the bone mass and change in the fraction of energy deposited with changes in source-target geometry with age. In this work, it is assumed (following work by Spiers (7) and Snyder (8)) that the fraction of energy deposited in the target organs varied linearly from 1.2 times the value for adults (1) at birth to 1 times the value for adults at age 20. The mass of the target organs was assumed to be directly proportional to the mass of calcium in bone, as estimated by the age dependent model.

The age dependences of the R(a,t)'s for the alkaline earth elements have been estimated as follows. Calcium in bone as a function of age reported by Mitchell (9) was compared to model calculations for a calcium intake by ingestion taken from ICRP Publication 23 (10) and the model parameters adjusted to give a good fit using FORSIMOPT (11). Next, comparison between the bomb fallout Sr-90 age and time dependent intake and bone content reported by Papworth and Vennart (12) and the model results was used to modify the age dependent parameters derived for calcium to get age dependent strontium parameters. Age dependent radium parameters were estimated from the results reported by Harrison et al. (13) for Ca-47, Sr-85 and Ra-223 in adults, combined with the age dependence derived for calcium and strontium as explained above. Finally, age dependent barium parameters were "questimated" from the results of the calcium, strontium and radium calculations.

The results of the age dependent bone remodelling developed for the alkaline earth is now being used along with other element specific data to derive better bone metabolism models for other elements. Progress to date on these models will be presented at the poster session.

REFERENCES

1. ICRP Publication 30, Pergamon Press, Oxford, 1979.
2. ICRP Publication 20, Pergamon Press, Oxford, 1973.
3. J.R. Johnson and R.C. Myers, Rad. Prot. Dos. 1, (2), 87, 1981.
4. J.R. Johnson, Health Physics 44, 91, 1983.
5. J.R. Johnson and D.W. Dunford, Atomic Energy of Canada Limited, Report AECL-7919, 1983.
6. J.R. Johnson, Rad. Prot. Dos. 3, (1/2), 47, 1982.
7. F.W. Spiers, Br. J. Radiology 47, 833, 1974.
8. W.S. Snyder, Lecture given at the 2nd Int. Summer School on Radiation Protection, Yugoslavia, 1973.
9. H.H. Mitchell, T.S. Hamilton, F.R. Steggerda and H.W. Bean, J. Biological Chemistry 158, 625, 1945.
10. ICRP Publication 23, Pergamon Press, Oxford, 1975.
11. M.B. Carver, Math. Comput. Simulation XXII, 298, 1980.
12. D.G. Papworth and J. Vennart, Phys. Med. Biology 18, 7, 1973.
13. G.E. Harrison, T.E. Carr, A. Sutton and J. Rundo, Nature, 209, 526, 1966.

PLUTONIUM AND STRONTIUM-90 IN THE HUMAN BODY AND PARAMETERS IN METABOLIC AND DOSIMETRIC MODELS

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Transfer of radionuclides from the environment to man and estimation of internal dose have been matters of major concern with respect to the effects of fallout to humans released by the atmospheric weapons testing, and is of increasing importance in view of the steadily developing nuclear power production and industrial activities related to nuclear fuel cycles.

In this paper, data on the distribution of plutonium in human tissues and strontium-90 in bone in the Japanese will be reported and discussed with respect to metabolic and dosimetric models, such as ICRP's. Emphasis will be put on the distribution of fallout plutonium in skeleton, or between trabecular and cortical bones, which is needed in extrapolating and analytical data for bone to the skeletal burden, and transfer factor for strontium-90 in the diet to bone in relation to a possible influence of larger content of stable strontium in the Japanese diet on the basis of unit calcium content than that in the Western nations.

Comparison will be made on the "measured" tissue burden and the "estimated" body burden from the environmental data using prediction models. Discussion will be made on problems, if any, to be encountered in the application of some metabolic and dosimetric models to the general public primarily in the authors' country.

INTERSPECIES COMPARISON OF THE METABOLISM AND DOSIMETRY OF INHALED MIXED OXIDES OF PLUTONIUM AND URANIUM

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Three studies were conducted to provide information on the biological fate, distribution of radiation doses among tissues, and implications for potential health consequences of an inhalation exposure involving mixed oxide nuclear fuel materials. Three different materials were studied using the same experimental protocol. In each study, Fischer-344 rats, Beagle dogs and Cynomolgus monkeys inhaled one of three aerosols: 750°C calcined mixed oxides of UO_2 and PuO_2, 1750°C sintered $(U,Pu)O_2$ or 850°C calcined "pure" PuO_2. These materials were collected from glovebox enclosures immediately after normal industrial processing of fuel materials. Lung retention, tissue distribution and mode of excretion of $^{238\text{-}240}Pu$, ^{241}Am and U (when present) were quantified by radiochemical analysis of tissue and excreta samples from animals sacrificed at selected times to 6.5 years after inhalation exposure.

Biomathematical models were formulated to provide good descriptions of each data set. Common rate constants were used, whenever possible, for each element and each species for each aerosol, thus providing a structure within which to interpret data from these studies.

METHODS AND MATERIALS

Inhalation exposure procedures used (1), physical chemical characteristics (2) and *in vitro* dissolution of U, Pu and Am in the three aerosols in several solvents (3) have been described. The lung retention data obtained by serial sacrifice of 4 rats, 2 dogs and 1 monkey each at times ranging from 64 days to 6.5 years after inhalation were fitted using a two-component, sums of negative exponentials function. Similarities and differences among the fitted lung retention functions for a single species exposed to the three different aerosols or for all three species exposed to a single aerosol were tested using a generalized F statistic. This allowed conclusions regarding whether the observed differences in lung retention were due to differences in the aerosol or to a species effect.

The biomathematical models formulated to describe the retention, distribution and excretion of the U (when appropriate), Pu and Am present in each aerosol used Mercer's mathematical expression for description of the process of dissolution of particulates deposited in lung (4). The biomathematical model was adapted from similar models found useful in describing data from similar studies in which dogs inhaled laboratory-produced aerosols of $^{238}PuO_2$ or $^{241}AmO_2$ (5,6).

For modeling the Pu results, the rate constants for all internal organs communicating with the blood compartments were initially set equal to the values used in the model for $^{238}PuO_2$ (5). Similarly, for modeling the ^{241}Am in these aerosols, the rate constants were initially set to the values used in the $^{241}AmO_2$ model (6). The values used in the expression describing dissolution of the particles deposited in lung were measured: geometric diameter and geometric standard deviation were determined by cascade impactor samples obtained during animal exposures; density was determined by x-ray diffraction analysis of the crystal lattice unit cell dimension on particles collected on filters during exposure;

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surface shape factors were determined on small aliquots of the exposure aerosol using an ^{85}Kr adsorption technique (7); and the dissolution constant was determined from *in vitro* dissolution studies that used simulated serum ultrafiltrate as the solvent (3).

The models were implemented using a simulation language (8) programmed in Fortran IV on a PDP VAX 11/780 computer. The simulations were run iteratively and the results plotted to judge conformance with the data set.

Figure 1 is schematic diagram of the biomathematical model with rate constants used in modeling the retention, distribution and excretion of Pu following inhalation of the three aerosols.

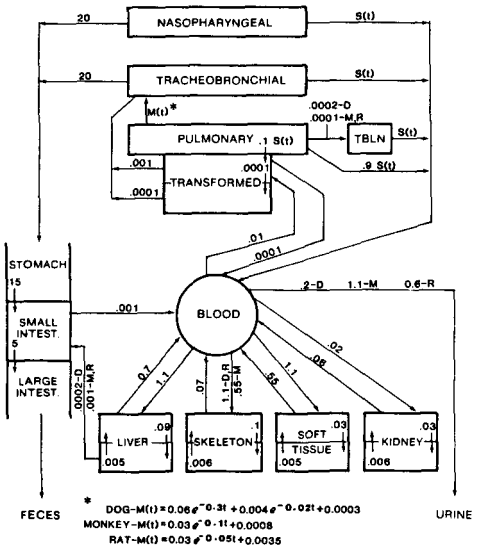

FIGURE 1. Schematic diagram of the biomathematical model used to describe the retention, distribution and excretion of Pu in dogs, monkeys and rats following inhalation of either UO_2+PuO_2, $(U,Pu)O_2$ or "pure" PuO_2. Where more than one rate constant is shown for a particular pathway, the suffix letter D (dog), M (monkey) or R (rat) indicates which constant is associated with each species. The function S(t) refers to the equation of Mercer (4).

RESULTS AND DISCUSSION

The similarity in the retention of Pu in the lung of dogs for each of the three aerosols, together with the model-generated curves (Figure 2) confirm the F statistic analysis that the lung retention of the three materials is not significantly different. Virtually identical results occur when the data and curves for monkeys or rats are similarly compared. The similarity occurs even though the 850°C-treated "pure" PuO_2 aerosol had a measured specific surface area almost 5 times greater than that of the other two aerosols (indicating the relative insolubility of the Pu component of these aerosols).

The role of species in determining lung retention is shown in Figure 3 for the rats, dogs and monkeys exposed to the 1750°C-treated $(U,Pu)O_2$. Lung retention was different for the three species, as determined by the F statistic. The differences among the three species appear to be entirely due to differences in mechanical clearance from lung, as evidenced in the fit of the model to the excreta data for each species. This indicates that dissolution of the particles deposited in lung is essentially constant among species. Predictions of lung retention, tissue distribution and excretion of Pu generated by the model for mixed oxides in dogs agreed with data from a study in which dogs inhaled laboratory-produced monodisperse aerosols of $^{239}PuO_2$ (9).

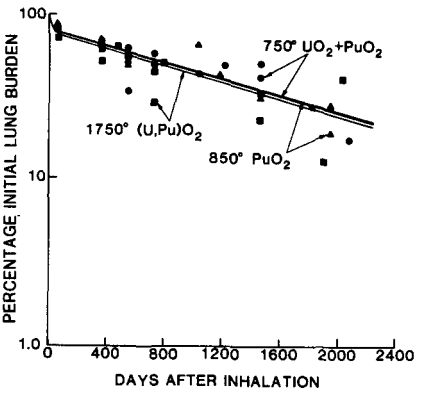

FIGURE 2. Lung retention of Pu in dogs after inhalation of either UO_2+ PuO_2 $(U,Pu)O_2$ or "pure" PuO_2. Data points represent individual animals and the curves (indistinguishable) are the results from the biomathematical model.

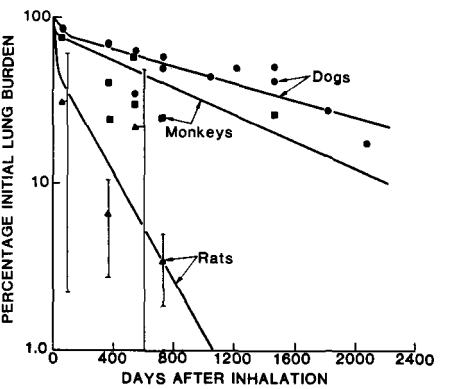

FIGURE 3. Lung retention of Pu in dog, monkey and rat following inhalation of $(U,Pu)O_2$. Data points for dog and monkey represent individual animals; those for rat are the mean ± 1 S.D. (n = 4). Curves are the result of the biomathematical model.

Changes were necessary in the rate constants found adequate for dogs to describe the retention, distribution and excretion of U, Pu and Am in monkeys and rats. These changes reflect differences in the metabolism of these elements in each species (see Figure 1). Using the model for dogs as a base, the rate of transfer for the second liver compartment to the small intestine (biliary route) was increased, as was the rate constant for urinary excretion, to describe the data for rats and monkeys. The rate of transfer of particles from tracheobronchial lymph nodes was decreased for both species compared to the dog. Also, the rate of uptake of material from blood to skeleton was decreased for monkeys, compared to dogs.

Results for the ^{241}Am component of these aerosols were quite similar to those for Pu. This is the result of the intimate association of the ^{241}Am in these aerosols as a decay product of ^{241}Pu. The dissolution rate employed in the model for the Am component was identical to the Pu rate because ^{241}Am is a minor mass component in the PuO_2 particles. *In vivo* dissolution of U from the two aerosols UO_2 + PuO_2 and $(U,Pu)O_2$ was different. This is thought to be due to the differing rate of dissolution that occurs when U is present as UO_2 versus the solid solution $(U,Pu)O_2$. This was measured in the *in vitro* dissolution studies and confirmed for each species in the animal inhalation studies.

Our results indicate that the Pu component in these mixed oxide nuclear fuels can be described adequately using the theory of Mercer (4) to describe the dissolution rate of particles deposited in lung. Physical chemical determinations of specific surface area, density, particle size and size distribution and the rate of dissolution of the particles were used in Mercer's equation to show that, for a single species, the lung retention of the Pu component was not different for the three aerosols studied.

Specific differences in lung retention, tissue distribution and excretion rate were discernible among the three animal species for each aerosol. These differences were attributable to different rates of mechanical clearance, rates of transfer from liver to small intestine, the biliary route and the fraction of the element in the blood compartment excreted rapidly through the kidney to urine.

These differences in retention, distribution and excretion rates among species must be considered before extrapolation of animal data to predict potential consequences of human inhalation exposure.

REFERENCES

1. Stanley, J. A., Eidson, A. F. and Mewhinney, J. A., 1982, *Health Phys.* 43: 521-530.
2. Eidson, A. F., 1982, *Health Phys.* 42: 531-536.
3. Eidson, A. F. and Mewhinney, J. A., 1983, *Health Phys.* 45: (in press).
4. Mercer, T. T., 1967, *Health Phys.* 13: 1211-1221.
5. Mewhinney, J. A. and Diel, J. H., 1983, *Health Phys.* 45: 39-60.
6. Mewhinney, J. A. and Griffith, W. C., 1982, *Health Phys.* 44: (Supl-1), 537-544.
7. Mewhinney, J. A., Rothenberg, S. J., Eidson, A. F. and Newton, G. J., 1983, in Radiation Dose Estimates and Hazard Evaluations for Inhaled Airborne Radionuclides, NUREG/CR-3313, LMF-105.
8. Pritsker, A. A. B., 1974, The GASP IV Simulation Language, Wiley and Sons, New York.
9. Guilmette, R. A., Diel, J. H., Muggenburg, B. A., Mewhinney, J. A., Boecker, B. B. and McClellan, R. O., 1983, *Int. J. Radiat. Biol.* (submitted).

APPLYING THE ICRP RISK FACTORS TO NUCLEAR MEDICINE DOSES

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The largest manmade radiation exposure to the general public results from the medical use of radiation. Nuclear medicine constitutes a significant part of this exposure. During 1983 there were about 7.5 million nuclear medicine administrations of radionuclides in the United States. Table 1 lists the approximate number of each nuclear medicine procedure performed according to the purpose of the procedure and the agent/radionuclide used (1-2). The nuclear medicine radiation dose equivalent to an individual is generally an order of magnitude larger than the dose equivalent received by the average radiation worker. The patient population is also much larger than the occupational worker group. These two factors make the nuclear medicine collective dose equivalent in person-sievert much larger than that for the radiation worker population; hence, for radiation protection purposes the nuclear medicine population should be studied.

Although the 1977 ICRP recommendations on radiation protection (3) were designed to control the radiation exposures of occupational workers and specifically state, "...it is not appropriate to apply the quantitative values of the Commission's recommended dose-equivalent limits to medical exposures," the ICRP concept of risk can be a valuable technique in evaluating the potentially detrimental effects of nuclear medicine procedures. Table 1, column 4, lists the organs that receive the highest dose. The original references should be consulted for more details regarding the dose to other organs. Column 5 gives the weighted equivalent organ dose which is the product of the average administered activity (column 3) and the organ dose per unit administered activity (from original references) and the appropriate weighting factors. A modified version of the ICRP 26 weighting factors was used as suggested by Roedler (4) and Persson (5). These factors can be called the somatic weighting factors because they only relate to the somatic effects. The weighting factors used are: Breast, 0.19; Red marrow, 0.16; Lung, 0.16; Thyroid, 0.04; Bone surface, 0.04; Skin, 0.01; and Remainder, 0.40. Column 6 provides the effective somatic whole-body dose equivalent per procedure.

It is interesting to note that most nuclear medicine procedures result in less than five effective millisieverts which is about the average annual effective dose equivalent for the radiation worker. The annual effective dose equivalent to the nuclear medicine population (column 7) can be obtained by multiplying the effective dose equivalent (column 6) by the number of patients undergoing the procedure each year (column 2).

Some care is required in applying the collective dose-equivalent. Because the population undergoing nuclear medicine procedures is not as healthy as the radiation worker population, they probably do not survive as long as the worker population and many may die before the effects attributable to radiation can manifest themselves. The age distribution of the two populations also differ. Several years ago the U.S. Bureau of Radiological Health reported on the age of the nuclear medicine population (6). Twenty-two percent of the patients were over 65 and these patients are unlikely to survive long enough to develop the conditions caused by radiation. Thus, the collective dose equivalent values could perhaps be reduced by about 25%.

Although the ICRP weighting factor model (3) was not intended for nuclear medicine procedures, the technique does allow for the initial examination of the significance of nuclear medicine to the total population exposure.

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REFERENCES

1) Personal Communication: Office of Training and Assistance, National Center for Devices and Radiological Health, Food and Drug Administration, Department of Health and Human Services, Rockville, MD 20857
2) Rhodes BA, Cordova MA: Adverse Reactions to Radiopharmaceuticals. J Nucl Med 21:1107-1110, 1980
3) International Commission on Radiological Protection: Recommendations of the International Commission on Radiological Protection. ICRP Publication 26, Oxford, Pergammon Press, 1977
4) Roedler HD: Radiation Dose to the Patient in Radionuclide Studies. IAEA-SM-247/206, pp 527-542, 1980
5) Persson BRR: Effective Dose Equivalent Concept in Radiopharmaceutical Dosimetry. In Third International Radiopharmaceutical Dosimetry Symposium. Oak Ridge, TN, 1980, pp 616-624, FDA 81-8166, 1981
6) McIntyre AB, Hamilton DR, Grant RC: A Pilot Study of Nuclear Medicine Reporting Through the Medically Oriented Data System, HEW Publication (FDA) 76-8045, Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402
7) National Council on Radiation Protection and Measurements: Nuclear Medicine - Factors Influencing the Choice and Use of Radionuclides in Diagnosis and Therapy. NCRP Report No. 70, Bethesda, MD, National Council on Radiation Protection and Measurements, 1982
8) Personal Communication: FDA-DOE Radiopharmaceutical Internal Dose Information Center, Manpower Education, Research, and Training Division, Oak Ridge Associated Universities, Oak Ridge, TN 37831-0117
9) Thomas SR, Atkins HL, McAfee JG, et al: Radiation absorbed dose from Tc-99m Diethylenetriaminepentaacetic acid (DTPA), MIRD Dose Estimate No. 12. J Nucl Med: in press
10) Prato FS, Vinitski S: Radiation dose calculations for inhalation of Tc-99m sulfur colloid radioaerosol. J Nucl Med 24:816-821, 1983
11) Brown PH, Krishnamurthy GT, Bobba VR, et al: Radiation-dose calculation for five Tc-99m IDA hepatobiliary agents. J Nucl Med 23:1025-1030, 1982
12) Roedler HD, Kaul A, Hine GJ: Internal Radiation Dose in Diagnostic Nuclear Medicine, Berlin, Verlag H. Hoffmann, 1978

NOTES TO TABLE 1:

* Numbers in parentheses indicate references from which absorbed dose per organ values were obtained.

In column 4, the following abbreviations were used:

Stom. wall - Stomach wall
Blad. wall - Urinary bladder wall
GB wall - Gall bladder wall
ULI wall - Upper large intestine wall
LLI wall - Lower large intestine wall

TABLE 1

PROCEDURE/Agent	Person-Procedures per Yr (10^3)	Average Administered Activity (MBq per Procedure)	Organ(s) Receiving Highest Dose	Weighted Equivalent Organ Dose (mSv per Procedure)	Effective Somatic Whole-Body Dose Equivalent (mSv per Procedure)	Collective Dose Equivalent (Person-Sv per Yr)
BRAIN SCANS						
Tc-99m pertechnetate[7]*	114	650	Stom. Wall	3.5	6.9	790
Tc-99m glucoheptonate[8]	333	555	Blad. Wall	3.4		
			Kidneys	2.1	5.6	1900
Tc-99m DTPA/DTPA(Sn)[9]	242	710	Blad. Wall	4.3	5.0	1200
LUNG PERFUSION						
Tc-99m MAA[7]	732	180	Lungs	1.6		
			Blad. Wall	1.1	2.9	2100
Tc-99m HAM[7]	79	155	Lungs	1.8	2.4	190
LUNG VENTILATION						
Xe-133[7]	435	670	Lungs	0.32	0.36	160
Kr-81m[7]	27	555	Liver	0.14	0.15	4.1
Tc-99m DTPA aerosol[10]	6	445	Lungs	7.2	12	72
Xe-127[7]	2	185	Lungs	0.038	0.052	0.1
THYROID UPTAKE						
I-131[7]	112	2.22	Thyroid	20	20	2200
I-123[7]	82	12.2	Thyroid	0.96	1.0	82
THYROID SCANS						
Tc-99m pertechnetate[7]	231	50	Stom. Wall	0.27	0.53	120
I-123[7]	158	12.6	Thyroid	1.0	1.1	170
I-131[7]	70	2.18	Thyroid	19	19	1300
BONE SCANS						
Tc-99m MDP[7]	1393	640	Red Marrow	0.91		
			Blad. Wall	0.70	2.0	2800
Tc-99m HDP[7]	413	640	Red Marrow	0.91		
			Blad. Wall	0.70	2.0	830
Tc-99m pyrophosphate[7]	25	640	Red Marrow	0.91		
			Blad. Wall	0.70	2.0	50
Tc-99m diphosphonate[7]	11	640	Red Marrow	0.91		
			Blad. Wall	0.70	2.0	22

TABLE 1 (continued)

PROCEDURE/Agent	Person-Procedures per Yr (10^3)	Average Administered Activity (MBq per Procedure	Organ(s) Receiving Highest Dose	Weighted Equivalent Organ Dose (mSv per Procedure)	Effective Somatic Whole-Body Dose Equivalent (mSv per Procedure)	Collective Dose Equivalent (Person-Sv per Yr)
LIVER/SPLEEN						
Tc-99m sulfur colloid[7]	1344	155	Liver	1.1		
			Spleen	0.70	2.0	2700
HEPATOBILIARY						
Tc-99m DISIDA[11]	213	150	GB Wall	2.6		
			ULI Wall	1.2	5.5	1200
Tc-99m HIDA[11]	11	150	GB Wall	2.2		
			ULI Wall	1.0	4.8	53
KIDNEY						
Tc-99m DTPA/DTPA(Sn)[9]	108	222	Blad. Wall	1.4	1.6	170
Tc-99m glucoheptonate[8]	91	555	Blad. Wall	3.4		
			Kidneys	2.1	5.6	510
I-131 hippurate[7]	61	13	Blad. Wall	3.7	3.8	230
Tc-99m DMSA[8]	6	145	Kidneys	2.0	2.5	15
SOFT TISSUE TUMORS						
Ga-67 citrate[7]	123	110	Red Marrow	2.9		
			LLI Wall	2.1	10	1200
CISTERNOGRAPHY						
In-111 DTPA[12]	9	37	Spinal Cord	9.6	9.6	86
Yb-169 DTPA[12]	6	37	Spinal Cord	12	12	72
HEART AND BLOOD STUDIES						
Tl-201 chloride[7]	343	110	Kidneys	2.8	8.0	2700
Tc-99m pyrophoshate[7]	230	560	Red Marrow	0.78		
			Blad. Wall	0.62	1.7	390
Tc-99m PYP-RBC[8]	281	740	Stom. Wall	1.3	3.8	1100
Tc-99m pertechnetate[7]	108	740	Stom. Wall	4.0	7.8	840
Tc-99m RBC[7]	35	740	Red Marrow	0.80	1.4	49
Tc-99m HSA[7]	21	370	Red Marrow	0.40		
			Blad. Wall	0.26	0.66	14
Tc-99m DTPA[9]	23	740	Blad. Wall	4.5	5.3	120

QUANTIFICATION OF THE RISKS OF MEDICAL EXPOSURE IN X-RAY DIAGNOSTICS AND NUCLEAR MEDICINE

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INTRODUCTION

Medical irradiation of the human body occurs in diagnostic X-ray procedures, diagnostic nuclear medicine by internally administered radionuclides, and in radiation therapy. In many countries, medical exposure gives the largest man-made contribution to the population dose. Quantification of risks from medical irradiation is, however, a controversial and difficult question, which has been discussed a lot during recent years (1-8). The risk can be quantified either by using different weighted dose concepts, or by making estimates of the expected number of fatal tumors induced by ionizing radiation.

RISK ESTIMATES FROM DIAGNOSTIC X-RAY PROCEDURES

Exposures from diagnostic X-ray examinations can cause various stochastic biological effects. Genetic effects will be expressed in the descendants of the patient, and somatic effects in the exposed individual. The risk estimates for medical exposures deal therefore both with the induction of genetic effects and the induction of malignant diseases, which contribute the main somatic effect.

Cancer mortality risks for specific sites have been estimated by the US National Academy of Science's BEIR Committes 1972 and 1980, UNSCEAR 1977, and ICRP in 1977. By using the factors of risk and the average dose equivalent for each organ and tissue, it is possible to estimate the expected total radiation risk involved in each medical diagnostic procedure.

The radiation risk expressed as the number of fatal tumours induced by ionizing radiation in a medical examination of type "j" can be assumed to follow a linear dose-resonse relationship as long as the radiation doses are low (<1 Gy). In this low dose range, the risk of an examination of type "j" can be given by the linear expression:

$$RISK(j) = \sum_{t} \bar{H}(j,t) \cdot b(t)$$

where: $\bar{H}(j,t)$ is the average dose equivalent (Sv) in a tissue, "t" and b(t) is the risk factor for that tissue per 10 000 man-Sv.

A large amount of data for the dose-equivalent in different organs and tissues are required to be able to make accurate risk estimates for diagnostic X-ray examinations. Such data are, however, only seldom available in practice. Only a few extensive studies are available in the literature (6,7,8). There is, however, a good correlation between the risk estimates obtained from summing up the risks for specific organs and the energy imparted "E" (3,4,5).

FIGURE 1 gives the risks for various types of diagnostic x-ray examinations with the limits for males and females corresponding to the expression below. Although there is a rough averaging involved in the estimation of the riskfactors most examinations fall into the area covered by the equations:

$$RISK(male) = 0.23 \cdot E$$

$$RISK(female) = 0.40 \cdot E$$

where: "RISK" is cancer mortality per million of examinations and "E" is the energy imparted per examination expressed in mJ.

RISKS FROM DIAGNOSTIC NUCLEAR MEDICINE

The same procedure of correlating the estimated stochastic risk and the energy imparted in nuclear medicine examinations results in a wider spread of the ratios between risks and the energy imparted. The correlation between risk estimates obtained from summing up specific organ risks and the energy imparted, obtained by multiplying the average whole-body dose with a body weight of 70 kg, is shown in FIGURE 2. This diagram indicates at least two groups of relationships: One relationship is found for I-131 iodide for thyroid examinations, and another relationship is found for all other radiopharmaceuticals, lying well within the same area as the X-ray examinations. This indicates that the energy imparted might also be a useful risk estimator in nuclear medicine.

THE EFFECTIVE DOSE-EQUIVALENT

The concept of effective dose-equivalent is used in order to estimate a total risk which is equal if the whole body is irradiated uniformly, or if there is non-uniform irradiation of the body. The effective dose-equivalent is defined according to the ICRP by the equation:

$$H_E = \sum_t \overline{H}(t) \cdot w(t)$$

where: $w(t)$ is a weighting factor representing the proportion of the risk resulting from tissue "t" to the total risk valid when the whole body is uniformly irradiated. $H(t)$ is the dose-equivalent in an organ or tissue of type "t". The effective dose-equivalent thus calculated for various types of diagnostic X-ray examinations are given in references 7 and 8.

DISCUSSION

For those individuals exposed to medical irradiation, it might be of interest to know the expected individual risk for a specific type of examination. The expected individual risk will, however, depend very strongly upon the age and sex of the individual in question. The use of the effective dose-concept for this purpose would, therefore, be somewhat confusing. The expected individual risk, "EIR", can be derived from summing up the risk for each exposed organ, weighted by the relative malignancy expectancy factor, which takes the sex and age of the individual into account (4).

REFERENCES

(1) PERSSON, B.R.R.,"Medical irradiation and the use of the Effective Dose Equivalent Concept", 5th Internat. Congress of IRPA, Book of Papers, Vol. I, Pergamon Press, Oxford, (1980) 155.

(2) PERSSON, B.R.R., "Review of various weighted dose concepts used for estimation of risks from medical X-ray diagnosis", In: Patient Exposure to Radiation in Medical X-ray Diagnosis. Commission of the European Communities, EUR 7438EN, Luxemburg (1981).

(3) WALL, B.F., SHRIMPTON, P.C., "Deliberations on a suitable quantity and technique for assessing the somatic risk for diagnostic radiology", In: Patient Exposure to Radiation in Medical X-ray Diagnosis. Commission of the European Communities, EUR 7438EN, Luxemburg (1981).

(4) Wall,B.F., "A possible method for estimating of somatic doses from diagnostic radiology and Nuclear Medicine", Presented at British Institute of Radiology Congress , April 15-16, (1982).

(5) PAULY, P., "Stochastic late effects after partial body irradiation in diagnostic radiology: Evaluation of approximative data". Rad. and Environ. Biophys. 15 (1978), 21.

(6) BENGTSSON, G., BLOMGREN, P.G., BERGMAN, K., "Patient exposures and radiation risks in Swedish diagnostic radiology". Acta Radiol. Oncol. 17 (1978), 81.

(7) HASHIZUME, T., MARUYAMA, T., NODA, Y., IWAI, K., NISHIZAWA, K., TATENO, "Stochastic Risk Estimation from Medical X-ray Diagnostic Examinations. 2. Risk Estimates of Individuals from x-ray Diagnosis". Nippon Acta Radiol. 41 (1981),59.

(8) JANKOWSKI, J.,"Evaluation of the risk of neoplasm induction in the Polish population in result of X-ray radiation applied for medical purposes".(in Polish) Studia Materialy Monograficzne No.3, Lodz, Poland (1980).

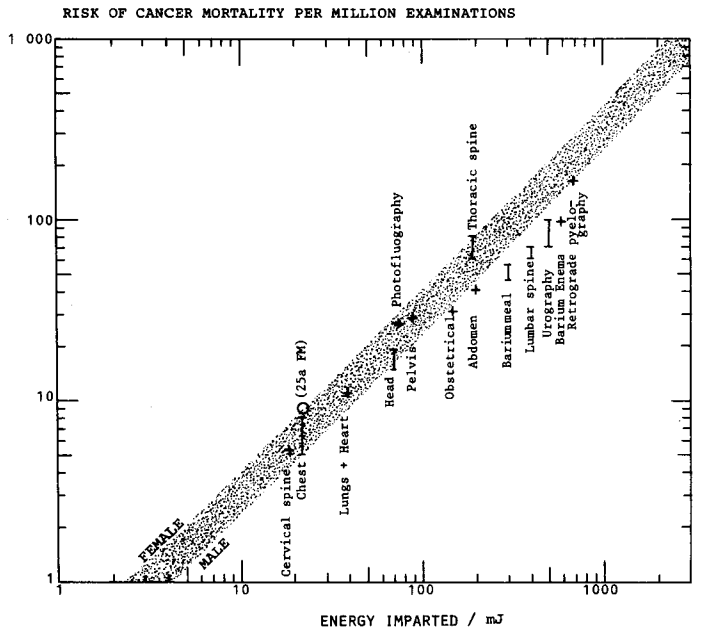

FIGURE 1. The correlation between the expected risk per million examinations and the energy imparted (mJ) for various types of diagnostic X-ray examinations.

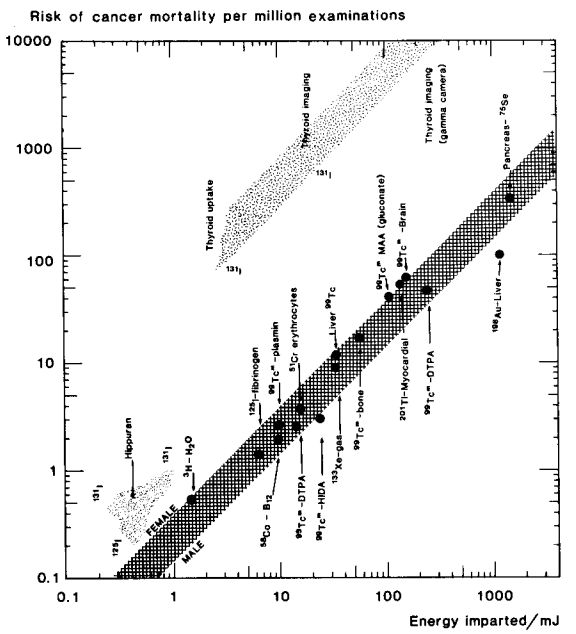

FIGURE 2. The correlation between the expected risk per million examinations and the energy imparted (mJ) for various radiopharmaceuticals used in diagnostic nuclear medicine procedures.

COMPARAISON DES RISQUES AU NIVEAU REGIONAL EXEMPLE DU SUD-EST DE LA FRANCE

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Motivations de l'étude

Les préoccupations que l'opinion publique manifeste vis-à-vis des risques susceptibles de résulter du développement de certaines activités humaines et notamment les activités industrielles, doivent inciter les responsables à prendre de plus en plus en compte cet aspect dans leurs choix et dans les décisions qu'ils sont amenés à prendre, et à intégrer toutes les conséquences s'y rattachant, qu'elles soient de nature sanitaire, économique, sociologique et psychologique.

Cette prise en compte doit s'effectuer dans le cadre d'une politique globale où tous les risques sont considérés, sans a priori, afin d'éviter de privilégier certains d'entre eux et créer pour certaines activités des handicaps fondés sur des perceptions bien souvent peu objectives.

Le meilleur niveau pour la mise en oeuvre d'une telle politique est sans aucun doute le niveau régional : il correspond à une certaine homogénéité des caractéristiques naturelles et humaines et voit se dessiner et se mettre en place les grandes options relatives au développement socio-économique et à la lutte contre les pollutions. Sa mise en oeuvre passe par un inventaire aussi exhaustif que possible des différents risques existants, l'évaluation de leurs conséquences en terme sanitaire et économique et la mise en évidence des groupes de population les plus exposés.

A cet effet, un programme d'étude, auquel collaborent plusieurs unités de l'Institut de Protection et de Sûreté Nucléaire du Commissariat à l'Energie Atomique, a été lancé en France avec l'aide de plusieurs organismes nationaux et de la Commission des Communautés Européennes*.

Pour lui donner un caractère concret, il a été choisi de l'appuyer sur un cas réel, celui d'une partie de la région Sud-Est de la France d'une superficie suffisamment vaste pour englober des activités industrielles très diversifiées, notamment dans le secteur de la production énergétique. Elle constitue donc un bon support pour chercher à établir une méthodologie générale.

Risques considérés et principes d'évaluation

Les risques auxquels peuvent être soumis les membres d'une population sont extrêmement divers selon leur origine et leur nature. Certains sont d'origine naturelle alors que d'autres résultent d'activités créées par l'homme : activités industrielles, agricoles, médicales, sociales... Ils peuvent provenir d'une activité professionnelle ou de la vie courante, d'une situation normale ou accidentelle, d'une exposition à des rayonnements ou à des substances chimiques, d'effets immédiats ou retardés, somatiques ou génétiques, létaux ou non-létaux.

La façon de procéder à leur évaluation est différente selon chacun des cas. Pour certains risques, notamment les risques accidentels, on peut disposer de références statistiques permettant leur appréciation directe : c'est le cas du nombre de décès et de lésions graves dûs aux accidents de transport ou de la fréquence des accidents et des maladies professionnelles dans les diverses branches industrielles. Dans d'autres cas, notamment lorsqu'il s'agit de risques causés par une exposition à des rayonnements ou des toxiques chimiques, l'appréciation est indirecte

* dans le cadre du contrat BIO-F-320-81-F passé entre la Direction Générale XII et le Centre d'Etude et de Développement en Hygiène et Sécurité

puisqu'elle passe par l'évaluation de l'exposition et des dommages correspondant : il est alors nécessaire de connaître la forme de la relation existant entre l'exposition et l'apparition d'effets nocifs.

On doit également étudier la distribution des risques au sein de la population afin de localiser les groupes qui, en raison de leur mode de vie, leur implantation géographique et leur profession, sont soumis au risque total le plus important.

Enfin, il faut pouvoir évaluer le coût social représenté par les différents risques, et la façon dont ils sont perçus par les individus.

Le programme d'étude : présentation et illustration

Un tel programme paraît extrêmement ambitieux : il ne peut être conduit que sur un temps suffisamment long et en procédant par étapes successives selon l'importance qui est attribuée a priori aux différentes sources de risques et selon l'aptitude plus ou moins grande à les évaluer.

Dans une première phase, l'effort a porté essentiellement sur :

- les risques radiologiques et non-radiologiques d'origine industrielle, notamment ceux qui sont engendrés par les différentes opérations entrant dans les cycles fuel, charbon et uranium, pour la production d'énergie ;

- les risques radiologiques dûs à l'irradiation naturelle (irradiation externe à l'extérieur et à l'intérieur des habitations et irradiation interne dûe à l'inhalation de ^{222}Rn à l'intérieur des habitations) ;

- les risques radiologiques dûs aux examens de radio-diagnostic systématique ;

- les risques découlant de l'exposition professionnelle.

Les tableaux 1 à 3 illustrent certains des résultats obtenus.

D'autres aspects ont également été abordés, notamment en ce qui concerne l'exposition domestique et urbaine et l'état des connaissances en matière de relation dose-effet pour les substances chimiques.

La méthode de travail adoptée comprend :

- le recensement des informations disponibles relatives à l'inventaire, la localisation et la connaissance des sources de risque, les caractéristiques de l'environnement et de la population, les émissions d'effluents, les résultats du contrôle des expositions professionnelles et de la surveillance de l'environnement. Ces informations sont souvent très dispersées entre les organismes concernés et, ayant été obtenues à d'autres fins, présentées sous des formes inadaptées

- la réalisation de campagnes de mesures des polluants dans les émissions des installations et dans l'environnement. Ces campagnes nécessitent la mise en oeuvre de moyens assez lourds et ne peuvent être conduites que ponctuellement

- la mise en place de réseaux de mesures. Cette solution, encore plus lourde et plus onéreuse, ne peut être utilisée que lorsque les paramètres à mesurer varient peu dans le temps : ainsi elle a été adoptée pour l'évaluation de l'irradiation naturelle

- l'utilisation de modèles de dispersion et de transfert dans l'environnement, après adaptation et si possible validation, qui permet d'évaluer les concentrations et les expositions dûes aux différents termes-sources.

Références

(1) Evaluation des conséquences radiologiques des rejets d'effluents de divers types de centrales (nucléaire, fuel, charbon). F. Anguenot et al. - Congrès annuel SFRP 1982 18/22 octobre 1982 - Avignon.
(2) Evaluation des conséquences sanitaires des rejets dans l'atmosphère d'effluents radioactifs et non radioactifs de divers types de centrales (nucléaire, charbon, fuel). J. Aigueperse et al. - VIème Congrès Mondial pour la qualité de l'air 16/20 mai 1983 - Paris.
(3) Study of natural irradiation associated with the dwelling place in France. A. Rannou et al. - Présenté à "International Seminar on Indoor Exposure to Natural Radiation and Related Risk Assessment" - October 3/5, 1983 - Anacapri, Italie.
(4) Influence of the way of living on domestic exposure-methodological study. R. Maximilien et al. - Présenté à "International Seminar on Indoor Exposure to Natural Radiation and Related Risk Assessment". October 3/5, 1983 - Anacapri, Italie.

T A B L E A U 1

EXPOSITION INDIVIDUELLE MAXIMALE MOYENNE ANNUELLE POUR LE GROUPE DE REFERENCE DUE AU REJET D'ELEMENTS RADIOACTIFS PAR DIVERS TYPES DE CENTRALES (en Sievert.an^{-1}, équivalents de dose effectifs)

Voie d'atteinte	Charbon	Fuel	Nucléaire
Irradiation externe	$5,4\ 10^{-8}$	$2,3\ 10^{-9}$	$8,9\ 10^{-8}$
Irradiation interne	$1,2\ 10^{-5}$	$6,6\ 10^{-7}$	$8,0\ 10^{-7}$

d'après références (1-2)

T A B L E A U 2

CONCENTRATION ATMOSPHERIQUE MAXIMALE MOYENNE ANNUELLE POUR LE GROUPE DE REFERENCE DUE AUX REJETS NON RADIOACTIFS DE CENTRALES THERMIQUES FOSSILES (en µg.m^{-3})

	charbon	fuel
SO_2	120	240
SO_4	1,2	2,4
NO_2	2,5	25
Formaldéhyde	-	$2,5.10^{-3}$
Nitrobenzène	-	$2,5.10^{-3}$
Poussières	16	3
Benzo-a-Pyrène	4.10^{-6}	4.10^{-8}

	charbon	fuel
As	8.10^{-2}	2.10^{-2}
Cd	0,3	2.10^{-2}
Cr	1,6	8.10^{-2}
Cu	0,6	-
Ni	1,6	32
Pb	0,2	0,6
V	4,0	80

d'après référence (2)

T A B L E A U 3

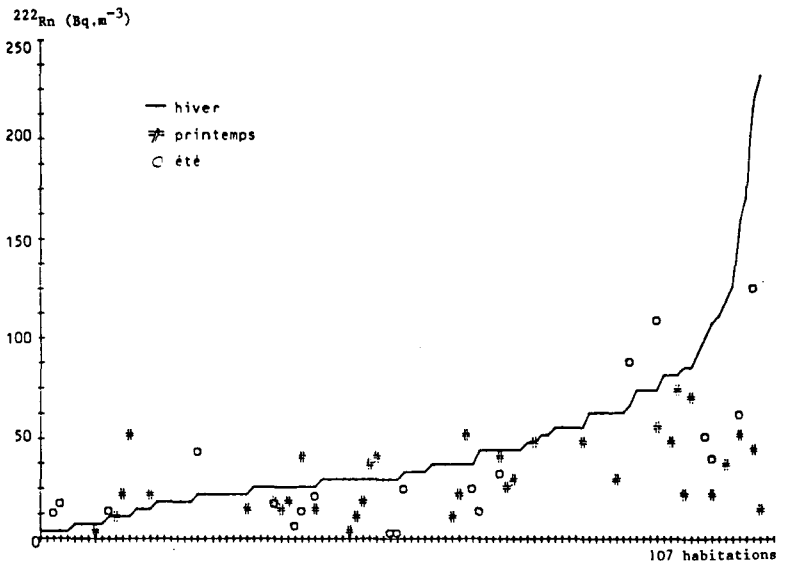
MESURES PAR DOSIMETRES THERMOLUMINESCENTS (TLD) DE L'IRRADIATION NATURELLE DANS DIFFERENTS DEPARTEMENTS DE LA ZONE GRAND DELTA, RAYONS COSMIQUES DEDUITS ($x10^{-5}$ Gy.an^{-1})

Département	A l'extérieur				A l'intérieur			
	Nbre dosimètre	Min	Moy	Max	Nbre dosimètre	Min	Moy	Max
Gard	137	21	49	93	135	13	52	153
Isère	195	9	36	69	216	10	41	94
Lozère	69	47	96	162	67	44	96	171
Rhône	197	0	48	191	235	0	48	187
Vaucluse	189	14	31	66	209	13	36	161

d'après référence (3)

F I G U R E 1

EXPOSITION DOMESTIQUE SELON LES SAISONS (^{222}Rn)



d'après référence (4)

ARE SMOKERS AT GREATER RISK FROM RADIATION EXPOSURE THAN NONSMOKERS?

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INTRODUCTION. Cigarette smoking and exposure to ionizing radiation are each known to be a cause of lung cancer. However, it is not clear whether smoking and radiation risks are additive, greater than additive such as multiplicative or possibly even less than additive. The report of the National Academy of Sciences, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980" (BEIR III) states, "If the lung cancer risk after radiation exposure is proportional to the usual age-specific rates for smokers and nonsmokers (as in the relative-risk concept, consistent with a multiplicative effect of radiation on cigarette-induced cancer), then the estimates of excess risk should be increased by about 50% to apply to smokers and reduced by a factor of about six for nonsmokers, as well as delayed in time." This would yield a factor of about ten difference in risk between smokers and nonsmokers. Whether this is a real difference could be of concern to smokers employed in occupations where there is a potential for radiation exposure and it could be important in predictive as well as in retrospective assessments of the health consequences of radiation exposures. The BEIR III report acknowledges the uncertainty of such a difference in risk by concluding that a purely multiplicative effect is unlikely, but the issue is not resolved. The following addresses some of the current information on the cigarette-smoking radiation-exposure question for different types of radiation exposure, e.g., whole-body exposure, inhalation of very short half-lived radioactive particles that irradiate primarily the tracheal and bronchial regions of the lungs (such as radon decay products) and inhalation of relatively long half-lived radioactive particles that irradiate primarily the pulmonary regions of the lung (such as plutonium particles).

EXTERNAL RADIATION. The major source of information on lung cancer induced by external radiation is the study of atomic bomb survivors. Examination of the smoking histories of lung cancer cases by Ishimaru et al. (1975) suggested no interaction between radiation and cigarette smoking. In this preliminary study there was no evidence of a multiplicative or promoting effect of cigarette smoking on radiation-induced lung cancer.

RADON DECAY PRODUCTS. A statistically significant increased incidence of lung cancer among uranium miners is correlated with their exposure to radon decay products emitting alpha particles (Federal Radiation Council, 1967; National Academy of Sciences, 1968; Lundin et al., 1971). Cancers in these miners appear to originate in the bronchial epithelium, the region of the lungs receiving the highest radiation dose (Harley and Pasternack, 1981). Many of the uranium miners were cigarette smokers. A report of the National Academy of Sciences (1968) concluded that the available data suggested that miners who smoked cigarettes were more

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*Operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO-1830.

susceptible to lung cancer than nonsmoking miners. Lundin et al. (1971) were less positive about smokers being more susceptible but acknowledged that cigarette smoking (as well as diesel smoke and other constituents of mine air) may have been contributory in some of the lung cancer cases among the miners by altering the deposition of radon decay products and thus the radiation dose in the respiratory tract or by acting as a promoting agent. Archer (1981) subsequently showed that the principal effect of cigarette smoking was to shorten the latent period for the induction of lung cancer among uranium miners. However, Whittemore and McMillan (1983) in a recent analysis of lung cancer mortality data from U.S. uranium miners conclude that the data suggest a strong multiplicative effect of cigarette smoking and exposure to radon decay products. The risk of lung cancer among uranium miners who were 20-pack-year smokers (i.e., one pack per day for 20 years) was about five times that of nonsmoking miners. This is contrary to the observation in radon-exposed Swedish zinc-lead miners that the lifetime risk of lung cancer was less among smokers than in nonsmokers (Axelson and Sundell, 1978). Similar results were observed in a life-span study of forty beagle dogs that were exposed daily for 4½ years to radon and radon decay products (Cross et al., 1982). Half of the dogs were also made to inhale cigarette smoke through their nose and mouths daily. One nasal cancer and one lung cancer occurred in the smoking dogs compared with two nasal cancers and seven lung cancers in nonsmoking dogs. The "protective effect" of smoking in the Swedish miners and possibly also in the beagle dogs is considered to have been due to the smoke causing increased mucus production or a thickened mucosa, which reduced the radiation dose to the basal cells of the bronchial epithelium.

Like the epidemiologists, pathologists have been changing their views on the role of cigarette smoking in radon-induced lung cancer. Auerbach et al. (1978) accepted the earlier data reported by Archer et al. (1976) as evidence for a synergistic effect of smoking and exposure to radon. However, in a more recent paper Saccomanno (1982) concluded that data on U.S. uranium miners do not support the thesis that cigarette smoking increases the induction of lung cancer or shortens the latent period. Information on the types of lung cancer that occur in smoking and nonsmoking uranium miners does not clarify the issue because it is difficult to separate out the relative effects of other factors, e.g., aging and induction latent time, in possibly determining the histological types of the tumors that eventually develop (Saccomanno et al., 1981).

Gottlieb and Husen (1982) noted that radiation was the primary cause of lung cancer among Navajo uranium miners who had a low frequency of cigarette smoking. Lung cancer is relatively rare among the Navajo. They concluded that cigarette smoke is a promoting agent. This concept is partially supported by experimental animal studies. Chameaud et al. (1980) found that exposure of rats to cigarette smoke at the conclusion of a protracted series of exposures to radon and radon decay products increased the incidence of lung cancer above that in rats not exposed to smoke; however, exposure to cigarette smoke before radon exposures did not increase the incidence.

At the present time, it seems that the most likely role of cigarette smoking in the induction of lung cancer among radon-exposed miners is that of a promoting agent. This would not be inconsistent with

Archer's observation that cigarette smoking shortened the latent period of induction of lung cancer.

RADIOACTIVE PARTICLES. Inhibiting effects of cigarette smoke on tracheal ciliary activity have been observed in numerous studies, for example, that of Dalhamn and Rylander (1965). The results were used to infer that cigarette smoking inhibits the clearance of inhaled particles from the lungs (Report of the Advisory Committee to the Surgeon General of the Public Health Service, 1964), which suggests the possibility that smokers exposed to airborne toxic substances such as radioactive particles are at greater risk than nonsmokers. Albert et al. (1975) found that inhaled particles were actually cleared more rapidly in smoking men than in nonsmokers. Lourenco et al. (1971) observed a temporary delay of a few hours in the clearance of test aerosol particles from the lungs of smokers. However, an effect of cigarette smoke on the clearance of inhaled $^{239}PuO_2$ has been demonstrated in rats and beagle dogs (Filipy et al. 1980, 1983). Groups of rats exposed to cigarette smoke or sham-exposed daily for seven months were given a single exposure to an aerosol of $^{239}PuO_2$ and the smoke exposures continued. Pulmonary clearance was markedly depressed in the smoke-exposed rats. These results were reproduced in an experiment with beagle dogs and in a second experiment with rats. The latter also showed that, if cigarette exposures were omitted either before or after inhalation of plutonium, the decreased clearance effect was not observed.

By depressing the clearance of plutonium in rats and dogs, smoking increases the radiation dose to the pulmonary regions of the lungs and, although it has not been demonstrated, possibly increases the risk of cancer. This follows from the observation that lung cancers induced by plutonium (and other insoluble radioactive particles) in experimental animals originate in the pulmonary region of the lungs, in tissues that receive the highest radiation doses. It can only be assumed that the same would occur in humans since plutonium-induced lung cancer has not been observed. For cigarette smoking to act as a cancer-promoting agent, it would seem that the active constituents must deposit in the irradiated tissues, in this case the pulmonary region.

CONCLUSIONS. Current information suggests that, if cigarette smoking interacts with radiation in the induction of lung cancer, it is probably as a promoting agent. There is some evidence of such an interaction in miners who were exposed to relatively high levels of radon and its decay products over extended periods, and there is evidence from experimental rats that were exposed to cigarette smoke after exposures to radon had been completed. Other data from both humans and experimental animals suggest that concomitant exposures may actually diminish the interaction of cigarette smoke with alpha radiation from radon decay products; however, the possibility of a multiplicative effect for other exposure regimes has not been dismissed.

In recent experimental animal studies cigarette smoking depressed clearance of insoluble particles, e.g., $^{239}PuO_2$, from the pulmonary regions of the lungs. However, this effect depended upon exposure to cigarette smoke both before and after inhalation of insoluble plutonium. Whether the effect on clearance actually increased the lung cancer risk is unknown.

From the above, it is still unclear whether cigarette smokers are at greater risk than nonsmokers to radiation-induced lung cancer at relatively high radiation doses and even more uncertain at low radiation doses.

REFERENCES

1. National Research Council, Committee on the Biological Effects of Ionizing Radiation. **The Effects on Populations of Exposure to Low Levels of Ionizing Radiation.** National Academy of Sciences, Washington, DC, 1980.
2. Ishimaru, T., R. W. Cihak, C. E. Land, A. Steer, and A. Yamada. **Lung Cancer at Autopsy in A-Bomb Survivors and Controls, Hiroshima and Nagasaki, 1961-1970. II. Smoking, Occupation, and A-Bomb Exposure.** *Cancer* 36:1723-1728, 1975.
3. Federal Radiation Council. **Guidance for the Control of Radiation Hazards in Uranium Mining,** Staff Report No. 8 Revised. Superintendent of Documents, U.S. Government Printing Office, Washington, DC, 20402, September 1967.
4. National Research Council, Advisory Committee from the Division of Medical Sciences. **Radiation Exposure of Uranium Miners,** National Academy of Sciences, Washington, DC, 1968.
5. Lundin, F. E., J. K. Wagoner, and V. E. Archer. **Radon Daughter Exposure and Respiratory Cancer Quantitative and Temporal Aspects,** National Institute for Occupational Safety and Health, National Institute of Environmental Health Sciences. *Joint Monograph* No. 1, 1971.
6. Whittemore, A. S., and A. McMillan. **Lung Cancer Mortality Among U.S. Uranium Miners: A Reappraisal.** *Journal of the National Cancer Institute,* 71(3):489-499, 1983.
7. Harley, N. H., and B. S. Pasternack. **A Model for Predicting Lung Cancer Risks Induced by Environmental Levels of Radon Daughters.** *Health Physics* 40:307, 1981.
8. Archer, V. E. **Health Concerns in Uranium Mining and Milling.** *Journal of Occupational Medicine* 23(7):502-505, 1981.
9. Axelson, O., and L. Sundell. **Mining, Lung Cancer and Smoking.** *Scand. J. Work Environ. Health* 4:46, 1978.
10. Cross, F. T., R. F. Palmer, R. E. Filipy, G. E. Dagle, and B. O. Stuart. **Carcinogenic Effects of Radon Daughters, Uranium Ore Dust and Cigarette Smoke in Beagle Dogs.** *Health Physics* 42:33-52, 1982.
11. Auerbach, O., G. Saccomanno, M. Kushner, R. Dawson Brown, L. Garfinkel. **Histological Findings in the Tracheobronchial Tree of Uranium Miners and Nonminers with Lung Cancer.** *Cancer* 42:483-489, 1978.
12. Archer, V. E., J. D. Gillan, and J. K. Wagoner. **Respiratory Disease Mortality Among Uranium Miners.** *Ann. NY Acad. Sci.* 271:280-293, 1976.
13. Saccomanno, G. **The Contribution of Uranium Miners to Lung Cancer Histogenesis.** *Recent Results in Cancer Research,* Vol. 82, Springer-Verlag, Berlin-Heidelberg, 1982.
14. Saccomanno, G., V. E. Archer, O. Auerbach, and M. Kushner. **Age Factor in Histological Type of Lung Cancer Among Uranium Miners, A Preliminary Report.** *Radiation Hazards in Mining: Control, Measurement and Medical Aspects.* M. Gomez, ed., Kingsport Press, Kingsport, TN, 1981.
15. Gottlieb, L. S., and L. A. Husen. **Lung Cancer Among Navajo Uranium Miners.** *Chest* 81:449-452, 1982.
16. Chameaud, J. R., R. Perraud, R. Masse, and J. Lafuma. **Contribution of Animal Experimentation to the Interpretation of Human Epidemiological Data.** *Proceedings of International Conference Radiation Hazards in Mining: Control, Measurement and Medical Aspects.* M. Gomez, ed., Kingsport Press, Kingsport, TN, 1981.
17. Dalhamn, T., R. Rylander. **Ciliastatic Action of Cigarette Smoke.** *Arch. Otolaryn.* 81:379-382, 1965.
18. **Smoking and Health—Report of the Advisory Committee to the Surgeon General of the Public Health Service,** Public Health Service Publication No. 1103, U.S. Department of Health, Education and Welfare, 1964.
19. Albert, R. E., H. T. Peterson, Jr., D. E. Bohning and M. Lippmann. **Short-Term Effects of Cigarette Smoking on Bronchial Clearance in Humans.** *Arch. Environ. Health* 30:361-367, 1975.
20. Lourenco, R. V., M. F. Klimek and C. J. Borowski. **Deposition and Clearance of 2μ Particles in the Tracheobronchial Tree of Normal Subjects—Smokers and Nonsmokers.** *J. Clinical Invest.* 50:1411-1420, 1971.
21. Filipy, R. E., et al. **Effects of Cigarette Smoke Exposure on Pulmonary Clearance of $^{239}PuO_2$ in Rats.** *Pacific Northwest Laboratory Annual Report for 1979 to the DOE Assistant Secretary for Environment,* PNL-3300, Pt. 1, Biomedical Sciences, 128-130, 1980.
22. Filipy, R. E., et al. **Cigarette Smoke and Plutonium.** *Annual Report for 1982 to the DOE Office of Energy Research,* PNL-4600, 71-73, 1983.

ANALYSIS OF THE HARM INVOLVED IN VARIOUS OCCUPATIONS

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In 1977 the International Commission of Radiological Protection (ICRP) published a report entitled "Problems Involved in Developing an Index of Harm" (ICRP-Publication 27, Pergamon Press, Oxford). The report, prepared for the Commission by Sir Edward Pochin, discussed the difficulties of making an appropriate comparison on radiation and other effects in occupational safety analyses. In the report a quantitative index was suggested that takes account of the length of life or full activity lost as a result of occupational accidents and diseases.

The basic data of occupational risks considered in the report were very limited and came only from a few nations. Therefore, a memorandum prepared by Sir Edward Pochin specifying the data needed to make a more broadly based assessment of an index of harm was sent by the Commission through various international organisations to their Member States. This paper is a response to the request for additional data.

Valuable information of the type requested can be drawn from the statistics of the industrial accident insurance system of the Federal Republic of Germany. The system covers about 20 million people. A basic legal requirement of the system is that accidents, classified in work accidents and road accidents as well as occupational diseases are to be reported. The minimum reporting requirement for an accident is that the injured person is absent from work for a period of more than three days. The minimum reporting requirement for an occupational disease is that there exists medical diagnosis that such a disease may be present. The number of reported cases fulfilling the minimum reporting requirement is considerably affected by changes of the social and economic situation. Such data are not very reliable for objective risk comparisons of various occupations. Therefore the data presented in this paper are based upon that fraction of the reported cases which led to acknowledgement of a certain degree of permanent disability. Because only a limited number of all the data available can be presented, the paper concentrates on work accidents reported during the period from 1977 to 1981 and on a limited selection of occupations.

One item requested in the memorandum referred to above concerned the age distribution of males and females involved in fatal work accidents in various occupations.

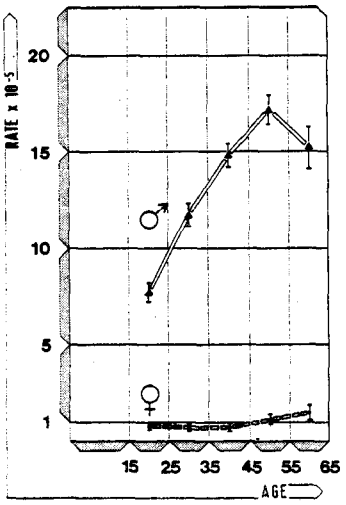

Fig. 1

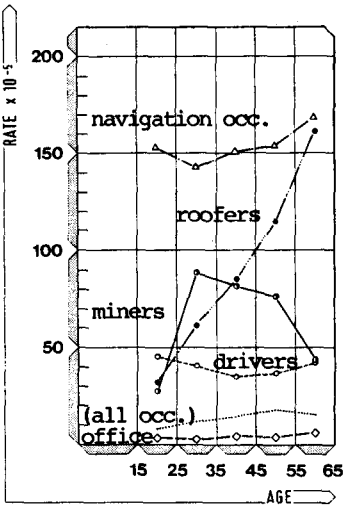

Fig. 2

Figure 1 shows the age distribution based on the annual number of fatal work accidents, averaged over all occupations considered, per 100,000 male or female workers respectively.

It illustrates that on average the male workers' accident rate is larger by more than one order of magnitude than the corresponding rate for female workers. It should however be mentioned that a few occupations exist where the female workers' accident rate is larger than the corresponding rate for male workers. This applies for example to the occupation of motor vehicle drivers.

Figure 2 shows the age distribution based on the annual number of fatal work accidents per 100,000 male workers in selected occupations. Due to a change of the scale of the accident rate axis, the average of all occupations considered, which is represented by a dotted line, now appears in the lower part of the figure. The figure illustrates not only the large differences of fatal work accident rates in the selected occupations but also the characteristic occupation related age distributions of the fatal work accident rates which differ considerably from the distribution obtained by averaging over all occupations considered.

The significant reduction of the accident rate of miners over 50 is due to the fact that labour legislation in the Federal Republic of Germany permits miners above this age to choose whether they are engaged in less dangerous mine work. No similar legislation applies to the other occupations referred to in fig. 2.

Another item requested in the memorandum referred to above concerned the age distribution of work accidents judged to involve permanent total or partial loss of working capacity.

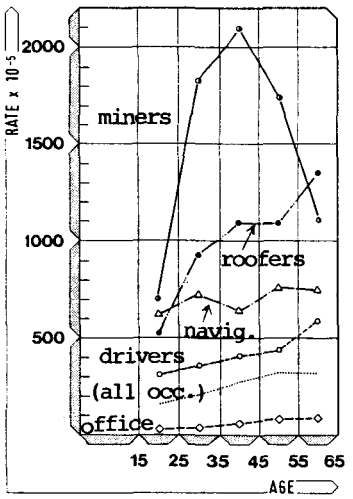

Fig. 3

Figure 3 shows the age distribution based on the anual number of non-fatal disabling accidents per 100,000 male workers in selected occupations. As in figure 2, the average of all occupations considered is represented by a dotted line. Obviously the occupation related age distributions, as found in the analysis of the fatal work accident rates, are largely maintained, but the order of the rates attributable to the selected occupations can change. If fatal work accidents are considered the highest rates relate to navigation occupations but if non-fatal disabling accidents are considered the highest rates relate to miners.

With respect to the significant reduction of the accident rate for miners over 50, again the explanation given above applies. In the other occupations selected the order of the rates appears at first sight to be maintained. It should be kept in mind however, that data on only a few selected occupations are presented in figures 2 and 3.

The annual numbers of fatal or non-fatal disabling work accidents are primarily measures of the probability with which these accidents occur. These numbers, though related to specific types of accidents do not take into account the severity of the accidents.

If fatal accidents occurring in an occupation are considered the severity of the accidents may be expressed by the average number of years lost per accident. By multiplying this average with the average number of fatal accidents occurring per 1000 workers, a measure is obtained which takes into account both the probability and the severity of the accidents.

If non-fatal disabling accidents occurring in an occupation are considered, the severity of the accidents may be expressed by the average percentage of disability caused per accident. By multiplying this average with the annual number of non-fatal disabling accidents occurring per 1000 workers, again a measure is obtained which takes into account both the probability and the severity of the accidents.

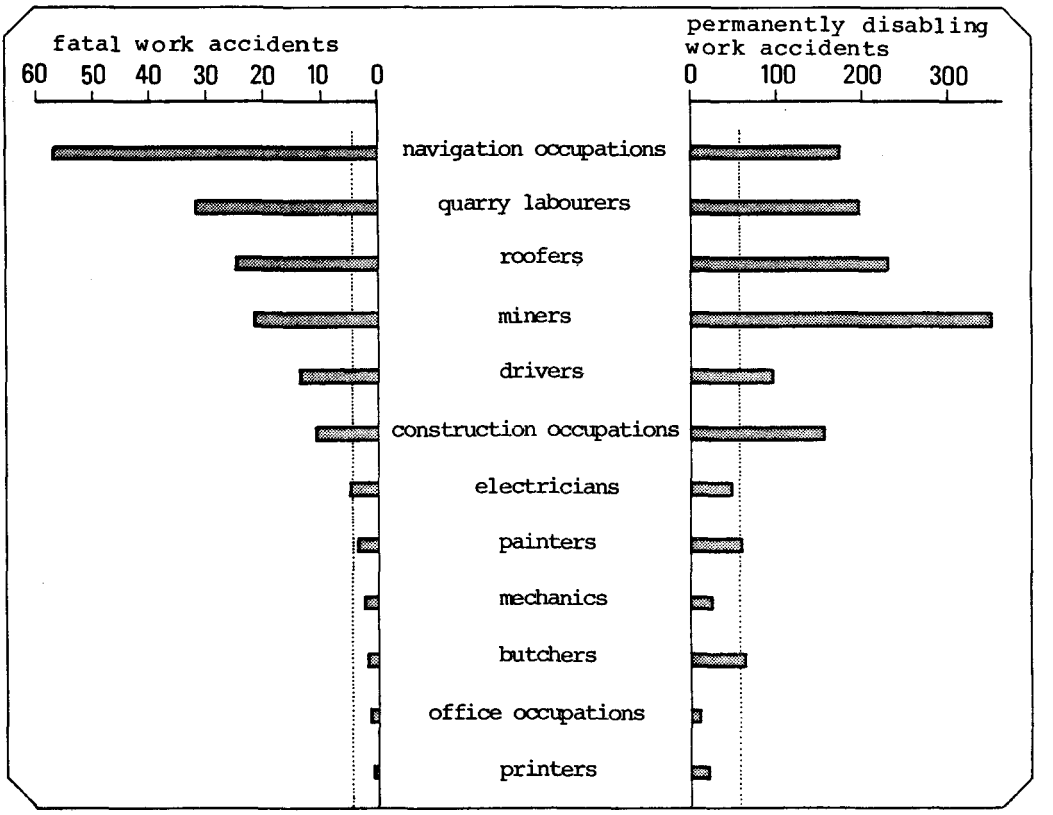

Fig. 4

In Figure 4 a comparison is made of the products of probability and severity as defined above for fatal (left side) and permanently disabling accidents (right side) occuring in selected occupations. The value of the product applicable to the average of all occupations considered is represented by a dotted line on each side.

The figure shows that significant changes of the rank order do not concern only miners and navigation occupations but also other occupations. Therefore in general it would appear to be impossible to assess the risk of non-fatal disabling work accidents from the risk of fatal work accidents.

It is hoped that information of the type presented here will be of help in developing an index of harm for occupational risks.

The authors wish to express their thanks to the Central Association of the Industrial Injuries Insurance Institutes for kindly making available the basic data needed for this study.

LUNG CANCER RISK FACTORS FOR RADON DAUGHTERS

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INTRODUCTION

In its current basic recommendations,the International Commission on Radiological Protection (ICRP) introduced a system of dose limitation (1) which limits the total individual risk from stochastic radiation effects, e.g. carcinogenesis, by taking into account the dose in all risk-relevant tissues of the human body. To meet these requirements, the "effective dose equivalent", H_E, has been defined as the product of the mean dose equivalent, H_T, and the weighting factor, w_T, in tissue T, summarized over all exposed risk-relevant tissues. This weighting factor w_T represents the ratio of the stochastic risk resulting from tissue T to the total risk from uniform whole body irradiation. From external low LET whole body irradiation a weighting factor $w_L = 0.12$ was assigned to the whole lung (1).

For the dosimetry of inhaled radon daughters the basal cells of the bronchial epithelium (TB) and the type II alveolar cells in the pulmonary region (P) are regarded as the main cells at risk for neoplastic changes. The recommended weighting factor $w_L = 0.12$ refers to the mean dose to the total lung ("Mean Dose"-concept). In the recently issued recommendations on "Limits for Inhalation of Radon Daughters by Workers" (2), ICRP recommends to split up the weighting factor for the lung into two separate weighting factors for both critical targets,$w_{TB} = w_P = 0.06$, equating the risk for tumour induction per unit dose in both tissues ("Regional Lung Dose"-concept).

This paper attempts to derive criteria for the quantification of the regional weighting factors w_{TB} and w_P on the basis of cellular radiation effects.

CRITERIA FOR THE QUANTIFICATION OF WEIGHTING FACTORS

Lung Cancer Incidence in Thorotrast Patients

Inhalation of short-lived radon daughters produces a very non-uniform dose distribution throughout the human respiratory tract, where the mean pulmonary tissue dose is about 10% of the mean bronchial basal cell dose. Because of this difference in dose, we are not able to attribute unequivocally the predominant appearance of lung tumours in bronchial tissue either to the higher dose or/and to a higher weighting factor.

In thorotrast patients, thoron escapes from the primary deposit of injected ^{232}Th in the reticulo-endothelial system and enters the lung where part of it decays while the rest is exhaled. Contrary to inhaled radon daughters, exhaled thoron and its daughters produce a quite different dose distribution throughout the respiratory tract. Based on the dose calculations by Grillmaier and Muth (3), modified for a generation-specific basal cell depth, the mean bronchial basal cell dose is only about 1/3 of the mean pulmonary tissue dose. This finding offers the possibility to check the role of the sensitive target distribution relative to the radiation dose distribution, the superposition of which determines the biological end-point. Despite the higher pulmonary dose, all lung tumours are bronchogenic carcinomas (4), suggesting that the sensitive target is the bronchial epithelium. It should be noted here, however, that no increase of the lung cancer incidence in thorotrast patients compared to the control group has been observed, although the doses are relatively high.

Uniform vs. Non-Uniform Dose Distribution

The weighting factor for the total lung has been derived from external low-LET exposure, where the total lung is irradiated nearly uniformly. Inhalation of radon daughters produces a very non-uniform dose distribution in conductive and alveolated air passages. The relative effects of uniform and non-uniform external radiation (200 kVp X-rays) on the induction of lung tumours have been studied in mice (5), which were irradiated either uniformly to the whole lung or via 72 cylindrical microbeams about 1 mm in diameter. Each geometry produced a peaked dose-effect curve for tumour induction. The most interesting result of this study is the displacement of the peak incidence from a value of 5 Gy after uniform irradiation to 1 Gy after non-uniform microbeam irradiation. In the latter case only approximately 20% of the lung is actually exposed, the irradiated part receiving five times the average lung dose. So a dose of 1 Gy to the lung from 72 microbeams involves 20% of the lung receiving a dose of 5 Gy and 80% receiving no dose. Therefore, considering the dose to the irradiated tissue only, the peak tumour incidence occurs at the same dose of 5 Gy and the percentage incidences at this dose are not statistically different from each other. This result suggests that the important factor in tumour induction is the dose absorbed by that fraction of tissue irradiated rather than the average dose to the whole lung. If we neglect the significantly smaller dose to the pulmonary tissue, thus parallelling the microbeam exposure, then lung cancer induction depends only on the dose deposited in bronchial tissue and not on the small dose contribution from the pulmonary region.

High Dose-Sites or "Hot Spots"

At doses below approximately 0.5 - 1.0 Gy, the alpha dose to a lung tissue volume reflects mainly the single hit probability for alpha particle interactions with sensitive cells. Thus, an increase in dose increases the number of cells traversed and, consequently, the probability for cancer induction. At very high doses, however, the competing effect of cellular inactivation leads to a reduction of the cancer induction probability, as has been observed for "hot particles" in the lungs. Because of the alpha depth dose distribution in bronchial epithelium, the shallow basal cells receive the highest doses, i.e. the highest frequency of alpha particle cell traversals, and are,therefore, the cells at the highest risk.

Preferential aerosol deposition at airway bifurcation sites in lobar and segmental bronchi may increase the dose to the shallow basal cells located at carinal ridges by about one order of magnitude (6). Information, however, is limited to large airway casts or models where deposition was maximum entering generation 3 and usually declined in distal airways. Because of fluid dynamics considerations, i.e. transition to laminar flow in peripheral bronchioles, and concomitant effects upon the relative efficiencies of dominant deposition mechanisms, we anticipate that the degree of bifurcation enhancement will decrease with progression through the respiratory tract. Thus we conclude that high dose-sites or "hot spots" with an enhanced carcinogenic potential occur only in large bronchial airways.

Quadratic or Cubic Dose Response

An analysis of lung cancer incidence among uranium miners led to the suggestion that the cancer induction function is a quadratic or cubic function of dose when allowing for concomitant cell killing (7). The dosimetric consequence of this finding is that all dose inhomogeneities in the human respiratory tract will become more significant, particularly at low alpha particle fluences where multiple cell traversals are negligible. Taking into account a nuclide concentration gradient across epithelial tissue, the higher risk to shallow basal cells, and enhanced

deposition at bronchial bifurcations, a distinct dose maximum will be obtained in lobar and segmental generations (8) which is about a factor 10 higher than the conventionally calculated mean basal cell dose (2). Thus, the pulmonary tissue dose is only about 1% of the highest bronchial basal cell dose. If we compare the squares or the cubes of bronchial and pulmonary doses, the contribution from the pulmonary dose to the total lung cancer induction probability will be only $1:10^4$ or even $1:10^6$.

Number Density vs. Total Number of Cells Affected

A consequence of the comparison of uniform and non-uniform lung irradiation (5) is that the critical factor for lung cancer (LC) induction is the number of cells in the irradiated tissue volume, i.e. the number density of sensitive cells, and not the total number of cells affected. Thus, a small tissue volume with a given induction probability is more effective than e.g. 100 small tissue volumes with a probability of 1/100 each. From this point of view, "hot spots" are more carcinogenic for the same number of basal cells intersected. From geometrical considerations, it follows also that the number density of basal cells in a bifurcation tissue volume at the carinal ridge is higher than along tubular airway segments. In their histological investigation of bronchial basal cell depths, Gastineau et al. (9) have found no basal cells in terminal bronchioles, suggesting that the number density of basal cells decreases with progression through the TB tree.

Critical Tissue Volume for Cancer Induction

Bevan and Haque (10) have found that a clump of about 15 - 20 adjacent sublethally damaged cells is required to form a malignant focus. Therefore, lung cancer probability is proportional to the probability for the occurrence of such a critical cell cluster. These numbers are consistent with data reported by Cohen and Creditor (11), who found that the extrapolation number for lung cancer therapy is approximately 18. This extrapolation number might be interpreted that a solid tumour consists of an ensemble of cell clusters with 18 malignant cells each which have to be inactivated separately to inactivate the whole tumour. The probability to form such a cell clump is highest for a tissue volume with the highest number density of sensitive cells, exposed to the highest dose. Apart from random accumulations of radionuclides, these critical sites with the highest probability can only be found at bronchial branching sites. Moreover, histological studies have found that preneoplastic and neoplastic lesions are concentrated at bifurcations (12).

Volume vs. Surface Effects

The weighting factor for the total lung has been derived from external low-LET uniform irradiation where no spatial correlation exists between a random energy deposition distribution and a random sensitive target structure in a tissue volume. This is still a reasonable assumption for alpha particle-emitters in the pulmonary region, although only on the macroscopic scale and not in microscopic sites. In the bronchial tree, however, there is a non-random target structure (morphological structure of bronchial epithelium) and a non-random alpha particle distribution (radionuclides are deposited on airway surfaces). Because of the finite range of alpha particles in tissue, the irradiated target structure is closely correlated with the surface nuclide distribution. For low doses the degree of correlation is given by the single hit probability for a defined configuration. The effect of volume vs. surface nuclide distribution relative to a given target structure can be illustrated by comparing the mean basal cell dose (surface) to the tracheobronchial compartment dose (volume) where the latter is smaller by about a factor 3. From solid angle considerations this correlation should be smaller in the first generations, where only a few basal cells are lying within alpha particle ranges, and should then increase in more peripheral air passages.

Radiological and Biological Promotion Factors

For a discussion of the effect of promotion factors on lung cancer incidence we assume that all sensitive cells in the lung have the same radiosensitivity. Based on the initiation-promotion concept of carcinogenesis, we identify malignant transformation of stem cells as the initiation event and stimulation of their mitotic activity as the promotion event (8). Radiological promotion can be achieved by killing the non-proliferating epithelial cells which originate from the stem cells. In the bronchial tree the epithelial cell dose parallels the deposition pattern with a maximum in lobar and segmental bronchi, while in the pulmonary region the epithelial cell dose is the same as the pulmonary tissue dose, and, thus, significantly smaller than in the bronchial tree. Biological, i.e. non-radiological promotion by inhaled synergistically acting lung irritants, such as cigarette smoke, dust, chemical agents, etc. may be simulated by the surface deposition pattern of aerosols in the micrometer range, which peaks again in lobar and segmental bronchi.

CONCLUSIONS

All criteria which have been derived in this paper to quantify regional lung cancer risk weighting factors, either on a theoretical or experimental basis, suggest that the weighting factor for bronchial tissue exceeds by far the weighting factor for pulmonary tissue. Therefore, the total weighting factor for the lung $w_L = 0.12$ should be applied to the bronchial region. The above discussion also suggests that lobar and segmental bronchi are the prime target for lung cancer induction.

REFERENCES

(1) International Commission on Radiological Protection (ICRP), 1977. ICRP Publication 26, Annals of the ICRP, Vol. 1, No. 3, Oxford, Pergamon Press.
(2) International Commission on Radiological Protection (ICRP), 1981, ICRP Publication 32, Annals of the ICRP, Vol. 1, No. 3, Oxford, Pergamon Press.
(3) Grillmaier R. and Muth H., 1971. Health Physics 20, 409.
(4) van Kaick G., Muth H., Kaul A. et al., 1983. Health Physics 44, Suppl. 1, 299.
(5) Coggle J.E. and Peel D.M., 1978. In: Symp. on Late Biological Effects of Ionizing Radiation, Vol. II, IAEA-SM-224/205, IAEA, Vienna.
(6) Hofmann W. and Martonen T., 1983. Health Physics 43, 147.
(7) Hofmann W. and Katz R., 1983. In: Proc. 8th Symp. on Microdosimetry, pp. 565, CEC Report EUR 8395.
(8) Hofmann W., 1984. Radiation Protection Dosimetry (in press).
(9) Gastineau R.M., Walsh P.J. and Underwood N., 1972. Health Physics 23, 857.
(10) Bevan J.S. and Haque A.K.M.M., 1968. Physics in Medicine and Biology 13, 105.
(11) Cohen L. and Creditor M., 1982. Radiation Research 91, 405.
(12) Auerbach O., Stout A.P., Hammond E.C. and Garfinkel L., 1961. New England Journal of Medicine 265, 253.

HEALTH EFFECTS ESTIMATION FOR CONTAMINATED PROPERTIES

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Tailings from the processing of ores in uranium mills have been a matter of considerable interest in the United States for the past two decades. Before the health hazard associated with the tailings was recognized, the excellent properties of the tailings materials as landfill led to their removal from the piles to be used as landfill at construction sites for houses, commercial buildings, and other locations. In a few cases the tailings were even incorporated in the materials that were used in building construction. As a result, the U.S. Congress has authorized and currently funds major programs to evaluate the need for and institute remedial actions designed to minimize health hazards from inactive tailings piles and from displaced tailings. When the displaced tailings are present on defined properties, these are referred to as "vicinity properties." As part of the overall remedial action program sponsored by the Department of Energy, our project was initiated to develop methods for estimating health effects projected to occur at vicinity properties and to apply those techniques to representative properties. To date, the bulk of our work has been concentrated in the Salt Lake City area in the state of Utah.

When structures are located on vicinity properties, the tailings may either be restricted to the land surrounding a building or may also lie under the foundation of the building. If the tailings are at least partially under the building, build-up of airborne radon daughters is likely to occur within the structure, to an extent that will depend upon the air exchange rate and other factors. The elevated radon daughter concentrations create an incremental risk of lung cancer in exposed persons. The radium, uranium, their daughters, and possibly other radionuclides in the tailings beneath and around a structure also give rise to gamma-ray exposure, which is usually of lesser consequence, but, in some cases, may be the only detectable potential cause of health effects. Finally, in the case of residential structures, radionuclides in tailings mixed with the soil in garden areas around the building may be a relatively important source of human exposure. Garden food crops grown in contaminated areas can incorporate radium and uranium in sufficient amounts through plant uptake from the soil to give rise to a calculable risk to exposed individuals from their ingestion. Thus, we are concerned with three potential sources of health effects--inhalation of radon and daughters giving rise to lung cancer, whole-body exposure from gamma-ray emission causing cancer in general, and ingestion of radionuclides, especially radium and uranium, as a cause of cancer in general and bone cancer in particular.

In estimating health effects, we have used radiological survey data collected by personnel of the Oak Ridge National Laboratory and Mound Facility. Gamma-ray measurements are easily obtained

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with scintillation survey meters and such data have been quite complete. The gamma-ray measurements of particular interest to us are those obtained at 1 meter above the ground or the floor in buildings. Radon gas and daughter measurements, which are of greater importance in estimating health effects, are more difficult and time-consuming and, therefore, fewer. Their values are much more variable than those of gamma radiation and, hence they are a less satisfactory measure of average annual exposure. The radon measurements were performed either on grab, continuous or integrated samples. Radon daughter measurements were obtained only for grab samples. Radionuclide concentrations in soil were always analyzed for ^{226}Ra and ^{238}U and, sometimes, ^{232}Th.

In estimating the number of health effects for contaminated properties, it is necessary to make certain assumptions. We adopted the following: (1) The property is considered in its current state, the "as is" condition. The only exception occurs when there is obvious underutilization of a building and additional occupancy is likely to occur in the near future. (2) Insofar as the data available to us permit the calculation of an average exposure, our risk estimates are based on average rather than maximal exposure. Maximal exposure estimates are too easily distorted by aberrant values. In the case of soil contamination, high concentrations of radionuclides in small areas will produce poorly representative risk estimates from ingestion of food crops. (3) Fifty-year periods of exposure are assumed, independent of the occupancy of the premises by any particular set of individuals. (4) An annual occupancy of 2,000 hours is assumed for commercial structures and 7,000 hours for residences.

There are several measures of health risk. We have focused our interest on the following: (1) the probability that an individual will incur cancer, either lung cancer due to radon daughter exposure, cancer of any type due to gamma-ray exposure or food ingestion, or bone cancer due to food ingestion; (2) the increase in the probability that an individual will develop cancer over the expected normal rate for the population; and (3) the total projected number of deaths for a property based on its occupancy, whether residential or commercial.

The number of health effects due to gamma-ray exposure is calculated by averaging the available exposure values over areas of occupancy and using a risk coefficient of 10^{-4} fatal cancers per rem of gamma-ray exposure (UNSCEAR 1977). In calculating lung cancer risk for radon daughter exposure one must make the best possible use of the available data. The risk coefficients for radon daughters were based on a method developed by Harley and Pasternack (1981), which has been adopted by the National Council on Radiation Protection and Measurements (NCRP) Task Group on Radon. The magnitude of the risk coefficient varies with the equilibrium factor and unattachment fraction of the radon daughters. The reference value for the risk coefficient is 5.6×10^{-3} lung cancer deaths per working level month per year for lifetime risk and lifetime exposure in the case of a population having the age distribution of the United States population in 1975. This value is applicable at an equilibrium factor of 0.7 and an unattached RaA/Rn ratio of 0.07. In our calculations we have adjusted the risk coefficient for the equilibrium factor but not

for the unattachment fraction due to the absence of relevant data. Measurements of radon and daughters should be taken at intervals throughout the year and should be of a continuous or integrated nature because of seasonal and diurnal variation in radon daughter concentrations. When integrated radon measurements were available, they were used; otherwise, grab values were employed. Because the risk coefficient is dependent on the value of the equilibrium factor, simultaneous grab radon gas and daughter measurements are desirable. When values for the equilibrium factor were not available, a default factor of 0.3 was used, based on our experience with this set of vicinity property data and information available from other investigators compiled under similar climatic conditions.

The calculation of effects due to ingestion of food crops is carried out by means of a computer program, PABLM, which has been developed at the Pacific Northwest Laboratory. For Salt Lake City we assume that 50% of the fruit and vegetable portion of the diet may be grown in an individual's home garden, a reasonable assumption this area. The program converts soil radionuclide concentration data into estimates of deaths due to cancer in general and bone cancer in particular. The latter is calculated separately because of the bone-seeking nature of the relevant radionuclides.

Examples of the results of health effect calculations in Tables 1 and 2 illustrate some of the variation in contamination conditions at different properties in the Salt Lake City area. Residence 1 involves gamma-ray, radon daughter and food pathway exposure routes. The relative contributions of these components are typical in that the radon daughter exposure predominates and the food pathway is next in importance. The tailings are confined to the land surrounding the house in Residence 2. As a result, no health effects are attributed to radon daughters; the food pathway is more important than the limited outdoor gamma-ray exposure that is assumed in this case. Health effect estimates for the commercial properties illustrate a comparison of cost-effectiveness for remedial action in that the projected cancer deaths for Commercial Property 2 approach the value for Commercial Property 1, but the former is a smaller property with lesser occupancy and would probably entail a lesser cost of cleanup for the benefit obtained. Both properties will require remedial action; however, the results of such calculations will assist in decisions regarding priorities for cleanup in numerous cases when the indications for remedial action are marginal.

This approach to assessing the hazard associated with the tailings material at properties may have important implications. If, after calculating projected health effects, we proceed to next relate our results to the cost of cleanup, we have a measure of the cost-effectiveness of a proposed remedial action. The need for remedial action may be measured in terms of the cost of cleanup per (1) cancer death, (2) man-rem or (3) years of loss of life span averted. An appropriate measure of cost-effectiveness may then be used in establishing priorities for cleanup when large numbers of candidates for remedial action are under consideration.

A second potential use of an accumulated data base containing such information would lie in the periodic reevaluation of stan-

Table 1

Cancer risk, increase of risk over normal, and projected deaths attributable to tailings material at contaminated properties.

Exposure Type	Occupancy	Individual Risk ($\times 10^{-4}$)	% Increase Over Normal Risk	Cancer Deaths
Gamma-Ray				
Residence 1	6	4.6	0.3	0.003
Residence 2*	4	0.4	0.02	0.0002
Commercial 1	160	1.7	0.1	0.027
Commercial 2	8	6	0.4	0.005
Radon Daughter**				
Residence 1	6	190	46	0.11
Residence 2	4	0	0	0
Commercial 1	160	29	7	0.46
Commercial 2	8	410	100	0.33
Food Pathway***				
Residence 1	6	46	3	0.03
Residence 2	4	25	1.5	0.01

*Calculations based on outdoor gamma-ray exposure of 500 hours.

**These values represent lung cancer for radon daughter exposure.

***No food pathway exposure was assumed for commercial properties.

Table 2

Contributions from exposure routes and total projected cancer deaths for properties in Table 1.

Property	Gamma-Ray Exposure	Food Pathway	Rn Daughter Exposure	Total Deaths
Residence 1	0.003	0.03	0.11	0.14
Residence 2	0.0002	0.01	0	0.01
Commercial 1	0.03		0.46	0.49
Commercial 2	0.005		0.33	0.34

dards. In the United States, the Environmental Protection Agency (EPA) sets and periodically reviews standards, including those relating to remedial action for inactive uranium mill tailings. Cost considerations have a role in the setting of those standards. Data based on direct measures of the cost-effectiveness of remedial actions may be useful to EPA in addressing this aspect of their review of current standards and to the Department of Energy in responding to proposed standards.

References

Harley, N. H. and Pasternack, B. S. (1981). "A model for predicting lung cancer risks induced by environmental levels of radon daughters," Health Phys. 40, 307.

UNSCEAR (1977). United Nations Scientific Committee on the Effects of Atomic Radiation. Sources and Effects of Ionizing Radiation. United Nations, New York.

ANALYSIS OF POTENTIAL RADIATION-INDUCED GENETIC AND SOMATIC EFFECTS TO MAN FROM MILLING OF URANIUM

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INTRODUCTION

Potential radiation-induced somatic and genetic effects are among the risks associated with radionuclides released during mining and milling of uranium. These risks are associated with inhalation and ingestion of the radionuclides and external exposure to radiation. Quantification of these risks is a sequential process of estimating rates of release to and dispersion and uptake of radionuclides within the ecosystem, determining the levels of exposures and uptake of the radionuclides by man, calculating radiation doses to organs at risk, and, finally, predicting potential somatic and genetic effects as a function of time and age of the exposed individuals.

Comprehensive methods for computation of each of the sequential steps were previously reported (Momeni et al. 1979, 1983). The dosimetric methods used here were previously applied to analyses of pathways of exposure from the Jackpile-Paguate Uranium Mines in New Mexico, USA (Momeni 1983) and generic analyses of uranium milling in the USA (USNRC 1980). The calculation of potential radiation-induced genetic and somatic effects from mining of uranium was previously reported (Momeni 1983; Momeni et al. 1983). This report constitutes a summary of the analyses of radiation-induced genetic and somatic effects associated with uranium milling using the previously published dosimetric models data (USNRC 1980). The risks were calculated using the PRIM code (Momeni 1983). The risk-analysis method utilized in the PRIM code is based on competitive morality in a dynamically changing population. The birth and death rates are calculated from the lifetable and cohort-component method and are based on 1969-1971 census statistics of the population in the USA. The probabilities of radiation-induced risks are calculated using age-dependent absolute and relative risk models. The risk coefficients were adopted from the National Academy of Sciences report BEIR III (NAS 1980).

PATTERNS OF MORTALITY

The patterns of mortality (i.e., specific death rates as a function of time) for all causes, including spontaneous and radiation-induced neoplasms, were calculated from demographic characteristics of the exposed population. The exposure patterns are for a hypothetical "model" mill processing 1800 tonnes of ore daily and operating continuously for 15 years as described by the U.S. Nuclear Regulatory Commission (USNRC 1980). The processed ore was assumed to have an average specific activity of 460 pCi U-238/g and to be under secular radioactive equilibrium. The exposed population was assumed to be 57 428 persons (USNRC 1980).

Cumulative five-year mortality from spontaneous neoplasms in the exposed population for an observation period of 85 years was calculated (Table 1). For the same period, radiation-induced mortality from the same cancers was predicted

*Work partially completed at Argonne National Laboratory, Argonne, IL, USA under contract W-31-109-Eng-38.

Table 1. Spontaneous Cancer Mortality for Five-Year Periods
Calculated using PRIM Code†[1]

Time (Years)	CA-1	CA-2	CA-3	CA-4	CA-5	CA-6	CA-7	CA-8	CA-9	CA-10	CAS	CAS/DEATH
5	18.5	96.6	28.8	121.0	39.6	2.6	24.9	24.9	24.9	342.1	724.0	2.07E-01
10	18.9	99.1	29.6	123.9	39.4	2.7	25.5	25.5	25.5	348.8	739.0	2.05E-01
15	20.4	104.8	32.3	134.6	41.8	2.8	28.4	28.4	28.4	373.9	795.7	2.02E-01
20	21.9	109.8	34.8	144.4	44.5	2.9	31.0	31.0	31.0	397.9	849.1	2.00E-01
25	23.2	114.1	37.0	152.7	47.3	3.0	33.3	33.3	33.3	419.0	896.0	2.04E-01
30	24.3	119.3	38.9	160.3	50.3	3.2	35.2	35.2	35.2	440.2	942.1	2.12E-01
35	25.2	125.7	40.6	167.7	53.3	3.3	36.8	36.8	36.8	461.6	987.9	2.16E-01
40	26.1	133.9	42.3	175.7	56.3	3.5	38.2	38.2	38.2	485.1	1037.4	2.10E-01
45	27.2	143.9	44.4	185.2	58.9	3.6	39.6	39.6	39.6	512.3	1094.3	2.20E-01
50	28.6	155.5	47.3	197.7	61.7	3.9	41.7	41.7	41.7	546.9	1166.5	2.25E-01
55	30.7	166.9	51.3	213.5	64.9	4.2	45.0	45.0	45.0	587.7	1254.3	2.32E-01
60	33.2	176.4	55.8	230.5	68.6	4.4	49.5	49.5	49.5	629.1	1346.4	2.34E-01
65	35.4	183.2	59.6	244.8	72.1	4.6	53.7	53.7	53.7	662.8	1423.4	2.32E-01
70	36.8	187.5	62.0	253.9	74.5	4.7	56.5	56.5	56.5	683.7	1472.5	2.29E-01
75	37.4	190.8	63.0	257.8	75.9	4.8	57.5	57.5	57.5	693.5	1495.8	2.27E-01
80	37.6	194.7	63.3	259.6	76.8	4.9	57.4	57.4	57.4	701.2	1510.5	2.26E-01
85	38.1	200.1	64.3	263.8	78.2	5.0	57.6	57.6	57.6	716.3	1538.4	2.29E-01
Total	483.6	2502.2	795.3	3287.1	1004.2	64.0	711.7	711.7	711.7	9001.9	19 273.4	

†[1] CA = Cancer; CA-1 = Leukemia; CA-2 = Lung; CA-3 = Stomach; CA-4 = Intestines; CA-5 = Breast; CA-6 = Bone; CA-7 = Liver + Pancreas; CA-8 = Sex + Urinary; CA-9 = Lymphoma; CA-10 = All Others; CAS = Total of all cancers; death = death from all cancers and ages.

by both the absolute and relative risk models to be less than 0.1 in any consecutive five-year period. Cumulative deaths (85-year integration) from all spontaneous neoplasms were 19 273.4, and from radiation-induced neoplasms were 0.5 and 1.3 as calculated by the absolute and relative risk models, respectively.

The ratios of predicted mortality from radiation-induced neoplasms to mortality from spontaneously induced neoplasms are 2.5×10^{-5} and 6.7×10^{-5} from the absolute and relative risk models, respectively. Based on these analyses, during the first five-year period, spontaneously induced neoplasms represent 20.7% of all deaths, increasing over the 85-year period to 22.9%. Similarly, radiation-induced deaths constitute 1×10^{-5}% of all deaths in the first period, increasing to 3×10^{-3}% over 85 years.

GENETIC EFFECTS

The numbers of radiation-induced dominant and multifactorial genetic disorders were calculated using equilibrium genetic-induction factors of 6.0×10^{-7} and 5.0×10^{-7} disorders per mrem for dominant and multifactorial effects, respectively. The exposure period for males was limited to ages older than 15, and for females to ages younger than 50. The exposure period of female germ cells was assumed to be continuous and cumulative from birth to copulation; for males, exposure of germ cells was assumed to be limited to only 36 days of continuous irradiation prior to copulation. The doses to gonads were normalized on the basis of radiobiological sensitivity by factors 0.8 and 0.2 for females and males, respectively (NAS 1980). The cumulative genetic disorders for the entire population over the 85-year period were predicted to be 8.0×10^{-3} and 6.7×10^{-3} for dominant and multifactorial disorders, respectively. The current incidence of genetic disorders from natural causes in the USA is 10.7% for live born (NAS 1980).

SUMMARY AND DISCUSSION

Potential mortality from natural causes and from radiation exposure conditions typical of those in the vicinity of uranium mills in the western USA was calculated. The exposure conditions were those assumed to exist in the vicinity of a hypothetical model mill as previously described (USNRC 1980). Dose rates to organs at risk were calculated as a function of time using the Uranium Dispersion and Dosimetry Code (Momeni et al. 1979). The changes in population size, birth rates, and radiation-induced and natural mortalities were calculated using the PRIM code (Momeni 1983) and are summarized in Table 2.

The population of the region within a radius of 80 km from the model mill is projected to increase from 57 428 to 75 638.6 during the 85 years of this analysis. Within the same period, the average birth rates for five-year periods increase from 5067.8 to 7436.1. The cumulative deaths within the five-year periods increase from 724 and 3501.8 from spontaneously induced neoplasms and all causes, respectively, to 1538.2 and 6718.2. In comparison to natural causes, radiation-induced mortality is negligible. The highest rate of death from radiation in any five-year period is only 0.2, compared with 1538.2 deaths attributable to spontaneous incidence. The total radiation-induced genetic disorders were much less than unity for the 85-year period of analysis, in contrast with the 10.7% natural incidence of these disorders.

Table 2. Cumulative Deaths from Natural and Radiation-Induced Causes

Time (Years)	Total Population	Total Births	Cancer Deaths: Radiation-Induced: Absolute Risk Model	Cancer Deaths: Radiation-Induced: Relative Risk Model	Cancer Deaths: Natural	All Deaths
5	57 428.0	5067.8	0.0	0.0	724.0	3501.8
10	58 994.0	5491.8	0.0	0.0	739.0	3597.7
15	60 888.1	5829.8	0.0	0.0	795.7	3940.5
20	62 777.4	5765.0	0.0	0.0	849.1	4238.6
25	64 303.7	5501.4	0.0	0.0	896.0	4384.1
30	65 421.0	5443.8	0.0	0.0	942.1	4452.7
35	66 412.1	6211.8	0.0	0.0	987.9	4566.6
40	68 057.3	5994.1	0.0	0.0	1037.4	4935.9
45	69 115.5	6221.4	0.0	0.0	1094.3	4979.4
50	70 357.4	6278.4	0.0	0.1	1166.5	5187.1
55	71 448.7	6397.1	0.0	0.1	1254.3	5412.3
60	72 433.6	6622.8	0.0	0.1	1346.4	5743.9
65	73 312.3	6822.0	0.1	0.1	1423.4	6127.4
70	74 006.9	6973.3	0.1	0.2	1472.5	6432.0
75	74 548.1	7111.7	0.1	0.2	1495.8	6600.6
80	75 059.1	7253.7	0.1	0.2	1510.5	6673.2
85	75 639.6	7436.9	0.1	0.2	1538.4	6718.8

REFERENCES

Momeni, M.H., Y. Yuan, and A.J. Zielen. 1979. The Uranium Dispersion and Dosimetry (UDAD) Code. NUREG/CR-0553, ANL/ES-72. Prepared for U.S. Nuclear Regulatory Commission by Argonne National Laboratory.

Momeni, M.H. 1983. Use of PRIM code to analyze potential radiation-induced genetic and somatic effects to man from Jackpile-Paguate mine. In Epidemiology Applied to Health Physics. CONF-83-0101. Health Physics Society.

Momeni, M.H., S.Y.H. Tsai, J.Y. Yang, A.B. Gureghian, and C.E. Dungey. 1983. Radiological Impact of Jackpile-Paguate Uranium Mines: An Analysis of Alternatives of Decommissioning. ANL/ES-131. Prepared for U.S. Department of Interior by Argonne National Laboratory.

NAS (National Academy of Sciences). 1980. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation (BEIR III). Report of the Advisory Committee on the Biological Effects of Ionizing Radiation. National Academy of Sciences - National Research Council.

USNRC (U.S. Nuclear Regulatory Commission). 1980. Final Generic Environmental Impact Statement on Uranium Milling. NUREG-0706. Office of Nuclear Material Safety and Safeguards.

SUR L'EVALUATION DES RISQUES INDIVIDUELS ET COLLECTIFS POTENTIELS DES REJETS NORMAUX OU EXCEPTIONNELS EN MILIEU AQUATIQUE

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I - INTRODUCTION

Dans l'évaluation des risques potentiels des rejets d'effluents dans le milieu aquatique, l'importance relative des rejets normaux des différentes installations conduit à s'intéresser tout d'abord aux effluents des usines de retraitement rejetés en milieu marin. En effet, on sait que les rejets de routine des installations nucléaires de la C.E.E. contribuent pour une part minime à l'irradiation de la population. L'équivalent de dose collective engagée du fait des rejets de l'ensemble des installations a été estimé pour 1978 / LUYCKX 82 / à environ 500 man-Sv, contre 62 000 man-Sv pour l'ensemble des autre sources d'irradiation (naturelle, médicale, retombées d'essais nucléaires, professionnelles, etc ...), soit moins de 1 % . Sur ces 500 man-Sv, 90% proviennent des effluents des usines de retraitement, rejetés principalement par voie liquide. Les quantités rejetées en sites côtiers (Windscale, La Hague) sont beaucoup plus importantes que celles rejetées en rivière (WAK, Marcoule) en raison des capacités d'acceptation de l'environnement. Pour ces rejets, le principal vecteur de contamination est l'ingestion de produits de la mer. De ce point de vue, une précédente étude montre que l'incorporation collective résultant de rejets en Méditerranée serait considérablement moins importante que celle due aux rejets en Manche (200 à 300 fois moins pour la population française pour un même rejet de ^{137}Cs / LOCHARD 81 /.

D'autre part, l'impact de rejets exceptionnels par leur nature, leur quantité, ou les conditions dans lesquelles ils seraient effectués, doit pouvoir être évalué, au niveau collectif et au niveau individuel, et il est important de connaître la distribution des doses individuelles et de localiser les groupes susceptibles d'être les plus exposés.

L'évaluation de l'impact des rejets en mer nécessite tout d'abord la connaissance du transport et de la dispersion des masses d'eau et des matières en suspension, puis celle des différents mécanismes et paramètres de transfert du milieu marin à l'homme / ANCELLIN 79 /. De nombreuses études, mesures et interprétations, ont déjà été effectuées dans ces domaines et les données collectées doivent permettre d'étayer des modèles ou d'envisager leur application à des cas bien définis. Pour le premier problème, concernant la dispersion des effluents, de nombreux modèles existent, en particulier des modèles locaux. Une étude en cours, dans le cadre de notre programme, porte sur l'application d'un modèle de circulation résiduelle à l'évaluation des concentrations jusqu'à de longues distances du point de rejet, en tenant compte de la climatologie des régions concernées.

II - ZONES ET MOYENS D'ETUDES

Divers organismes ont de longue date effectué des études orientées vers la protection radiologique ou vers les recherches océanographiques et biologiques, dont la confrontation nous paraît riche d'enseignements.

2.1. La présence de radioactivité dans les eaux marines est non seulement un sujet de problème mais aussi une source d'enseignements. Rappelons brièvement

qu'en différentes zones de l'Atlantique Nord-Est, des prèlèvements d'eau de mer et des analyses de divers radioéléments (surtout le ^{137}Cs) ont été effectués systématiquement depuis de nombreuses années. Ce sont les différentes parties de la Mer du Nord explorées par l'Institut Océanographique de Hambourg depuis 1969 pour la partie Nord, et 1971 pour la partie Sud et pour la Manche, d'une part ; la Mer d'Irlande, les Minches, le Nord de l'Ecosse, surveillés par le F.R.L. (Lowestoft), d'autre part, (11 croisières effectuées de 1971 à 1978). La Baltique et les mers adjacentes sont également explorées et les résultats de mesures sont intéressants notamment pour l'étude du comportement des radioéléments dans le milieu marin. Les radioéléments considérés comme traceurs ont permis d'observer le déplacement des masses d'eau et des matières en suspension (soit en suivant un radioélément particulier, surtout le ^{137}Cs, soit en examinant les rapports entre les niveaux d'activité $^{238}Pu/^{239+240}Pu$, $^{241}Am/^{239+240}Pu$, $^{239+240}Pu/^{137}Cs$ qui permettent d'en détecter l'origine, connaissant les proportions présentes dans les effluents. Les trajets moyens des effluents, qui dépendent fortement des conditions météorologiques et des caractéristiques hydrographiques, ont été estimés, ainsi que les facteur de dilution, et diverses explications sont proposées pour les variations saisonnières des courants.

Dans les zones côtières ont été recueillis pour l'analyse de la radioactivité des échantillons de produits de la pêche côtière, des sédiments et divers échantillons biologiques : côtes britanniques, côtes françaises de la Manche, de Brest à Dunkerque et plus particulièrement côtes normandes et bretonnes (Laboratoire de Radioécologie Marine de la Hague), et d'autre part, côtes françaises de la Méditerranée (Laboratoire de l'AIEA, Monaco). La comparaison des résultats d'analyses de différents radionucléides montre que les proportions transférées de l'eau aux sédiments varient selon les radionucléides et selon les régions et les paramètres d'environnement. L'interprétation est difficile et une connaissance approfondie du milieu est nécessaire à l'interprétation du comportement des éléments considérés / IAEA-TECDOC 265, 1982 /.

Les principaux trajets des matières en suspension ont été décrits, tant pour la Mer du Nord / KAUTSKI 81 / que pour la Mer l'Irlande / MAUCHLINE 80 /. Ils sont à compléter pour la Manche. D'après les mesures effectuées depuis 1976 dans les sédiments littoraux de sites répartis entre Brest et Dunkerque, la dispersion des radionucléides provenant des rejets industriels, essentiellement de la Hague, semble se faire selon deux directions différentes / GUEGUENIAT / : la fraction soluble (^{125}Sb, ^{137}Cs) s'orientant vers la Manche Orientale, comme le laissait prévoir l'étude de site (rejets de rhodamine, largage de bouées lestées), la fraction particulaire (^{144}Ce, ^{95}Zr, ^{106}Ru en partie) s'orientant vers l'Ouest, jusqu'au secteur compris entre l'embouchure du Trieux et la baie de Morlaix). Le piégeage de certains éléments dans le Golfe Normand Breton suggère l'existence d'une cellule de circulation résiduelle cyclonique, s'ajoutant au courant résiduel hivernal allant du Cotentin vers la Bretagne (en été, une partie du courant Ouest-Est se renverse, au contraire, en direction de la pointe de la Cornouaille).

On a noté un décalage d'environ deux ans entre le rejet à l'émissaire et la présence de ^{144}Ce dans les sédiments sur le littoral breton. Les tempêtes peuvent jouer un rôle dans le transfert par la remise en suspension des vases contaminées.

De nouvelles mesures dans l'eau de mer et les sédiments (programme en cours du Laboratoire de Radioécologie Marine, auquel participe l'Association) devraient permettre de compléter et d'interpréter ces résultats, comparativement avec ceux résultant de l'application des modèles adéquats.

2.2. Pour la recherche océanographique, il existe dans l'Atlantique Nord-Est d'importants moyens (stations et navires) d'observation et d'enregistrement

des vagues et des marées, ainsi que des données météorologiques. En Méditerranée, des moyens ont été développés dans le but d'étudier les interactions air-mer (Programme Medalpex) (bouée océanographique de Calvi).

2.3. Enfin, des programmes de recherches en biologie marine sont susceptibles de favoriser non seulement la compréhension des transferts, mais aussi celle des phénomènes de dispersion. Leurs méthodes d'étude combinent les mesures à la mer, la télédétection et la modélisation mathématique.

- Les mesures à la mer comprennent celles des paramètres physiques tels que température, salinité, et des paramètres biologiques tels que teneurs en phytoplancton, éléments nutritifs ou chlorophylle, entre lesquels des corrélations ont pu être établies.

- La télédétection, aéroportée ou par satellite, permet d'obtenir des images sur lesquelles on peut détecter les zones où des différences de température (photo infrarouge) ou de couleur (mesures de réflectance) permettent, compte tenu des corrélations précédentes, de déceler la variation de caractéristiques hydrobiologiques et la présence de courants, de tourbillons, ou de fronts.

Les zones ainsi décelées peuvent être examinées plus en détail, en vue d'études théoriques (par exemple pour l'intercalibration des méthodes) ou en vue d'applications pratiques (aide à la navigation et à la pêche, description des panaches turbides en zône côtière) ou même les deux à la fois. Citons par exemple - l'étude de la croissance du phytoplancton dans des régions de fronts thermiques en présence de tourbillons, sur le pourtour des Iles Britanniques et à proximité de la Bretagne / PINGREE 75 et 78 / - l'étude de l'enrichissement biologique sur le front d'Ouessant (interprétation d'images fournies par LANDSAT /VIOLLIER 80 pub 7 / qui suggère une interprétation intéressante pour l'étude des phénomènes de diffusion (possibilité d'échanges verticaux grâce à une certaine érosion de la thermocline en bordure du front, due à l'augmentation d'intensité des courants de marée en période de vive-eau) - l'étude du champ thermique à la surface de la mer sur le plateau celtique / GRALL 81 /, comparant les images obtenues par exploitation des données de satellites (programme SATIR) et les résultats de mesures effectuées par deux navires.

Citons encore des expériences d'intercalibration entre les survols aériens et les mesures à la mer (" vérité-mer ") qui ont permis de décrire le gradient côtier des propriétés hydrobiologiques (turbidité des eaux, concentration en chlorophylle) dans le détroit du Pas-de-Calais / VIOLLIER 80 / (Lille et Wimereux).

Dans le cadre du projet EURASEP, une intercalibration préliminaire entre images fournies par Coastal Zone Color Scanner et mesures à la mer avait été effectuée en Mer Ligure, entre Nice et le Cap Corse en Mars 1979. Les premières analyses ont mis en évidence l'importance des corrections atmosphériques.

Les avantages que constituent l'étendue et la répétitivité des images fournies par satellites permettent d'espérer beaucoup de données utiles à la compréhension de certains phénomènes océanographiques. L'interprétation doit cependant rester prudente et tenir compte des particularités locales et saisonnières. Dans l'Atlantique Nord-Est, la fréquence de la couverture nuageuse constitue un inconvénient de cette méthode en réduisant le nombre des images utilisables, d'où l'intérêt de mesures aéroportées. Dans la Méditerranée, par contre, des conditions atmosphériques beaucoup plus favorables ont permis d'établir des cartes mensuelles des fronts grâce à 1200 images obtenues sur deux ans, et d'étudier les variations saisonnières et interannuelles de leur position et de leurs contours / PHILIPPE 82 /.

- La modélisation mathématique peut contribuer à ces études sur le plan pratique, en prévoyant approximativement les limites des zones marines où des mesures " in situ " détaillées seraient le plus utiles, compte tenu des moyens et du temps nécessaires.

Ainsi, le modèle numérique de marée de/PINGREE et GRIFFITHS 78 / permet de prédire la position et les contours de la plupart des régions de fronts. Les différences entre prédictions et observations par satellite ou mesures à la mer sont liées notamment à la profondeur et à la satinité des eaux, aux variations géographiques du flux de chaleur, aux effets du vent et des vagues, ainsi qu'aux variations saisonnières.

Inversement, de nombreuses données étant nécessaires à la mise en oeuvre des modèles mathématiques, il serait intéressant de mettre à profit la télédétection pour compléter les autres sources d'information. Selon NIHOUL/82 / qui a souligné les exigences de qualité et les progrès à réaliser en vue de l'application aux modèles hydrodynamiques, la meilleure voie serait la mesure directe des flux, permettant de déterminer les conditions à l'interface air-mer, dont une formulation trop simplifiée ou approximative pourrait conduire à de notables erreurs.

III - CHOIX D'UN MODELE

La complexité des problèmes abordés conduit en effet à s'orienter vers des modèles explicatifs fondés sur les principes de l'hydrodynamique et la connaissance des caractéristiques du milieu marin et de leur variabilité. Certains résultats d'observation paraissent contradictoires entre eux ou avec les prévisions / GUEGUENIAT 79 / / JEFFERIES 82 /. Plusieurs explications sont possibles : non seulement différences de comportement des radionucléides, modification des quantités rejetées ou des conditions de rejet, mais aussi variabilité des conditions climatiques ou occurence de situations extrêmes, pouvant se conjuguer avec l'influence de paramètres négligés ou dont la fluctuation est mal connue. Ceci plaide en faveur d'une modélisation suffisamment détaillée, écartant des simplifications abusives, ce qui toutefois n'exclut pas les schématisations.

Dans une analyse comparative de différents modèles hydrodynamiques / MEJON 83 /, effectuée dans le cadre de notre programme, les principes, les méthodes de résolution et les limitations des modèles sont étudiés de façon approfondie, en prenant pour exemples d'application l'étude des marées et tempêtes (Mer du Nord), l'étude des courants résiduels (Mer du Nord, Manche) ou des courants littoraux (côtes françaises de la Manche et de l'Atlantique). Les modèles de dispersion des polluants sont présentés, en particulier celui mis au point à l'Université de Liège, qui a été retenu pour une étude à plus grande échelle (plateau continental nord-européen limité par l'isobathe de 200 mètres environ). Au cours de cette étude, effectuée sous la direction de J.C. NIHOUL, on souhaiterait répondre aux questions suivantes :

- Quels sont à l'échelle européenne les trajets prévisibles des polluants, en fonction de la nature de ceux-ci, des conditions de rejet et des caractéristiques de l'environnement ?

- Quelles seront les concentrations de ces éléments dans les vecteurs de contamination les plus importants à plus ou moins longue échéance ?

Les applications seront faites en simulant des situations météorologiques typiques ainsi que des situations réelles pour lesquelles existent des observations in situ (1976). Les résultats de calculs seront autant que possible comparés aux résultats des mesures de radioactivité, et pourront éventuellement être pris en considération pour le choix d'autres zônes à explorer.

EIN VERGLEICH ZWISCHEN KONSERVATIVEN UND REALISTISCHEN DOSISABSCHÄTZUNGEN FÜR DIE UMGEBUNGSBEVÖLKERUNG VON KERNKRAFTWERKEN.

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Moderne Kernkraftwerke geben bei Normalbetrieb nur geringe Mengen radioaktiver Stoffe an die Umwelt ab; in der Umgebung treten daher keine messbaren Immissionen auf. Die Strahlenbelastung der Umgebungsbevölkerung kann nur aufgrund der gemessenen Emissionen über Abluft und Abwasser, unter Zuhilfenahme von Ausbreitungsmodellen und Uebergangsfaktoren sowie Annahmen über die Lebensgewohnheiten der Bevölkerung berechnet werden. Solche Rechnungen benutzen konservative d.h. die ungünstigsten Annahmen bezüglich der Modellparameter und ergeben eine obere Grenze für die Dosis. Bei der Herleitung der Abgabelimiten für die Betriebsbewilligung, ausgehend von einer als Schutzziel festgelegten Dosislimite für die Umgebungsbevölkerung (Schweiz: 20 mrem/Jahr) ist dies auch gerechtfertigt. Für die tatsächlichen Strahlendosen durch die effektiven Abgaben ist dagegen eine realistische Rechnung vorzuziehen, da die konservative zu einer Dosisüberschätzung führt. In der vorliegenden Arbeit wird anhand der Radioaktivitätsabgaben des Kernkraftwerkes Mühleberg (bei Bern/Schweiz) die konservative mit Varianten einer realistischen Dosisabschätzung verglichen. (Tab. 1) Die Dosen der verschiedenen Belastungspfade wurden für den kritischen Ort d.h. die Stelle, wo die grössten Immissionen auftreten (Bauernhof "Ufem Horn" 500 W des KKW), Ausbreitungsfaktor χ_L =5.5E-7 s/m^{3}bzw. ψ_L=7.0E-4 s/m^2, berechnet. Als weitere Variante wird ein Mittel über die Nahumgebung des Werkes angegeben. (mittlere Ausbreitungsfaktoren : χ_L=1.7E-7 s/m^3 , bzw. ψ_L=1.4E-4 s/m^2). Mit Ausnahme der Dosen durch die Bodenstrahlung, die nach (3) und (9) berechnet wurden, und der realistischen Abschätzung der Edelgasdosen nach (5) und (6) wurden die Dosen gemäss (2) berechnet.

Als Basis für die Berechnungen dienten die Radioaktivitätsabgaben des KKW Mühleberg, gemäss (1) und (4), gemittelt über die Jahre 1978-82. (Tab. 2).Die Ausbreitungsfaktoren sind den durch Feldmessungen unterstützten Berechnungen von (5) entnommen. Als mittlere Wasserführung für die Aare, den Vorfluter des Werkes, wurde 120 m^3/s angenommen, gemäss dem Hydrographischen Jahrbuch der Schweiz.

Bei der Strahlendosis durch radioaktive Edelgase nimmt die konservative Rechnung dauernden Aufenthalt im Freien am kritischen Ort an, wobei auch die β-Hautdosis miteinbezogen wird. Bei der realistischen Rechnung wird nur die γ-Dosis berücksichtigt, und eine reduzierte Aufenthaltszeit im Freien angenommen: kritische Bevölkerungsgruppe (z.B. Bauern, Fischer) 44 h/Woche d.h. 2300 h/Jahr, für das Bevölkerungsmittel 10 h/Woche d.h. 520 h/Jahr. Für den Aufenthalt im Hausinnern wird eine Verminderung der Strahlendosis um einen Faktor 10 angenommen. Die konservative Rechnung summiert alle Edelgasnuklide in Xe-133-Aequivalent und ermittelt die Dosis mittels Verdünnungsfaktor für β-Immersion und γ+β-Dosisfaktor für Xe-133. Die realistische Rechnung benutzt den Verdünnungsfaktor für γ-Submersion, berücksichtigt den Zerfall bei kurzlebigen Nukliden zwischen Freisetzung und Aufpunkt und berechnet die Dosis für jedes einzelne Nuklid mittels des γ-Dosisfaktors. Die Dosen durch Edelgasinhalation werden in jedem Fall vernachlässigt.

Bei den Dosen durch Aerosolinhalation (die externe Bestrahlung durch Aerosole aus der Abluftwolke wird vernachlässigt) wird bei der realistischen Rechnung eine reduzierte Aufenthaltsdauer im Freien angenommen, wie bei den Edelgasen. Im Hausinnern wird eine um einen Faktor 3 verminderte Aerosolkonzentration angenommen. Auch hier berücksichtigt die realistische Rechnung bei kurzlebigen Nukliden den Zerfall zwischen Freisetzung und Aufpunkt. Die gleichen Ueberlegungen wie für die Aerosole gelten auch für die Inhalationsdosen durch Jod-131.

Für die Strahlenbelastung durch Jod-131 über den Milchpfad wird ein täglicher Milchkonsum vom 0.4 L für Erwachsene bzw. 0.8 L für Kleinkinder angenommen, sowie für das Sommerhalbjahr (Grünfütterungszeit) ein Luft-Milch-Transferfaktor von 700 pCi/L pro pCi/m^3 Luft(3),(10). Während die konservative Rechnung ausschliesslich den Konsum der Milch von am kritischen Ort grasenden Kühen voraussetzt, wird bei der realistischen Rechnung mit einer Milchmischprobe aus der gesamten Umgebung gerechnet. Dies dürfte - mit Ausnahme weniger Selbstversorger - auch die Regel sein. Auch bei Kleinkindern dürfte die ausschliessliche Ernährung mit frischer Kuhmilch eine seltene Ausnahme sein. Berücksichtigt man weiter, dass die ICRP-26 die Schilddrüsendosis nur noch mit 3% gewichtet, dann müssen die hier angegebenen Ganzkörperdosen durch Jod-131 noch um einen Faktor 5 reduziert werden.

Bei der Bodenstrahlung durch abgelagerte, langlebige Radionuklide als Aerosole wird eine (trocken + Niederschläge) Ablagerungsgeschwindigkeit von 1.3 cm/s angenommen, sowie eine Akkumulation (mit teilweisem Eindringen in den Boden) über 50 Jahre bei der konservativen Rechnung bzw. 10 Jahre bei der realistischen Rechnung. (Dosisfaktoren nach (3) und (9)) Bei der realistischen Rechnung gilt wiederum, wie bei den Edelgasen, eine reduzierte Aufenthaltszeit im Freien und eine Reduktion der Strahlendosis im Hausinnern um einen Faktor 10.

Für den Ingestionspfad via Ablagerung aus der Luft und Anfnahme der Pflanzen aus dem Boden wird ein Jahreskonsum von 146 Liter Milch, 158 kg pflanzliche Produkte und 58 kg Fleisch angenommen. Während die konservative Rechnung von einer Akkumulation im Boden über 50 Jahre ausgeht und ausschliesslich Produkte verwendet, die am kritischen Ort erzeugt wurden, rechnet die realistische Rechnung mit Produkten, die aus der gesamten Umgebung des KKW stammen und einer Akkumulation im Boden von nur 10 Jahren. Eine Belastung durch Bewässerung von Kulturen mit Wasser aus dem Fluss unterhalb des Werkes wird nicht in Betracht gezogen.

Bei der Strahlendosis durch Trinkwasser geht die konservative Rechnung davon aus, dass der gesamte tägliche Wasserkonsum von 2.2 Liter aus dem Fluss unterhalb des Werkes gedeckt wird. Die realistische Rechnung nimmt an, es handle sich ausschliesslich um (unkontaminiertes) Quellwasser. Die Belastung über den Trinkwasserpfad ist somit Null. Die Bestrahlung durch Schwimmen im Fluss und Wandern am Ufer (Bestrahlung durch im Fluss abgelagerte radioaktive Stoffe) wird vernachlässigt.

Für die Belastung durch Fischverzehr nimmt die konservative Rechnung einen Jahresverzehr von 20 kg Fisch aus dem Fluss unterhalb des Werkes an, während die realistische Rechnung für die kritische Bevölkerungsgruppe (Fischer) 5 kg/Jahr (d.h. ca. 100 Fische pro Jahr à 200 - 300 g, wovon 1/2 bis 2/3 essbarer Anteil, verzehrt von einer dreiköpfigen Familie = 5 kg/Jahr) sowie für das Bevölkerungsmittel 0.5 kg pro Jahr (Süsswasserfischkonsum in der Schweiz) annimmt.

Insgesamt ergibt somit die realistische Rechnung für den kritischen Ort eine Reduktion der gesamten Strahlendosis gegenüber der konservativen Rechnung von etwa einem Faktor 5 für die kritische Bevölkerungsgruppe (Bauern, Fischer, Förster etc.), für die übrige Bevölkerung gar um einen Faktor 25. Im Mittel über die ganze Nahumgebung des Kernkraftwerkes ist die realistisch abgeschätzt Strahlendosis durch die Abgaben des Werkes gar um einen Faktor 40 kleiner als diejenige durch die konservative Rechnung. Auch die hier vorgeführten sogenannten realistischen Abschätzungen enthalten immernoch gewisse konservative Annahmen. Deren vollständige Elimination dürfte jedoch schwierig sein, und würde auch nicht mehr zu einer ins Gewicht fallenden weiteren Reduktion der berechneten Strahlendosen der Umgebungsbevölkerung führen. Es ergibt sich jedoch klar aus den hier durchgeführten Abschätzungen, dass die konservative Dosisberechnung eine recht grosse Sicherheitsmarge gegenüber den tatsächlichen Dosen der Umgebungsbevölkerung hat.

Tabelle 1: Aus den Radioaktivitätsabgaben berechnete Strahlendosen der Umgebungsbevölkerung in µrem/Jahr für konservative und realistische Annahmen. (Erklärungen siehe Text)

Belastungspfad µrem/Jahr	Abgabelimiten gemäss Reglement konservative Rechnung kritischer Ort	tatsächliche Abgaben konservative Rechnung kritischer Ort	tatsächliche Abgaben realistische Rechnung kritische Gruppen kritischer Ort	tatsächliche Abgaben realistische Rechnung kritischer Ort Bevölkerungsmittel	tatsächliche Abgaben realistische Rechnung Mittel für Umgebung Bevölkerungsmittel
Edelgase, Bestrahlung aus der Abluftwolke	11'000	120	10	4.8	0.62
Jod-131, Inhalation	14	0.52	0.25	0.18	0.056
Jod-131, Ingestion Milchpfad	110	4.0	1.2	1.2	1.2
Aerosole, Inhalation	110	0.8	0.38	0.27	0.23
Aerosole, Ablagerung Bodenstrahlung	640	4.5	0.85	0.39	0.12
Aerosole, Ingestion landw. Produkte	310	2.2	0.24	0.24	0.24
Aerosole, Ingestion Milch	76	0.53	0.07	0.07	0.07
Aerosole, Ingestion Fleisch	51	0.36	0.05	0.05	0.05
Wasser, Trinkwasser	400	13	0	0	0
Wasser, Fischverzehr	6100	200	50	5	5
Summe	19'000	350	63	12	7.6

LITERATURANGABEN:

(1) Jahresberichte der Eidg. Kommission zur Ueberwachung der Radioaktivität (KUER)
(2) Schweizerische Verordnung über den Strahlenschutz (30.6.76)
(3) H. Pfeiffer: HSK 12/149 Vorschlag der HSK für die Festlegung der Abgabelimiten für das KKW Leibstadt und Berechnung der resultierenden Dosen der Bevölkerung.
(4) H. Pfeiffer u. J. Czarnecki: ASK 11/57 Aerosolabgaben des KKM. 9.10.80
(5) G. Schriber: Beiträge zur Bestimmung der Strahlenbelastung in der Umgebung von Kernkraftwerken. Dissertation Universität Bern 15.2.1979.
(6) H.D. Brenk u. K.J. Vogt: Dosisfaktoren zur Berechnung der Strahlenexposition durch radioaktive Abluft aus kerntechnischen Anlagen. KFA Jülich, Jan. 1977.
(7) D.C. Kocher: Dose-Rate Conversion Factors. Health Physics 38, April 80, 543ff

Tabelle 2: Radioaktivitätsabgaben des Kernkraftwerkes Mühleberg
Mittelwerte der Jahre 1978 - 82; nach (1) und (4).

Abluft: Edelgase

Isotop	HWZ	Ci/a
Kr- 89	3.17 m	0.5
Xe-137	3.9m	68
Xe-135m	15.6m	47
Xe-138	17m	105
Kr- 87	76m	2.7
Ar- 41	1.83h	0.5
Kr- 88	2.8h	95
Kr- 85m	4.4h	66
Xe-135	9.2h	40
Xe-133	5.27d	65
Kr- 85	10.3a	4

Summe der Edelgasabgaben, umgerechnet in Xe-133-Aequivalent: 3400 Ci/Jahr

Abgabelimite gemäss Reglement: 300'000 Ci/Jahr Xe-133-Aequivalent

tatsächliche Abgaben ca. 1.1% der Abgabelimite.

Abluft: Jod-131

Isotop	HWZ	mCi/a
J -131	8.07	18

Abgabelimite gemäss Reglement: 500 mCi/a

tatsächliche Abgaben ca. 3.6 % der Abgabelimite.

Abluft: Aerosole

Isotop	HWZ	mCi/a
Rb- 89	15.2m	2500
Cs-138	32.2m	1800
Ba-139	82.7m	1500
Mn- 56	2.85h	110
Sr- 92	2.7 h	80
Sr- 91	9.5h	11
Mo- 99	66.0h	0.23
J -131[1])	8.07d	0.3
Ba-140	12.2d	1.1
Cr- 51	27.8d	0.26
Ce-141	33d	0.007
Nb- 95	35d	0.0015
Sr- 89	51d	0.7
Co- 58	71d	0.015
Zn- 65	245d	0.14
Ag-110m	253d	0.003
Ce-144	285d	0.16
Mn- 54	290d	0.036
Cs-134	2.1a	0.05
Sb-125	2.77a	0.03
Co- 60	5.26a	0.43
Sr- 90	28a	0.017
Cs-137	30a	0.26

Summe der Aerosolabgaben mit HWZ grösser als 8 Tage: 3.51 mCi/Jahr

Abgabelimite gemäss Reglement: 500 mCi/Jahr (HWZ>8d)

tatsächliche Abgaben ca. 0.7 % der Abgabelimite.

Abwasser:

Isotop	HWZ	mCi/a
H - 3	12.3a	14'000
Cr- 51	27.8d	13
Mn- 54	290d	20
Co- 58	71d	10
Co- 60	5.26a	200
Fe- 59	45d	0.1
Zn- 65	245d	40
Sr- 89	51d	20
Sr- 90	28a	10
Y - 90	64.2h	10
Nb- 95	35d	0.7
Zr- 95	65d	0.6
Ru-103	40d	0.1
Ag-110m	253d	0.5
Sb-124	60d	0.1
J -131	8.07d	8
J -133	21h	0.6
Cs-134	2.1a	120
Cs-137	30a	540
La-140	1.67d	2.5
Ba-140	12.8d	2.5
Ce-141	33d	1

Summe der Abwasserabgaben (ohne H-3) umgerechnet auf ein Nuklid mit einem C_w-Wert von 10^{-4} Ci/m^3: 330 mCi/ Jahr

Abgabelimite gemäss Reglement: 10'000 mCi/Jahr 10^{-4}-Aequivalent; H-3: 500'000 mCi/a

tatsächliche Abgaben ca. 3.3 % 10^{-4}-Aequivalent und ca. 2.8 % für H-3.

m = Minuten
h = Stunden
d = Tage
a = Jahre

[1]) nur Aerosol-Jod

LITERATURANGABEN (Fortsetzung):

(8) Allgemeine Berechnungsgrundlage für die Strahlenexposition bei radioaktiven Ableitungen mit der Abluft oder in die Oberflächengewässer. BMI Bonn, 15.8.79
(9) H. Völkle u. J. Czarnecki: Beweissicherung KKW Gösgen-Däniken: 1. Zwischenbericht über die radiologische Umgebungsüberwachung. Nov 1978.
(10) J. Czarnecki, H. Pfeiffer u. H. Völkle: ASK 17/43 Begründung für die Abgabegrenzwerte für radioaktive Stoffe für das Kernkraftwerk Gösgen-Däniken. 22.2.80

RELATIVE RADIATION HAZARDS OF COAL BASED AND NUCLEAR POWER PLANTS IN INDIA

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Coal, like most materials found in nature, contains trace quantities of the naturally occurring primordial radionuclides. The levels of these radionuclides are usually low. However, they assume importance when large quantities of coal are burnt in thermal power plants for electricity generation. India has to largely depend on its more than 85770 million tonnes (MT) of recoverable coal reserves for her energy needs during next few decades. The annual requirement of coal by 2000 AD is expected to go beyond 425 MT. Thermal power stations generated about 57% of the total power production of 29000 MWe in the year 1982, consuming about 40 MT of coal (1). Combustion of coal disperses the radioactivity in the coal over large areas in the vicinity of thermal power stations (TPS) through fly-ash emissions from stacks. Many of these TPS are located in areas which are thickly populated and as such the environmental impact studies assume importance. A study was undertaken to assess the radioisotope concentration in coal collected from different mines spread all over the country. The summary of levels of ^{226}Ra, ^{228}Th and ^{40}K content in coal from some of these coal fields in India is given in Table 1. Most of the power plants use coal produced in the vicinity of power plants. Hence, the release of radioactivity from such power plants can be calculated, knowing its location, installed capacity, ash content of coal and control measures taken for particulate emissions.

Region		Radioactivity content in Bq/kg					
		COAL			FLY ASH		
		^{226}Ra	^{228}Th	^{40}K	^{226}Ra	^{228}Th	^{40}K
Neyveli coal fields		81.4	48.1	199.8			
Western coal fields							
(a) Madhya Pradesh		22.2	40.7	103.6			
(b) Maharashtra		14.8	25.9	70.3			
(c) Nagpur		29.6	40.7	99.9			
Nasik TPS	1978	44.4	29.6	114.7	159.1	133.2	340
480 MWe	1980	33.3	37.0	62.9	103.6	207.2	326
	1982	18.5	33.3	122.2	70.3	122.1	348
Neyveli TPS	1978	14.8	-	7.4	866	107.3	-
600 MWe	1980	18.5	37.0	14.8	759	111.0	56
	1981	25.9	37.0	151.7	762	111.0	-

Table 1: Natural radioactivity content in coal from Neyveli and Western Coal Fields regions and coal and fly-ash samples from Nasik and Neyveli TPS.

In this study, we have covered the TPS at Neyveli and Nasik. These plants are of fairly modern design with efficient fly-ash control. Lignite coal from South India has the highest uranium content and coal from Western coal-fields the lowest. Hence TPS operated on lignite coal at Neyveli and on Western coal at Nasik were studied for radiological risk assessment. The coal, fly-ash and bottom ash samples from these power stations were analysed for their radioactive content using low-level gamma ray spectrometer (2,3). The results of analysis are also

given in Table 1. For calculating the release of radioactivity through the stacks of these TPS, it is assumed that the coal contains 30% ash and utilities have 95% fly-ash control to abate particulate pollution. It is further assumed that 15 tonnes of coal per day is required to generate 1 MWe electricity and the TPS operates for 180 days in a year, on an average. Practically all the releases from the stacks get dispersed and deposited within a circle of 80 Km radius around the utility. The total population living in these areas is estimated using average population density statistics from 1981 census for the states under consideration (4).

In considering the radiation health hazard from the operation of TPS, inhalation exposure is considered as important because of insoluble nature of fly-ash, enrichment of radionuclides in fly-ash over coal and because the radium inhaled through fly-ash would be present in higher concentrations in the lungs than in the other soft tissues. Air-borne release is thus considered as the main pathway through which the people living around coal-fired TPS are exposed to enhanced levels of natural radioactivity. The collective radiation dose commitment due to inhalation to population living in the 80 Km radius area have been estimated using the assumptions given in UNSCEAR 1977 (5) regarding average dilution factor and variation of the concentrations as a function of distance and corrections for population density. The collective effective dose commitments thus calculated for the atmospheric releases from these two TPS are given in Table 2 for lung, bone marrow and bone lining cells. It also gives total activity of ^{226}Ra and ^{228}Th released from these TPS. The collective effective dose commitments and releases of ^{40}K have not been estimated because potassium compounds are very soluble in the lung and its concentration in the body is homeostatically controlled.

Name of the TPS Capacity & Population Density	Activity Release GBq/year		Body tissue under consideration	Collective Effective Dose Commitment in Person-Sv/Year			
	^{226}Ra	^{228}Th		^{226}Ra	^{228}Th	^{228}Th	Total
Nasik	2.2	3.0	Lung	0.20	0.7	9.5	10.4
480 MWe			Bone marrow	0.0040	0.01	0.82	0.84
200/Km^2			Bone Lining cells	0.03	0.05	12.24	12.3
Neyveli	15.5	2.4	Lung	2.6	1.00	13.90	17.5
600 MWe			Bone marrow	0.05	0.02	1.19	1.26
370/Km^2			Bone Lining cells	0.39	0.08	17.8	18.3

Table 2: Estimates of collective effective dose commitment from operations of thermal power plants arising from Inhalation during the cloud passage.

Operation of nuclear power plants also releases some radioactivity to the environment. Recent cost benefit methodology calls for comparative risks from different energy sources. Hence the relative radiation risk of TPS assessed above has been compared with that of BWR type reactor at Tarapur. The estimate of collective effective dose equivalent commitment per unit power generated by BWR type reactor has been given by Hamilton (6). It is assumed in this comparison that Nasik and Neyveli TPS are operated with same power output as Boiling Water Reactors, other conditions remaining the same. The noble gas releases from Heavy Water Reactors of the type at Kota are about 5% of that for BWR, hence the dose is between 1-5% depending on the distance from the reactor and hold-up facilities provided in the reactor (7). The conversion of the collective dose commitments into collective effective dose equivalent commitment allows a better evaluation of the impact of the various radionuclides and of the various pathways. Table 3 summarizes the collective effective dose equivalent commitments resulting

from atmospheric discharges of Nasik and Neyveli TPS and similar capacity nuclear power stations operated as BWR and HWR type reactors at Tarapur and Kota, respectively. It is seen from the table that the doses from nuclear power plants are less than 10% of those from TPS type power generating stations.

Name of TPS & capacity	Person-Sv/Year		
	Actual	If operated as BWR	If operated as HWR
Nasik 480 MWe	1.72	0.15	0.0080 (Max)
Neyveli 600 MWe	2.80	0.34	0.017 (Max)

Table 3: Estimates of collective effective dose equivalent commitments from the TPS and same power plant operated as BWR or HWR.

A survey of the environmental gamma radiation dose was carried out around Nasik and Neyveli TPS and Tarapur Atomic Power Station (TAPS) and Rajasthan Atomic Power Plant (RAPP). This was done using locally fabricated gamma radiation survey meter having sensitivity of 0.1 μR/Hr. Table 4 gives the results of this survey. At the time of survey TAPS was operating at 150 MWe, RAPP at 150 MWe and Nasik and Neyveli power stations at 480 MWe and 600 MWe, respectively. Ash pond areas of TPS gave relatively higher radiation doses of 19.5 μR/hr and 20.2 μR/hr for Nasik and Neyveli TPS, respectively. The radiation doses in the environments of all the power stations were comparable and were in the range of variations of natural background doses,(Table 4).

In conclusion, the study shows that the radiation doses from TPS are comparable to those for nuclear power plants. All the doses are in the range of natural background radiation dose, the highest doses being about three times the average natural background dose of about 7 μR/hr. It is also seen that the collective effective dose equivalent commitments computed for 80 Km radius around TPS are an order of magnitude higher than those for nuclear power plant of similar size.

ACKNOWLEDGEMENTS

Authors express their sincere thanks to the authorities of Nasik and Neyveli TPS for providing the necessary samples for analysis. Measurements on the samples were made using a sensitive anticoincidence type gamma-ray spectrometer fabricated through the assistance of IAEA research contract No.RC/2717.

REFERENCES

(1) India 1982, A reference annual, Chapter 18, Energy, Publications Divisions, Govt. of India, New Delhi, 283-302 (1982)
(2) Lalit, B.Y. and Ramachandran, T.V., Natural radioactivity in Indian food stuffs, (In) Adams, J.A.S. & Lowder, W.M. ed., Proc. of Natural Radiation Environment III, Rice University, Houston,Texas, 800-809 (1980).
(3) Mishra, U.C., Lalit, B.Y., Shukla, V.K. and Ramachandran, T.V., Standardized low level measurement methods for environmental studies, (In) Proc. of IAEA Symp. on "Methods of Low Level Counting and Spectrometry", IAEA-SM-252/66, IAEA, Vienna, 189-199 (1981).
(4) India 1982, A reference annual, Chapter 1, Land and people, Publications Division, Govt. of India, New Delhi, 1-11 (1982).
(5) UNSCEAR, Sources & Effects of Ionizing Radiation, Annex B: Natural sources of radiation, United Nations, New York, 35-114 (1977).
(6) Hamilton, L.D., IAEA Bulletin, 22, No.5/6, October, 35-77, (1980)
(7) UNSCEAR, Ionizing Radiation: Sources and Biological Effects, Annex F: Exposures resulting from nuclear power production, United Nations, New York, 249-331 (1982).

Details of Power Station & Date of Sampling	Sampling Location	Radial Distance from Centre & Direction Km	Dose Rate μR/hr	Details of Power Station & Date of Sampling	Sampling Location	Radial Distance from Centre & Direction Km	Dose Rate μR/hr
Nasik TPS	Power Stn.	-	7.0	Neyveli TPS	Power Stn.	-	9.8
480 MWe	Chemistry Lab	0.5 NE	4.5	600 MWe	Fly-ash Yard	-	11.5
Stack ht.	Coal handl-			Stack ht.	Entrance		
80 M	ing area	1.0 S	12.0	20M, 60M	Gate	0.5 NE	8.2
June 1982	River Bank	1.0 NE	20.1	July 1983	Water treat-		
	Ash Pond-II	1.2 N	11.8		ment lab.	0.5 W	8.6
	Pump House	1.5 N	10.5		Ash Pond-I	1.0 E	16.9
	Guest House	2.0 SW	6.5		Ash Pond-II	1.5 SW	20.2
	Location at	2.5 S	12.6		High dump		
	Ash Pond-II	3.0 NE	19.5		yard	1.5 E	12.0
	Nasik Rly.				Ash Pond-I	2 WS	16.7
	Station	6.0 SW	7.0		Guest House	3.0 E	13.5
					C.T.Office	7.0 NE	7.4
Tarapur BWR				Kota HWR			
150 MWe	Power Stn.	-	18.6	150 MWe	RAC Campus	-	19.3
Stack ht.	Security			Stack ht.	Phase I		
110 M	Gate	1.6 SE	18.5	100 M	Colony	2 NE	21.9
Aug.1983	Kuden	3 NEE	9.4	Sept.1983	RPS Colony	6 W	20.8
	TAPS Colony	4 SE	8.8		Phase II		
	BARC				Colony	7 N	20.8
	Quarters	5 E	8.9				
	Chinchni	5 NEE	8.6				
	Kurgaon	8 E	7.0				
	Boisar	12 SE	6.8				

Table 4: Typical radiation doses observed at some power plants in India.

GRAPHICAL PRESENTATION OF STATISTICAL RISK PROJECTIONS

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The purpose of this work will determine if a simple graphical presentation of dosimetry data can be easily used quantitatively (statistically) to evaluate the population sub-group which contributes most heavily to total collective dose (person rem).

The method used is to plot individual dose of the exposed group on statistical graph paper (dose value vs. cumulative probability). A straight line is drawn through the points and the usual statistical parameters are identified; then, the smooth data is transferred to a log-plot of cumulative probability vs. dose for further graphical analysis to compare risk.

The results of using this graphical display and analysis allows 1) any easy method to determine the statistical parameters, 2) extrapolative determination and evaluation of low probability valves of dose and 3) determination of the population sub-group which contributes most to the total collective dose.

The author concludes that use of this technique for dosimetric analysis, using data collected from operating sites, provides an easily used mechanism to quantitatively and objectively evaluate the radiation exposure control program. This graphical tool is also an excellent method to communicate results and conclusions.

AGE DEPENDENT DOSE FACTORS FOR THE THYROID GLAND AND THE RISK RESULTING FROM THE USE OF STABLE IODINE TO PROTECT THE GLAND[+)]

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By ingestion of inactive iodine the uptake of radioactive iodine into the thyroid and by this the radiation exposure of the thyroid can be diminished. However, an increased iodine intake causes a series of local and systemic side effects. Before iodine will be used in radiation protection it is usefull to compare the biochemical risk resulting from the use of stable iodine and the radiation risk.

Biochemical risk

Generally, only the causation of hyperthyroidism or hypothyroidism is crucial by an increased iodine intake in greater groups of the population. The effects of iodine are dose-dependent. As a function of a continuous iodine intake per day the functional diseases of the thyroid are figured in a diagram (Figure 1).

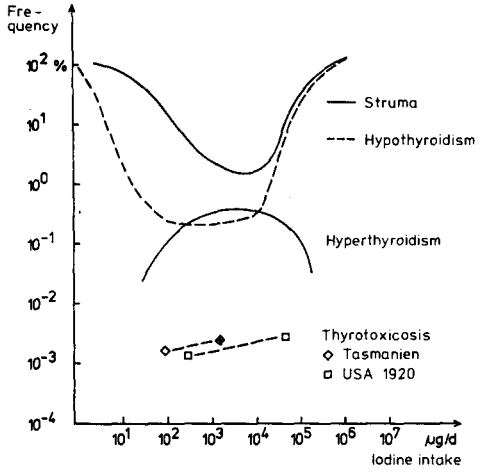

Fig. 1: Incidence of thyroid diseases as a function of the daily iodine intake

This figure shows, that there is a range of small disturbance of iodine intake between 1o and $1o^4$ ug per day and beginning with iodine quantities of more than 1o mg per day, changes in thyroid function have to be expected as complications. Concerning the incidence of hyperthyroidism and thyreotoxicosis we obtained from a study

+) with Support of the Bundesminister des Innern of the Federal Republic of Germany

in 7o hospitals in the Federal Republic of Germany the data which are set out in Table 1.

Disease	Incidence per year N	%
Hyperthyroidism	5966o	o.o96
of this with iodine contamination	15321	o.o25
Thyreotoxicosis	1o53	o.oo17
of this with iodine contamination	75o	o.oo12
of this with fatal outcome	375	o.ooo6

Tab. 1: Incidence of hyperthyroidism and thyreotoxicosis per year and in per cent of the population per year in the Federal Republic of Germany, estimated by GLÖBEL (1981), (inhabitants of the FRG: $62 \cdot 1o^6$)

As a result thyreotoxicosis is to consider as the actual crucial risk of increased iodine intake. For a short-term increased iodine intake of 1oo mg per day result the following risks:

hypothyroidism 2 : $1o^3$
hyperthyroidism 1 : $1o^4$
thyreotoxicosis 3 : $1o^5$

Radiation risk

The following effects of radiation must be considered as the main risks:

thyroid carcinoma
leucemia
lifetime shortening
hypothyroidism.

The somatic risks as a function of the radiation dose, as taken from literature, are shown in Figure 2.

Over the course of a lifetime the natural radiation exposure of the thyroid is about o.1 Gy. This is related to a spontaneous thyroid carcinoma rate in the Federal Republic of Germany of about o.2 % (OBERDISSE 198o). In the range between o.1 and 1o Gy only the carcinoma risk of the thyroid and the risk of hypothyroidism are to be into account as crucial effects of radiation. The radiation dose in the thyroid following incorporation of radioactive iodine is on the one hand dependent on the percentage of radioiodine uptake into the thyroid and on the other hand from the weight of the thyroid and on the biological half-life of iodine in the thyroid. Beside this the physical properties of the incorporated iodine isotope must be taken into account. In the average 55 % of incorporated iodine are stored in the thyroid in the Federal Republic of Germany. The radioiodine uptake in per cent as a function of the daily iodine intake is shown

in Figure 3.

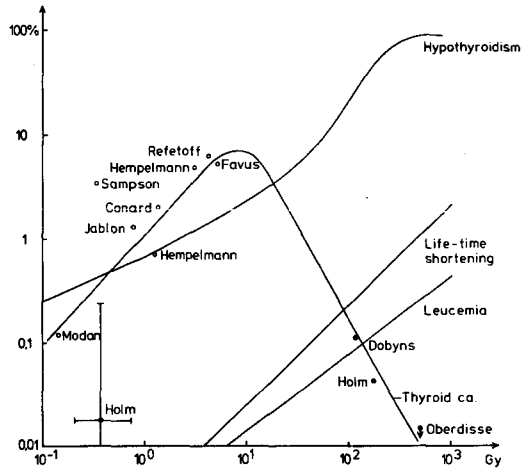

Fig. 2: Somatic risks as a function of radiation dose

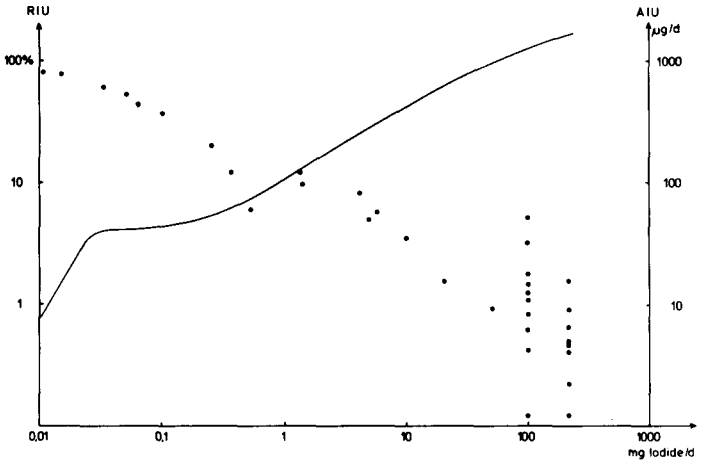

Fig. 3: Radioiodine uptake in per cent as a function of the daily iodine intake (NCRP 55, GLÖBEL 1981)

With regard to the age dependent biological half-life of iodine and the age dependent weight of thyroid it amouts per incorporated kBq ^{131}J the energy dose dependent on age set out in Figure 4.

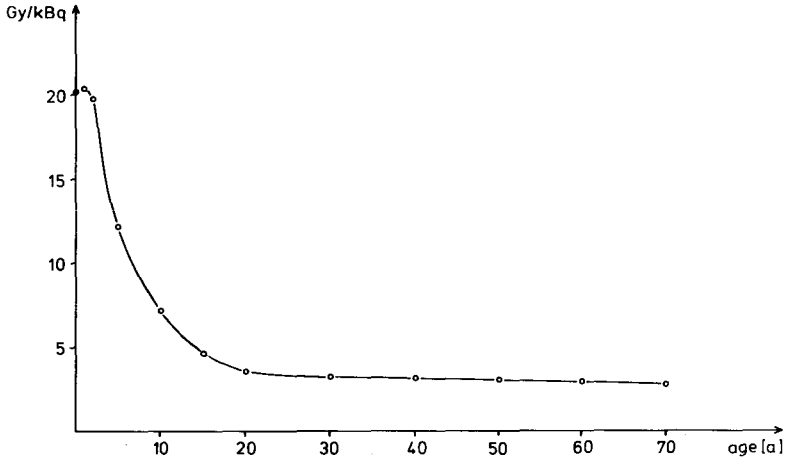

Fig. 4: Energy dose in the thyroid as a function of age

In case of an accident an intake of 1oo mg of stable iodine per day will diminish the uptake of radioactive iodine by a factor up to 1oo in the Federal Republic of Germany. Consequently in the range between o.1 and 1o Gy the risk to develop a thyroid carcinoma will be diminised also by the same factor.

Literature

(1) GLÖBEL, B., H. GLÖBEL, H. MUTH, E. OBERHAUSEN
Die Schilddrüsenfunktion bei normalem und erhöhtem Jodangebot als Grundlage für die Abschätzung des Risikos bei der Verwendung von Jodidtabletten im Strahlenschutz.
Zivilschutzforschung 12 (1981), Osang-Verlag, Bonn

(2) NCRP Report Nr. 55
Protection of the thyroid gland in the event or releases of Radioiodine. National Council on Radiation Protection and Measurements, Washington D.C.(1977)

(3) OBERDISSE, K., E. KLEIN, D. REINWEIN
Die Krankheiten der Schilddrüse
G. Thieme Verlag, Stuttgart (198o)

SOMATIC DOSE INDEX IN RADIOLOGICAL EXAMINATIONS

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INTRODUCTION

The risk involved in X-ray examination for an individual and a population can be described with different indices. For the determination of these indices doses resulting from the X-ray examination are needed for organs particularly sensitive to radiation. Epidemiological investigations show the thyroid gland, the lungs, the breasts and the bone marrow to be organs sensitive to radiation. The dose delivered to them is generally considered in calculating the somatic dose index, which is used to describe the somatic risk. The somatic dose index can be defined as an effective uniform whole body dose which has the same somatic detriment as the set of non-uniform doses absorbed by the individual organs /6/. When studying a somatic dose index, the other tissues can, in addition to the above-mentioned organs, be considered as an organ group of their own /6/, for which a whole body dose is used as an organ dose.

CALCULATION OF ORGAN DOSES AND SOMATIC DOSE INDEX

In the present study, the dose assessment is based on the experimental formulae for radiation output, backscatter factor, depth dose curve and off-axis scatter used in the literature /6,10/. The formulae for backscatter factor and off-axis scatter were corrected in order to get a better agreement with experimental results. The energy imparted for calculation of whole body dose is obtained by summing up the doses along the central axis as presented by Carlsson /2/. The addition to the energy imparted resulting from the radiation scattered outside the primary field, is corrected by multiplying by the coefficient 1.12. The coefficient has been obtained by comparing the energy imparted computed with this program with the energy imparted computed by the Monte Carlo method /13/.

With this program, doses can be computed in individual points or as organ doses, whereby the doses of the major organs are computed when needed by means of several measuring points as a weighted average. In the red bone marrow there are 18 measuring points and the weighting factors are determined according to bone marrow distribution for adults /4/ and for children /3/. The 10 measuring points of the lungs and the 6 points of the chest have been regarded as being of equal weight. The doses of the ovaries, the uterus, the testes and the thyroid gland are calculated with one measuring point. The co-ordinate system used in the calculation program is placed at the surface of the phantom closest to the X-ray tube in the direction of the exposure.

The adult sizes can be changed as a function of weight /12/. Of the adult co-ordinates, the depth co-ordinate alone is transformed, because the organ locations in the other two directions are assumed to remain the same and only the depth increase or decrease as compared with the basic co-ordinate system changes.

To determine children's radiation doses, the program transforms the organ co-ordinates according to body size and age. The vertical co-ordinate is transformed by means of the sitting height /8/, because the sitting height describes the change in the organ location better than the full height, since in childhood the growth of the lower extremities is more pronounced than that of the body. The other two co-ordinates are transformed by means of the mean circumference /12/. The thickness and width are computed from the circumference based on the child's age by approximating with an ellipse, whose axes are computed.

The somatic dose index is calculated by means of the formula /7/

$$I_D = \frac{\Sigma S_i \alpha_i D_i}{\Sigma_i S_i \alpha_i} \quad , \text{ where}$$

i indicates the organ, S_i the relative severity of somatic effect, α_i the risk coefficient for the effect in organ i and D_i organ dose. The somatic dose index for men is computed from the doses of the red bone marrow, the thyroid gland, the lungs and the whole body. For women the dose received by the breasts is also considered.

The lung tissue absorption, which is smaller than that of the other tissues, was considered in computing doses in the lung measuring points directly involved in the primary field.

The dose computation does not consider the dose decrease due to bones in the separate measuring points, because the locations and thicknesses of the bones are not determined in the co-ordinate system used. When determining the bone marrow dose, the absorption in the bone was thought to be offset by the approximately 10 % additional dose owing to the photoelectrons emitted from dense bone.

The calculated depth doses, backscatter factors and off-axis scatter were compared with experimental values presented in literature /1,5,14/ and the calculated organ doses, energy imparted and somatic dose indices with values calculated with the Monte Carlo method /7,11,13 /.

Table 1 shows the somatic dose indices of some roentgen examinations calculated by the dose program presented here and the Monte Carlo method /6/.

The somatic dose indices and organ doses, however, can be compared only for their order of magnitude, because, apart from the sources of error in the application of computation model functions, differences are caused by the exposure values and co-ordinates, which may differ from those used by Rosenstein.

Table 1. Somatic dose index for some X-ray examinations compared with the values calculated by Laws et al. /7/. Total filtration is 3.75 mmAl.

Examination	kV	mAs	Field area cm^2	FSD cm	Somatic dose index this study male mrad/R*	this study female mrad/R*	Laws et al. male mrad/R*	Laws et al. female mrad/R*
thorax AP***	120	5	30x30	180	223	539	243	520
thorax PA***	120	5	30x30	180	206	139	231	169
vertebrae** (th)	73	100	15x29	84	115	100	149	412[1)]
vertebrae** (lumbal)	70	80	17x21	82	45	29	73	39
pelvis**	72	63	29x23	75	69	36	57	30
hip**	72	52	12,5x18	90	24	13	24	12

*** HVL 4 mmAl
** HVL 3 mmAl
* dose indices are presented against 1 R measured free in air at fokus-skin distance mrad = 10^{-2} mGy
1) broad field

EXPERIMENTAL VERIFICATION

The doses obtained with the dose computation program were compared with the doses obtained from the X-ray examination simulations made with anatomical slice phantom measurements (Alderson-Rando phantom) and inserted LiF (TLD-100) thermoluminescence dosimeters. Measurements were made in pelvic, hip, pelvimetry and gastrointestinal examinations. The doses computed at the different phantom points for these X-ray examinations and the doses measured with the TL dosimeter at the corresponding points were compared with each other. The mean and S.D of the relation of the computed and measured doses were about $1.3 \pm (0.3 \pm 0.6)$ for all points. For the points middle in the primary field the agreement was still better.

The substantial differences in doses in some single measuring points are generally explained by the fact that the points are close to the field border, which even with a minor deviation in the definition of field location results in a major change in the dose computed due to the substantial dose diminution in the border area. Moreover, bones may reduce the measured dose, but the dose computation program does not consider the bone absorption.

DISCUSSION

For collective dose determinations the present calculation method can be regarded as sufficiently good, because the most noteworthy errors are made, not in dose calculations, but in incorrect or insufficient initial data, i.e. basic phantom model, exposure values, number of examinations, size, age and sex distribution of patients and so on. Another question is whether the energy imparted alone would be more effective in describing the

patients' radiation impact. Some factors support this method. During the fluoroscopic examination many parameters (kV, mA, field site and size, number of exposures, fluoroscopic time) may vary making it impossible to record all the data needed for organ dose calculation. The only exact method to determine the true patient radiation impact is to measure area exposure with a calibrated area exposure monitor and calculate the energy imparted. Besides, in some cases the whole body dose alone comprises a large part of the somatic dose index.

REFERENCES

/1/ Archer, B.R., Whitmore, R.C., North L.B., Bushong, S.C., Bone marrow dose in chest radiography: The posteroanterior vs. anteroposterior projection, Radiology 133, 211-216, 1979.

/2/ Carlsson, C., Determination of integral absorbed dose from exposure measurements. Acta Radiologica, 1/6, 1963.

/3/ Cristy, M., Active bone marrow distribution as a function of age in humans. Phys. Med. Biol. 26/3, 389-400, 1980.

/4/ Ellis, R., Healey, M., Schleren, B., Tucker, T., A system for estimation of mean active bone marrow dose, U.S. Department of Health. Education and Welfare Publication, HEW (FDA),1976.

/5/ Harrison, R.M., Backscatter factors for diagnostic radiology (1-4 mmAl). Physics in Medicine and Biology 27, 1465-1474, 1982.

/6/ Karila, K., Kettunen, E., The Exposure Pulses and Quality Control of Medical X-Ray Equipment (in Finnish), STL-B19, Institute of Radiation Protection, Helsinki, 1978.

/7/ Laws, P.W., Rosenstein, M., A somatic dose index for diagnostic radiology. Health Physics 35, 629-642, 1978.

/8/ Ojajärvi, P., The Adolescent Finnish Child. A Longitudinal study of the Anthropometry, Physical Development and Physiological Changes during Puberty. Thesis, Helsinki University, Department of Medical Science, 1982.

/9/ Rannikko, S., Tapiovaara, M., Determination of active bone marrow dose. In: Small radiation doses and their biological effects in radiological diagnostics, STL-B22 The Institute of Radiation Protection, Helsinki, 1978.

/10/ Rathjen, M., Ewen, K., v.Endt, H., Berechnung von Organdosen bei Röntgenaufnahmen and Röntgendurchleuchtungen. Röntgen Blätter 34, 363-370, 1981.

/11/ Rosenstein, M., Organ doses in diagnostic radiology U.S. Department of Health, Education and Welfare Publication HEW (FDA) 76-8030, 1976.

/12/ Size Chart for Childrens', Ladies' and Mens' Wear. Central Federation of Finnish Clothing Industries VATEVA (in Finnish), Helsinki, 1980.

/13/ Tapiovaara, M., Handbook of X-Ray Diagnostic Physics II; Patient Doses (in Finnish), STL-B54, Institute of Radiation Protection, Helsinki 1983.

/14/ Trout, D.E., Isodose Curves in a Phantom due to Diagnostic Quality X-radiation, Health Physics 33, 359-367, 1977.

EVALUATION OF THE RADIOLOGICAL RISK RESULTING FROM ROAD TRANSPORTATION OF TRITIATED WATER

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INTRODUCTION

The objective of this paper is introducing a methodology for the evaluation of the acceptability of certain conditions proposed for the road transportation of tritiated water in drums.

For such purpose, the acceptability criterion proposed by the Argentine Licensing Authority (1) has been adopted. This criterion consists in verifying that the points representative of the two values (the probability of incurring a given dose vs. the resulting effective dose equivalent) fall within the area qualified as acceptable in Figure 4.

DESCRIPTION OF THE METHODOLOGY

Considering the transportation conditions (total volume of liquid to be transported, tritium concentration in the liquid, probable meteorological conditions at the site and time of the accident), the radiological conditions resulting from the incident are evaluated (effective dose equivalent incurred by the most-exposed individual) and the corresponding probability of occurrence.(2)

The proposed methodology has been drafted in Figure 2. In order to evaluate the probable dose to be incurred by an individual potentially exposed to the consequences of the accident, for a given volume of liquid to be transported and as per the possible degrees of severity of the accident under analysis, an estimation is made of the volume of spilt liquid. For the purpose of calculating evaporation and taking into account considerations made by several authors (2,3), the spilt area was considered as independent from the characteristics of the road at the site of the accident. As a function of the meteorological conditions prevailing, the evaporation rate of the spilt liquid may be calculated per unit area. The total evaporation rate may then be established on the basis of the area determined above.

The concentration of steam in the air in the surroundings of the accident site, conservatively evaluated, along with the concentration of tritium in the transported water, are used for the calculation of the effective dose equivalent, admitting a certain period during which the exposed individual would remain at the site under study.

In order to quantify the probability (PT) of incurring a given dose for a hypothetical individual living in the vicinity of the accident site or travelling by the road where the accident occurs, the following items must be taken into account:

1. The probability for an individual to be involved in the radiological consequences of the accident (PI), which may be evaluated by means of the following equation(2):

$$PI = T\ d\ N\ ft$$

where:

T is the accident rate per kilometer travelled;

d is the length of the road implying consequences upon the individual under analysis;

N is the number of trips per year; and,

ft is the permanence factor (fraction of time during which the individual is thought to remain at the accident site)

d and ft will depend on the fact that the individual lives or travels in or by the area of the accident.

2. The probability of incurring a given dose (PH), which is a function of the degree of severity , of the area classification and of the meteorological conditions. The PH value may be evaluated as follows:

$$PH = Ps\ Pm$$

where:

Ps is the probability for the accident to occur in a given area and to have a given degree of severity; and,

Pm is the probability of occurrence of the various possible meteorological conditions at the site of the accident.

Therefore:

$$P_S = fs\ fp$$

where:

fs is the fractional occurrence of accidents of a given degree of severity; and,

fp is the fractional occurrence of accidents in areas with diverse populational density, defined for each degree of severity.

After an evaluation of PH and PI, it may be concluded that:

$$Pt = PH\ PI$$

Finally, on the basis of the two values: effective dose equivalent and probability for an individual to incur such dose as an

effect of the accident, the graph in Figure 1 establishes the boundaries for acceptability or non-acceptability of the proposed conditions, as far as the various representative points are contained within the acceptability area.

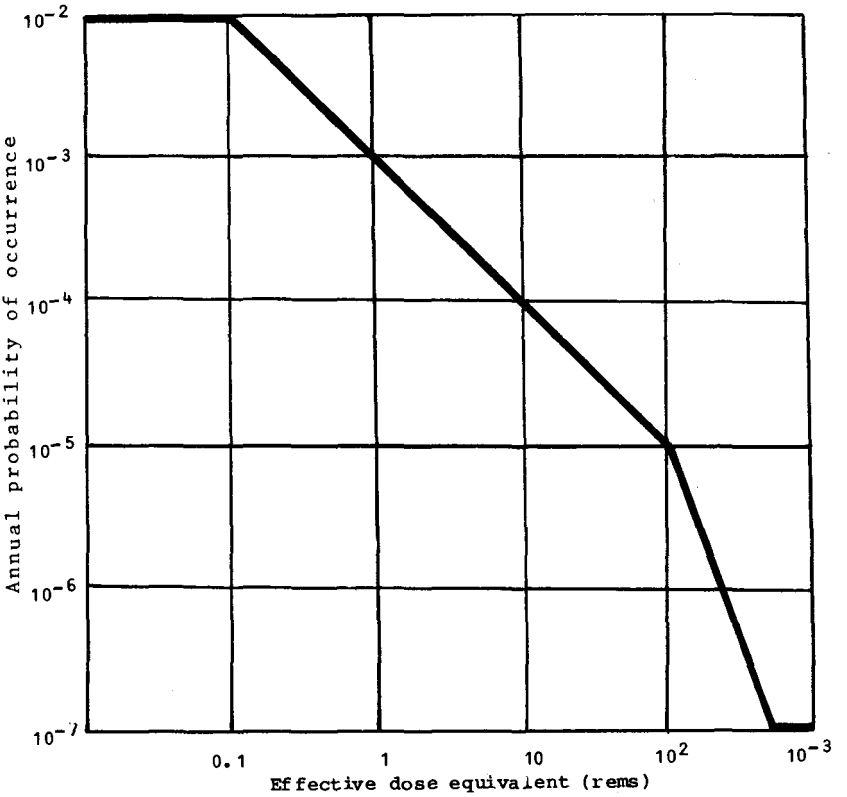

Figure 1. Acceptability curve (Ref. 1)

BIBILIOGRAPHY

1. Argentina, Comision Nacional de Energia Atomica. Norma CALIN 3.1.3.

2. Menossi, C.A., Segado, R.C. and Reyes, R. PATRAM 83 The International Symposium on Packaging and Transportation of Radioactive Materials, New Orleans, Louisiana, USA, 15-20 May, 1983. III-13

3. Gorman, D.J., Wong, K.Y. and Watson, D.A. 1980 BSA-80-3

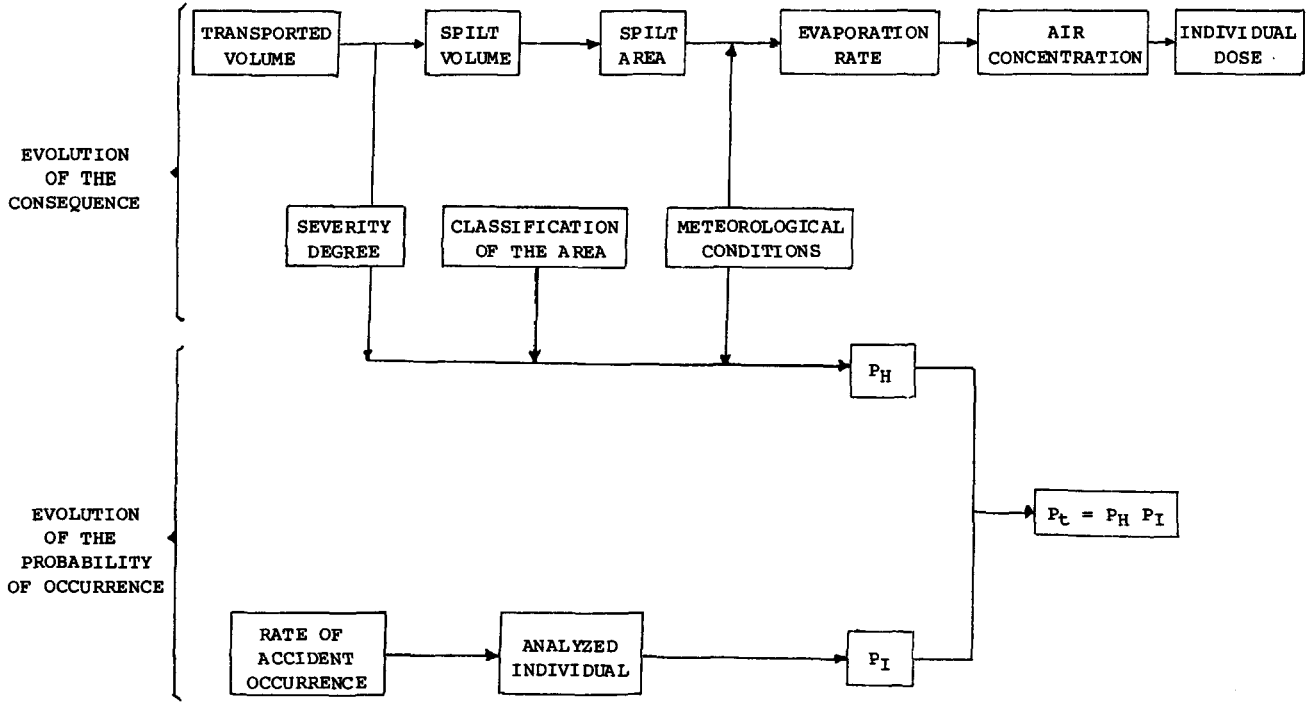

Figure 2. Flowsheet of the methdology applied

A COMPARATIVE PROBABILISTIC RISK ASSESSMENT OF NUCLEAR AND FOSSIL FUEL POWER PLANTS

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ABSTRACT

A methodology for comparing the adverse health effects of nuclear and fossil fuel power plants was developed. The methodology consists of: (a) Probabilistic Risk Assessments (PRA) of accidents and routine operation for both nuclear and fossil fuel power plants, resulting in CCDF(*) risk curves. (b) Analysis and consideration of uncertainties in risk curves, and (c) Methods, based on Utility Theory principles, for comparing risk curves and for safety oriented decision-making. Thus, the health effects of both types of power plants are presented on an equal basis.

The proposed methodology makes possible safety-oriented intercomparisons of power plants (nuclear vs. fossil) or different technologies of the same type of power plant located at any site. The methods developed in this study are not limited only to power plants, and can be implemented in any safety-oriented process of decision-making under conditions of uncertainty.

A. INTRODUCTION

Routine operational risks of nuclear power plants are extremely low (1,2). The risks of potential accidents in nuclear power plants have been extensively investigated but only lately using probabilistic methods (e.g. Refs. 3,4). The results of the probabilistic risk assessments (PRA), i.e. the magnitude of the various health effects (early fatalities, late cancer fatalities, thyroid nodules, etc.) have been presented by CCDF.

Unlike the risks of nuclear power plants, the risks of fossil fuel power plants were, in most cases, calculated using deterministic methods. To our knowledge, only in one study (5) were the health effects of routine emissions from fossil fuel power plants assessed by applying probabilistic methods in the consequence modelling. Moreover, in all the risk studies, only routine operational releases, and not possible accidents in the fossil fuel plants, were considered. Such accidents may arise either from fires (6) (mainly in the storage tanks of oil plants, or in the coal piles of coal plants), or from meteorological episodes that prevent the dilution of the pollutants. Such episodes have occurred, e.g. in London in 1952, where about 4000 fatalities were reported, due to a significant increase in the air concentrations of pollutants. In the present study, probabilistic methods are used (7) to assess the routine operational and accidental risks from both nuclear and fossil fuel power plants.

B. REQUIREMENTS FOR COMPARATIVE RISK ASSESSMENT OF NUCLEAR AND FOSSIL FUEL POWER PLANTS

A full safety-oriented comparison of nuclear and fossil fuel power plants requires that the risks of both types of power plants be presented on an equal basis (7). This requirement is fulfilled in the present study, by performing for both types of power plants, (a) an evaluation of both routine operational and accidental risks (b) probabilistic calculations of early and late fatalities(**)

(*) CCDF - Complementary Cumulative Distribution Function.

(**) Early fatalities are defined as those occurring within months after exposure to a pollutant, while late fatalities are those occurring within years after exposure.

presented in the form of CCDF risk curves.

Figure 1 shows an arbitrarily chosen example of early and late fatalities CCDF risk curves obtained in a study (7) on a 1000 MWe coal power plant (Fig. 1a) and a nuclear power plant (Fig. 1b) located at the same site in a densely populated area. In Fig. 1a the early fatalities result from possible meteorological episodes, whereas the late fatalities are caused by routine emissions. In Fig. 1b, both early and late fatalities are caused by severe accidents of low probability-high consequences.

C. COMPARISON OF THE RISK CURVES (7)

In order to compare CCDF risk curves, three main problems remain to be solved: (a) the uncertainties in the CCDF risk curves, (b) comparison of intersecting curves and (c) the different attitudes of the public towards various risk categories. The first two subjects are discussed below while all of them are discussed in detail in Ref. 7.

C1. The Uncertainties in the CCDF Risk Curves

The uncertainties in the risk curves are caused by uncertainties in almost all the submodels and parameters comprised in the PRA (7). Due to these uncertainties, a PRA performed for any plant-site combination (PSC) (*) results in a family of risk curves forming an "uncertainty band", rather than in a simple curve (7,8). The "width" of the uncertainty band may be defined as the difference or the ratio of the expectation values (μ values) of the upper (e.g. 95%) and the lower (e.g. 5%) CCDF risk curves. (Note: These expectation values can easily be obtained by integration of the CCDF risk curves over their whole range of consequences.) It has been shown (7) that, depending mainly on the specific population distribution patterns, the width of the uncertainty bands may vary from site to site for a given power plant and even from plant to plant for the same site. Figure 2 shows hypothetical risk uncertainty bands of 2 PSC: P_1SC-A (plant 1 at site A) and P_1SC-B (plant 1 at site B). Figure 2a shows a straightforward situation: for both the 95% and the 50% curves, site B is preferable to site A from the risk standpoint. However, Fig. 2b shows a more complicated situation: P_1SC-A seems to be preferable to P_1SC-B when comparing the 95% curves, but worse than P_1SC-B when comparing, say, the 50% curves. Such a situation must be considered in risk comparisons. It was suggested (7) that if the ratio $\mu_{B1}95\%:\mu_{A1}95\%/\mu_{A1}50\%:\mu_{B1}50\%$ is greater than 0.5, PSC-B will be preferred. If the ratio is less than 0.5, PSC-A will be preferred.

C2. Comparison of Intersecting Risk Curves

In some instances, the CCDF risk curves intersect each other. The problem is illustrated in Fig. 3. Curves A and B may represent the risk of early or late fatalities obtained in PRA of two PSC. Risk curve A seems to be less severe than risk curve B up to point M, but more severe than risk curve B beyond this point. Can one of the PSC curves be distinctly preferred over the other from a safety point of view, or is it a matter of different individual preferences? A similar issue has been dealt with in the field of economics, concerning portfolio analysis and selection (9). Here, similar curves represented the uncertainties in the returns (profits) of different investment alternatives. The solutions given in portfolio analysis were based on Utility Theory considerations. Applying this methodology to the intersecting PSC risk curves, two criteria have been developed (7): (a) the Risk Aversion Criterion (RAC): According to this criterion, if the integral of curve A is greater than that of curve B for any value of X (that is, subtraction of the cumulative area under curve B from the cumulative area under curve A, for any X, yields a positive number), then curve A is preferred over curve B by all "risk averters"(**), (b) the Mean Variance Criterion (MVC):

(*) Plant Site Combination (PSC) means a combination of a certain plant, located at a given site.

(**) A "risk averter" is a person whose "utility function" is concave. For detailed discussion, see Refs. 7,9.

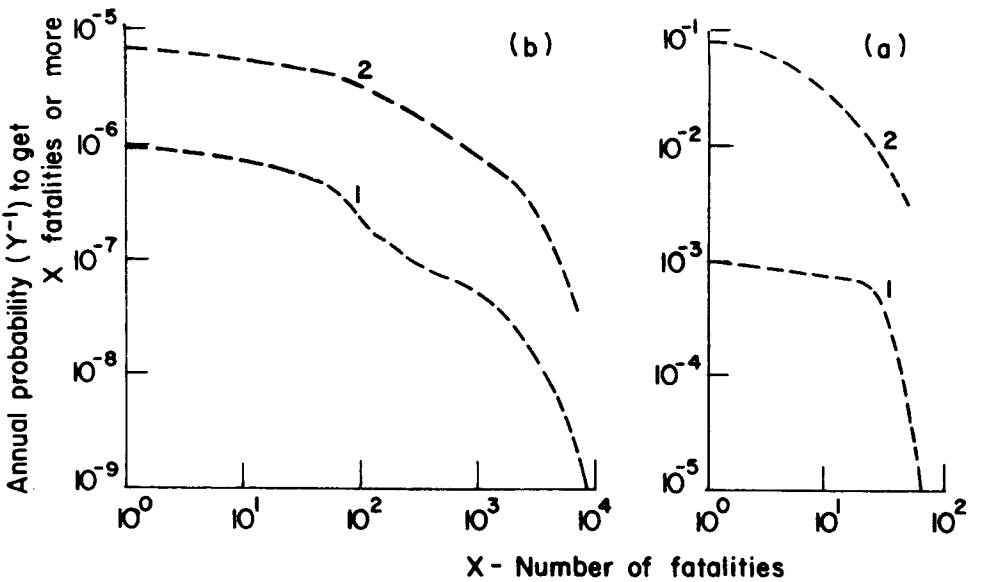

Fig. 1. CCDF risk curves for early and late fatalities caused by a coal-fired power plant (a) and a nuclear power plant (b) located at the same site (7). Curve 1 - early fatalities, Curve (2) - late fatalities.

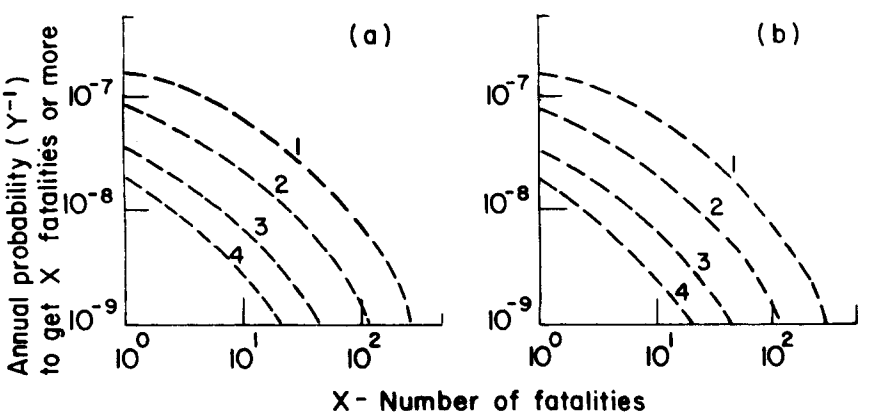

Fig. 2. Uncertainty bands of hypothetical CCDF risk curves for two plant site combinations, P_1SC-A and P_1SC-B (7). a) Straightforward case: 1.P_1SC-A 95%, 2. P_1SC-B 95%, 3. P_1SC-A 50%, 4. P_1SC-B 50%, b) Complicated case: 1. P_1SC-B 95%, 2. P_1SC-A 95%, 3. P_1SC-A 50%, 4. P_1SC-B 50%.

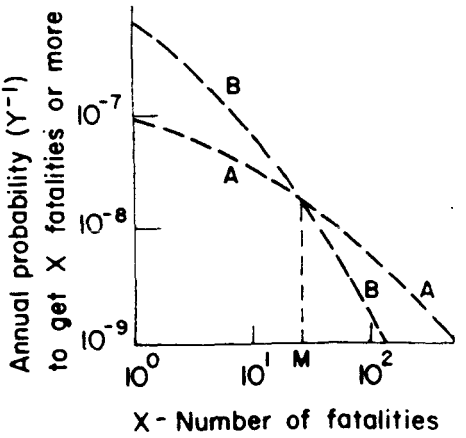

Fig. 3. Hypothetical intersecting CCDF risk curves.

According to this criterion, if both the expectation value and the variance of curve A are smaller than the corresponding values of curve B, then curve A will be preferred over curve B. Again, this holds for all "risk averters". Due to some mathematical weaknesses of the MVC criterion, it is recommended (7) that this criterion be applied only when the RAC criterion cannot be used (e.g. in cases where X values can be found for which the integral of curve A is not greater than the integral of curve B).

REFERENCES

(1) Bayer, A., The Radiological Exposure of the Population in the Rhine-Meuse Region by Nuclear Installations, during Normal Operation. Commission of the European Communities Report (June 1978).
(2) Baker, D.A., Peloquin, R.A., Population Dose Commitments due to Radioactive Releases from Nuclear Power Plant Sites in 1979, U.S. NRC Report NUREG/CR-2850 (PNL-4221) Vol. 1 (Dec. 1982).
(3) U.S. NRC - Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power PLants. WASH-1400 NUREG 75/014. (Oct. 1975).
(4) Gesellschaft fUr Reactor Sicherheit - Deutsche Risikostudie Kernkraftwerke. Verlag TUV Rheinland (Bonn, 1981).
(5) Granger-Morgan, M. et al., A Probabilistic Methodology for Estimating Air Pollution Health Effects from Coal-Fired Power Plants. BNL-23581 and Energy Systems and Policy 2, 3 (1978).
(6) Starr,C. and Greenfield, M., Public Health Risks of Thermal Power Plants. UCLA-ENG-7242 (May 1972).
(7) Stern, E., Risk Assessments of Various Types of Power Plants and a Methodology for Decision Making (from Safety Point of View) for their Building and Siting. Ph.D. Thesis, The Hebrew University of Jerusalem, Jerusalem (July 1983).
(8) Kaplan, S. and Garrick, J.B., On the Quantitative Definition of Risk. J. Risk Ana;ysis 1, 11-27 (1981).
(9) Levy, H. and Sarnat, M., Investment and Portfolio Analysis. John Wiley and Sons, N.Y. (1972).

INTERNATIONAL RADIATION PROTECTION ASSOCIATION

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RADIATION – RISK – PROTECTION

COMPACTS VOLUME II

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8 Radiation Protection Policies
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Edited by A. Kaul, R. Neider, J. Pełnsko,
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RESULTS OF THE GERMAN THOROTRAST STUDY*

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Thorotrast, trade name of an X-ray contrast medium used during 1930 to 1950, consists of a 24% colloidal solution of Thoriumdioxide stabilized by dextrine. After intravascular injection Thoriumdioxide aggregates are accumulated in the reticuloendothelial system (liver 59%, spleen 29%, red bone marrow 9%, calcified bone 2%, lungs .7%, kidneys .1%). The results of KAUL and NOFFZ (1) indicate the following mean tissue dose rate (based on 25% ml intravascular Thorotrast injection in a 70 kg person): liver 24 rad/y; spleen 70 rad/y; red bone marrow 9 rad/y; endosteal layer in bone 16 rad/y; main bronchi 13 rad/y; and kidneys .4 rad/y.

The German Thorotrast study was started in 1968. Names and addresses of more than 5000 patients who received angiography with Thorotrast (70% cerebral angiography) were obtained from records at different hospitals in West Germany. The control group was selected from patients who were treated in the same time in the same hospital but not injected with Thorotrast (Table 1).

Table 1. German Thorotrast Study.

Patients ascertained	Thorotrast	Control
Patients examined	878	660
Non-examined patients whose survival times were:		
(a) more than 3 years (confirmed causes of death)	1257	1168
(b) less than 3 years	1918	615
Non responders	190	1475
Patients not traceable	916	1233

* Supported by the Federal Ministries of Research and Technology and of the Interior (FRG) and by EURATOM

Patients whose findings and fate could be evaluated has been divided in two subgroups: the group of examined patients, who are invited for reexamination every two years (2) and the group of non-examined patients whose causes of death were established from medical records (3). Patients who died within the first three years after injection or hospitalization were excluded from the study. A large number of patients could not be traced as a result of the general circumstances of World War II. Other patients did not reply to the inquiring letter.

The causes of death (examined and non-examined patients) are listed in Table 2. The most outstanding observation in the Thorotrast group is the extremly high number of 246 cases of liver cancer (65% carcinomas and 35% hemangiosarcomas). 30% of the liver tumour patients suffered also from liver cirrhosis. The shortest latency period was 16 years (Fig.1). The accumulated tissue dose of the liver ranged from 200 rad to more than 1500 rad.

Myeloproliferative diseases (mostly classified as acute myeloid leukaemia) occurred in 27 patients of the Thorotrast group and in two patients of the control group. The shortest appearance time in leukaemia patients was 5 years. The calculated accumulated dose to red bone marrow ranged from 50 to 400 rad. A special clinical problem of the Thorotrast patients are the paravascular deposits which occurred in about 30% of the examined patients. Large Thorotrast

Table 2. Causes of death of examined and non-examined patients (combined).

Disease	Thorotrast n = 1689 of 2135	Control n = 1280 of 1828
Liver cancer	256*	2
Myeloproliferative disease	27	2
Chron. lymphatic leukaemia	2	2
Non-Hodgkin lymphoma	12	5
Sarcoma at injection site	1	-
Bone sarcoma	4	1 (?)
Lung cancer	40	34
Pleural mesothelioma	3	-
Kidney cancer	4	1
Liver cirrhosis	132 (+ 75)	43
Bone marrow failure	16	1
Other diseases	1192	1189

*75 combined with liver cirrhosis

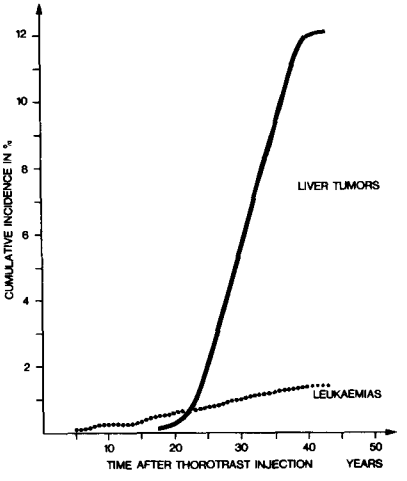

Figure 1: Cumulative incidence of liver tumours and leukaemias in examined and non-examined patients (n=2135)

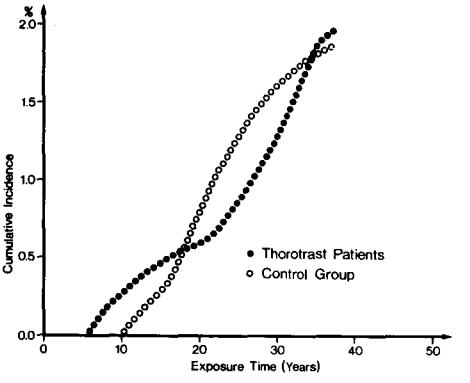

Figure 2: Cumulative incidence of bronchogenic carcinomas in examined and non-examined patients of Thorotrast (n=2135) and control group (n=1828)

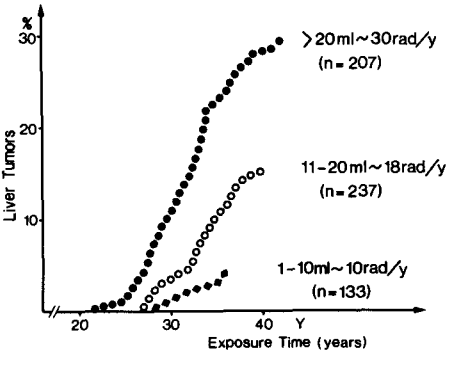

Figure 3: Cumulative incidence of liver tumours in examined Thorotrast patients with different liver dose rates

granulomas frequently cause different clinical symptoms, such as chronic inflammation, abscess, paralysis of nerves, thrombosis of veins. Only in one patient a malignant sarcoma had developed at the edge of a large thorotrastoma.

Four bone sarcomas have now appeared in the total of 2135 Thorotrast patients. These tumours are histologically confirmed (3 osteosarcomas and 1 chondrosarcoma of the femur and humerus resp.). These patients were injected at the age of 13, 16 and 23 (2x) years. One patient of the control group died from a vertebral bone tumour located near the skull.

Lung carcinomas occurred with about the same frequency in Thorotrast and control patients (Figure 2). This is surprising since the accumulated average tissue dose to the main bronchi ranged from 30 to 1400 rads. The ratio of smokers to non-smokers was equal in the examined group of Thorotrast and control patients. Among the fatal non-neoplastic diseases, liver cirrhosis is clearly predominant. Fatal aplastic anemia, agranulocytosis or thrombocytopenia occurred in 16 Thorotrast patients but in only one control patient.

Exact information of the injected volume of Thorotrast are obtained by measurements of the whole body counter and by the analysis of the X-ray films. Correspondingly the calculation of the dose effect relationship were carried out only in the group of examined patients. Thorotrast patients having X-ray projected paravascular deposits larger than 5 cm^2 were excluded from the evaluation. The Thorotrast patients were divided in three subgroups according to the measured radioactivity and the mean dose rate to the liver respectively.

The cumulative incidence of liver tumours of each group is presented in Figure 3, showing the different steepness of the curves depending on the dose rate in the liver. It is quite evident, that there is a dose-effect-relationship regarding the mean tissue dose to the liver. It was however not possible to assess a comparable dose-effect-relationship with regard to myeloproliferative diseases.

Literature:

(1) Kaul, A., Noffz, W. (1978): Tissue Dose in Thorotrast Patients. Health Physics 35: 113-121

(2) van Kaick, G., Muth, H., Kaul, A., Immich, H., Liebermann, D., Lorenz, D., Lorenz, W.J., Lührs, H., Scheer, K.E., Wagner, G., Wegener, K., Wesch, H. (1983): Recent Results of the German Thorotrast Study - Epidemiological Results and Dose Effect Relationships in Thorotrast Patients. Health Physics, Vol.44, Suppl.1, 299-306

(3) van Kaick, G., Lorenz, D., Muth, H., Kaul, A. (1978): Malignancies in German Thorotrast Patients and Estimated Tissue Dose. Health Physics 35: 127-136.

EPIDEMIOLOGIC STUDIES ON PATIENTS WITH IODINE-131 DIAGNOSTIC AND THERAPY[+)]

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Radiation can produce somatic and stochastic damages in the thyroid. We dispose of these experiences since many years after external, internal and mixed internal and external irradiation of the human thyroid. We got these experiences after external irradiation mainly in younger poeple. The results obtained in these studies are summarized in Figure 1.

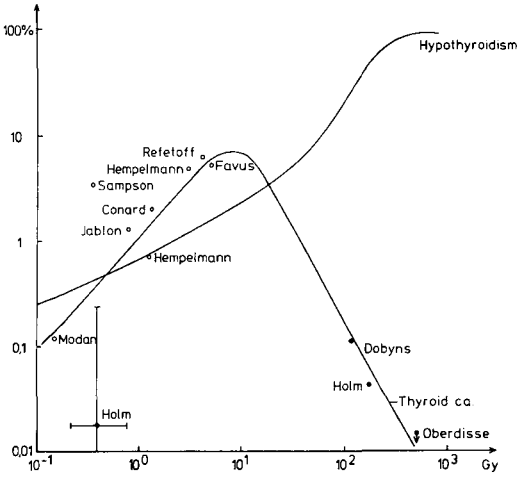

Fig. 1: Thyroid-cancer and hypothyroidism as a function of radiation dose.

The risk of hypothyroidism increases continuously from about $1o^{-1}$ Gy to $1o^{3}$ Gy. In the dose range of hyperthyroidism therapy it is about 8 % and in the range of cancer therapy it reaches almost 1oo %. According to the investigations of MODAN (1974), HEMPELMANN (1975), JABLON (1971), CONARD (197o), REFETOFF (1975), FAVUS (1976), SAMPSON (1974), who examined mainly the cancer-risk after external or mixed irradiation of the thyroid the cancer-risk increases almost linear between $1o^{-1}$ Gy and $8 \cdot 1o^{o}$ Gy and reaches a maximum near 1o Gy with an incidence of 8 %. These examinations include the atomic bomb survivors in Japan and at the Marshall islands. Above a dose of 1o Gy the cancer-risk decreases because there exists an increasing distruction of thyroid tissue (DOBYNS (1974), HOLM (198o) and OBERDISSE (198o)). HOLM performed studies in Sweden on a group of patients who incorporated ^{131}J for diagnostic and therapeutic purposes. The result of these investigations differs by a factor of 1o to 1oo from the risk, which was expected in comparison with the results of the

+) with Support of the Bundesminister des Innern of the Federal Republic of Germany

other authors. He found a smaller risk. If the lower dose rate after application of ^{131}J compared to the others groups of patients is the or the only reason of the lower risk is not still clear. Previous papers show that the frequency of thyroid cancer in a population can be influenced by many factors. The essential factors are: supply with iodine of the population, genetic assumtions, timing of observation, age distribution of patients in study, as well as the criterion of examination, for example morbidity or mortality from thyroid carcinoma or finally the histological classification (GLÖBEL (1983)). Since 25 years the radio-iodine uptake test was performed with ^{131}J in the Federal Republic of Germany. Therefore there exist a group of patients sufficent great in number to study how far the risk to produce thyroid cancer, after external and mixed irradiation, differs from that obtained by HOLM after incorporation of ^{131}J. Therefore we have contacted the patients examined at the nuclear medicine hospital of the University of the Saarland at Homburg, hitherto 15o79. Several dose groups could be formed: patients with one or more diagnostic application of ^{131}J and patients who incorporated ^{131}J for therapy. The study is not yet closed. We contacted 81 % of the patients, and we found 84 thyroid carcinomas. The distribution on the several groups is shown in the following table. In general there where used 2 MBq iodine-131 for one uptake test and o,2-2 GBq for therapy of hyperthyroidism.

	number of patients in study (N)	thyroid cancer incidence % mortality	% morbidity	(N)
unexposed poeple	-	o,21	o,6	-
one uptake test	611o	-	o,44	27
two uptake tests	3231	-	o,77	25
three uptake tests	2567	-	o,7o	18
four or more uptake tests	1988	-	o,5o	1o
therapy of hyperthyroidism	1183	-	o,34	4

Table 1: Incidence of thyroid cancer in patients who incorporated iodine-131 and in unexposed poeple.

The frequencies of thyroid cancers found in these groups of patients differ not essential from that frequency considered to exist in the unexposed population. With the results in Figure 1 we expected at least the tenfold number, if the risk factor after external and mixed irradiation of the thyroid was true for the case of internal irradiation. It seems that our results confirm those of HOLM. The examinations relative to the causation of thyroid hypofunction are not yet finshed and shall not be discussed here.

We believe, that the lower dose rate in the group of patients who incorporated ^{131}J compared with the groups exposed to external and mixed radiation is not the only reason for the smaller risk.

During the external irradiation of the thyroid in most cases the neck-head-area was for different reasons the target of a therapeutic use of x-rays.

The immunological system, the thymus gland in the younger age, was situated partially in the primary beam. According to the results of the last years it seems to be sure that disturbances of the immunological system can cause thyroid diseases, evaluated as auto-immune-diseases. For example the induction of goitre and nodular goitre. They lead in an increased percentage to thyroid cancer.

The immunological system and the whole body of the atomic bomb survivors had been exposed to radiation as well as the thyroid. The stress, to which these persons were also exposed, can cause impairments of the immunological system. Re-examinations of patients irradiated externally revealad in a considerable dimension thyroid hypofunctions and nodular goitres. We assume that the damage of the immunological system and the induction of auto-immune processes, are probably the reason of the increasing incidence of thyroid cancer after external and mixed in radiation. The application of ^{131}J exposes almost only the thyroid. The other tissue of the organism get only 1/2ooo of the thyroid dose. The frequent occurance of nodular mutation of the thyroid after external radiation exposure probably shows also, that the estimation of dose for the external and mixed irradiation is wrong. The given frequency of nodular mutation of about 1o % of the examined persons let us presume that the true radiation dose was higher than estimated in the publications.

Literature

(1) CONARD, R.A., DOBYNS, B.M. and SUTOW, W.W.
Thyroid neoplasies as late effect of exposure to radioactive Iodine in fallout.
JAMA 214, 2 (197o), 316-324

(2) DOBYNS, B.M., SHELINE, G.E., WORKMAN, J.B., TOMPKINS, E.A., McCONAKEY, W.M. and BECKER, D.V.
Malignat and benign Neoplasms of the thyroid in patients treated for hyperthyroidism.
J.C.E. u. M. 38, 6 (1974), 976-998

(3) FAVUS, M.I.
Thyroid cancer occurring as a late consequence of head-and-neck irradiation.
The Journal of Medicine 298 (1976), 1o19-1o25

(4) GLÖBEL, B., OBERHAUSEN, E. and GLÖBEL, H.
Häufigkeit von Schilddrüsenerkrankungen und einige Faktoren die sie verändern.
Schilddrüse 1983, G. Thieme Verlag, Stuttgart (in press)

(5) HEMPELMANN, L.H., HALL, W.J., PHILLIPS, M., COOPER, R.A. and AMES, W.R.
Neoplasms in persons treated with x-rays in infancy.
Journal of the National Cancer Institute 55, 3 (1975) 519-53o

(6) HOLM, L.E.
Incidence of malignant thyroid tumors in man after therapeutic and diagnostic doses of ^{131}J.
Department of General Oncology, Radiumhemmet Stockholm, Sweden (198o)

(7) JABLON, S.
Cancer in japanese exposed as children to atomic bombes.
Lancet (1971), 77o6

(8) MODAN, B., BAIDATZ, D., MART, H., STEINITZ, R. and LEVIN, S.C.
Radiation induced head and neck tumors.
Lancet I, (1974), 7852

(9) OBERDISSE, K., KLEIN, E. and REINWEIN, D.
Die Krankheiten der Schilddrüse.
G. Thieme Verlag, Stuttgart (198o)

(1o) REFETOFF, S., HARRISON, J., KARANFILSKI, B.T., KAPLAN, E.L., GROOT, L.J. and BEKERMAN. L.
Continuing occurence of thyroid carcinoma after irradiation to the neck in infancy and childhood.
The New England Journal of Medicine 1, (1975), 171-175

(11) SAMPSON, R.J.
Occult thyroid carcinoma in Olmsted County, Minnesota: prevalence at autopsy compared with that in Hiroshima and Nagasaki, Japan.
Cancer 34 (1974), 2o72-2o76

IMPACT OF LIFETIME RADIATION EXPOSURE AND OCCUPATIONAL HISTORY ON ESTIMATES OF MORTALITY RISK FOR SELECTED CANCER SITES AMONG HANFORD WORKERS

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We are conducting case-control studies of several cancer types among Hanford workers to examine the importance of lifetime radiation and non-radiation occupational exposure on risk estimates.

Studies of health effects, including causes of death, for workers in the nuclear industry have focused on cumulative radiation exposure. The radiation information usually has been limited to that available from a single facility while neglecting exposures received at other facilities. These studies often neglect to ascertain lifetime occupational histories which may be important in disease etiology.

The results of our studies and their implications for the epidemiology of cancer among nuclear workers will be discussed.

HEALTH EFFECTS AMONG PLUTONIUM WORKERS

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Concern about health effects that may be associated with plutonium has existed since its discovery. For example, Dr. Glenn Seaborg allocated 10 mg of the first 500 mg of plutonium for health effects studies in animals. These early efforts demonstrated that plutonium is absorbed poorly by the gastrointestinal tract, significant deposition occurs on bone surfaces and in the liver, and small amounts are excreted in urine and feces.[1] Subsequent laboratory studies have demonstrated excess tumors of the lung, liver and bone among plutonium exposed animals.[2] Investigations of human post mortem tissues have shown that principal concentrations of plutonium occur in the lung, liver, bone, and tracheobronchial lymph nodes.[3]

Health effects in humans have been demonstrated for a number of internal emitters. These include lung cancer among uranium miners exposed to radon-222,[4] liver neoplasms among patients who received thorium-232 (thorotrast),[5] leukemia among polycythemia vera patients who were treated with phosphorus-32,[6] thyroid tumors among Marshall Islanders exposed to iodine-131 as a result of atmospheric nuclear tests,[7] osteogenic sarcomas and sinus carcinomas in radium dial painters as a result of exposure to radium-226,[8] and bone sarcomas, leukemia and liver tumors among ankylosing spondylitis patients treated with radium-224.[9] Of these, the latter two have been the most useful for estimating health effects in humans due to plutonium exposures.

There are important differences between the biological behavior of radium and plutonium. For example, radium is absorbed from the gastrointestinal tract to a much greater extent than plutonium. Radium-226 deposits throughout the bone whereas plutonium deposits initially on the bone surface. Radium has produced bone sarcomas in exposed dial painters; however, the cumulative bone doses have been much greater than are estimated to be present in plutonium-exposed workers to date. Because of the differences between radium and plutonium, and concerns regarding extrapolating results from animal studies to humans, studies of potential effects in humans are necessary. The following studies summarize our investigations of workers who have been exposed to plutonium.

Human Studies of Plutonium Workers

In the early 50's a group of Manhattan Project workers with estimated systemic burdens ranging from 7 to 230 nCi were selected for clinical followup. Thirty-two years of follow-up with quinquennial clinical evaluations show no untoward health effects associated with plutonium deposition. These include clinical manifestations as well as mortality. Only two deaths occurred in these workers compared to four expected based on US male rates.[10]

Another investigation of Los Alamos workers included 241 who demonstrated systemic body burdens of 10 or more nCi. All of the 224 male and 17 females were contacted, or death certificates were obtained for those who were deceased. Life table analyses did not demonstrate excesses for any causes of death, and significant deficits were observed for total cancer, lung cancer and cardiovascular mortality when compared to US rates.[11] Although small in numbers, this group represents approximately one-third of all known plutonium exposed workers in the US with 25% or more of the maximum permissable body burden (40 nCi).

An investigation of cancer incidence among workers employed at Los Alamos National Laboratory from 1969-1978 has also been completed.[12] Incident cancers were identified from records of the New Mexico Tumor Registry and compared with rates for the State of New Mexico. No significant excesses were found for any site specific cancers or for all cancers combined for males or females. For Anglo males (non-Hispanic white), significantly fewer than expected cancers were observed for all cancers combined, primarily because of marked deficits for smoking related neoplasms. For example, whereas 24.2 lung cancers were expected, only 6 were observed, and whereas 7.9 oral cancers were expected, only 2 were observed. Among females, no smoking related cancers were observed. However, slight excesses were discovered for breast and uterine cancers, although they were not statistically significant. In general, the distribution of cancers among Los Alamos workers appears similar to that found among high socioeconomic groups.

In response to a three-fold excess of melanoma reported for Lawrence Livermore workers, an investigation of melanoma incidence[13] and a case control study of melanoma among Los Alamos workers employed from 1969-1978[14] were conducted. No significant excesses of melanoma were found. Also, no differences were found between cases and controls for plutonium body burdens, cumulative external radiation exposure, or type of employment. However, it was found that cases were more likely than controls to be college graduates. This is consistent with other studies that suggest increased risk of melanoma among higher socioeconomic groups and also suggests that lifestyle characteristics may be important risk factors for melanoma.

Recent studies of standardized mortality and relative risk among workers at the Rocky Flats Plant have failed to demonstrate significant excesses for neoplasms of greatest concern for plutonium and other radiation workers.[15,16] White male workers employed between 1952 and 1979 show significantly fewer than expected cancers for all neoplasms, and lung cancers when compared to the US population. We discovered an unhypothesized excess of benign and unspecified neoplasms (ICD 8th Rev. Codes, 210-239). All were intracranial in location, but the death certificate information was inadequate to identify specific origin or, in some cases, whether malignant or benign. No bone cancers were found and other radiogenic cancers did not differ from expectation. Analyses of duration of employment and latency did not affect these results. Standardized mortality ratios (SMRs) among workers cumulatively exposed to 1 or more rem of external radiation were significantly higher than expected for benign and unspecified intracranial tumors. However,

SMR's for all other causes of death were lower than, or did not differ from, expectation. Similarly, no SMR's were significantly higher than expected for workers demonstrating systemic plutonium body burdens of 2 or more nCi.

When workers who experienced 1 or more rem cumulative whole body exposures were compared to those with less than 1 rem, no significant excess risks were observed. The relative risk for unspecified intracranial tumors was not significantly elevated, probably because of the small numbers of exposed and unexposed cases available for analysis. In the case of plutonium exposure, relative risks could not be calculated because no exposed cases were observed.

The excess of brain tumors that was observed led to conduct of a case control study designed to investigate differences in exposure between brain tumor cases and three sets of controls selected from the same worker population. No significant differences were observed between cases and controls for cumulative external radiation exposures, plutonium exposures, or nonradiation occupational hazards.(17)

Conclusion

Although it is still too early to arrive at definitive conclusions regarding untoward health effects in humans, especially cancer, that may be associated with exposure to plutonium, the results of our early investigations have all been negative. A variety of study methods and populations have been employed, as shown in Table I, that range from in-depth clinical follow-up of a

TABLE I

SUMMARY OF HEALTH EFFECTS STUDIES AMONG PLUTONIUM WORKERS

Study Population	Method	Results
Manhattan Project Workers	Clinical follow-up	No suspicious health problems
Los Alamos Pu Workers (>10 nCi)	Mortality follow-up	No excess mortality
Los Alamos Workers	Incidence follow-up	No excess incidence
	Case-control study of melanoma	No association with exposure
Rocky Flats Workers	Mortality follow-up	Excess of benign and unspecified brain tumors associated with >1 Rem
	Case control study of brain tumors	No association with any type of exposure

small, highly exposed population to follow-up of mortality among an entire cohort of nuclear workers. In all instances, deleterious effects have not been discovered to be associated with the plutonium exposure. However, the excess of intracranial tumors discovered among one cohort of workers was associated with external radiation exposure >1 Rem. This latter finding was not replicated in a case-control study where no differences in exposure to external radiation (or to plutonium or other occupational exposures) were found when brain tumor cases were compared to controls.

These investigations of plutonium workers have suffered from the small populations of exposed workers available for study and from the limited periods of follow-up present for the majority of workers. These shortcomings can only be overcome through continued observation and study.

REFERENCES

1. W. H. Langham, Health Phys. 22, 943-952 (1972).
2. W. J. Bair and R. C. Thompson, Science 183, 715-722 (1974).
3. J. F. McInroy, pp 249-270 in The Health Effects of Plutonium and Radium. W. S. S. Jee, Ed. JW Press, Salt Lake City, 1976).
4. V. E. Archer, F. E. Lundin and J. K Wagoner, Health Physics 25, 351-371, (1973).
5. G. VanKaick, D. Lieberman, D. Lorenz, et al., Health Physics 44, Supp. 1, 299-306, (1983).
6. B. Modan and A. M. Lilienfeld. Medicine, 44, 305-344 (1965).
7. P. R. Larson and R. A. Conard. BNL-42104 Brookhaven National Laboratory, Upton, NY, (1978).
8. R. E. Rowland, A. F. Stehney and H. F. Lucas. Radiat Res 76, 368-383 (1978).
9. C. W. Mays and H. Spiess, 16th Midyear Topical Meeting of the Health Physics Society CONF-83-101, (NTIS, Springfield, Virginia, 1983) pp. 159-166.
10. G. L. Voelz, L. H. Hempelmann, J. N. P. Lawrence and W. A. Moss, Health Physics 37, 445-577, 1979.
11. G. L. Voelz, G. S. Wilkinson, J. W. Healy and G. L. Tietjen, 16th Midyear Topical Meeting of the Health Physics Society CONF-83-101, (NTIS, Springfield, Virginia, 1983) pp 318-327.
12. J. F. Acquavella, G. S. Wilkinson, L. D. Wiggs, et al., 16th Midyear Topical Meeting of the Health Physics Society, CONF-83-101, (NTIS, Springfield, Virginia, 1983), pp 338-345.
13. J. F. Acquavella, G. L. Tietjen, G. S. Wilkinson, et al. Lancet; pp. 883-884 (April 17, 1982).
14. J. F. Acquavella, G. S. Wilkinson, G. L. Tietjen, et al. Health Physics 45, 445-477, (1983).
15. G. L. Voelz, G. S. Wilkinson, J. F. Acquavella, et al., Health Physics 44, Supplement No. 1, 1983, 493-503.
16. G. S. Wilkinson, G. L. Voelz, J. F. Acquavella, et al., 16th Midyear Topical Meeting of the Health Physics Society, CONF-83-101, (NTIS,Springfield, Virginia, 1983) pp 328-332.
17. M. Reyes, G. S. Wilkinson, G. L. Tietjen, et al. Los Alamos National Laboratory report, LA-9804-MS, (June 1983).

ENQUETE EPIDEMIOLOGIQUE FRANCAISE SUR LES MINEURS D'URANIUM : DIFFICULTES ET PROGRES

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Une enquête épidémiologique des mineurs d'uranium a été entreprise en France dans le cadre d'une collaboration étroite entre le Service de Médecine du Travail (COGEMA) et les Département de Protection Sanitaire et de Protection Technique de l'IPSN (C.E.A.)

Elle vise à étudier les causes de mortalité par cancer de cette population de mineurs, notamment le risque de mortalité par cancer bronchopulmonaire et à établir une relation éventuelle entre l'exposition au radon et à ses descendants et l'augmentation du taux de cancer.

CHOIX DE LA POPULATION ENTRANT DANS L'ETUDE

En France, les divisions minières les plus importantes sont la Crouzille (Haute-Vienne), la Vendée, le Forez (Saône et Loire) et l'Hérault. Ce sont soit des mines souterraines dans lesquelles travaillent les mineurs de fond, soit des mines à ciel ouvert où se trouvent les mineurs de jour. Les 2 types de mineurs sont retenus dans notre enquête. Le siège minier de l'Hérault n'étant ouvert que depuis peu de temps n'entrera pas dans notre étude. Les régions des 3 mines concernées sont des régions essentiellement agricoles, avec très peu d'exploitations industrielles et la population des mineurs d'uranium peut être considérée comme une population très stable..

L'exploitation de la première mine d'uranium débute vers 1947. Nous avons donc décidé d'inclure dans cette étude tous les mineurs ayant eu plus de 3 mois de fond et tous les mineurs jours, depuis l'ouverture des mines jusqu'au 1.1.1980. Un certain nombre d'étrangers (Italiens et Polonais surtout) ont, à des moments variés, fait partie des mineurs d'uranium. Nous sommes obligés de les exclure de notre enquête, sauf s'ils ont opté pour la nationalité française. En effet, certains sont restés très peu de temps, puis sont retournés dans leur pays et il nous est impossible de les suivre au niveau médical et de connaître leur date ou cause de décès. D'autre part, ils risqueraient de constituer un mauvais échantillon dans l'éventualité d'une comparaison à un groupe témoin constitué par la population française.

Actuellement, nous sommes en possession de plusieurs fichiers permettant de recenser la quasi totalité des mineurs d'uranium. Un premier fichier, établi par le Service de Médecine du Travail des Mines, a recensé tous les mineurs ayant eu plus de 3 mois de fond depuis l'ouverture des mines (1947) jusqu'au 1.5.1972. Il représente 2.443 sujets. Un deuxième fichier a permis de réunir tous les agents ayant le statut mineur (jour et fond) de 1967 à 1980. Il représente 5.280 sujets. Comme les 2 fichiers se recoupent pour une certaine époque (1967 à 1972) et que certains mineurs de fond déjà présents avant 67 le sont encore après 72, il est évident qu'un certain nombre de mineurs se retrouvent dans les 2 fichiers. La fusion des 2 fichiers est actuellement en cours. Elle permettra de déterminer le nombre exact de mineurs faisant partie de cette étude. D'autre part, les mineurs

de jours ayant quitté les mines avant 1967 ne figurent pas dans ces 2 fichiers. Un recoupement avec un fichier administratif est donc nécessaire pour retrouver la totalité des mineurs de jour.

LA DOSIMETRIE DES MINEURS D'URANIUM

La surveillance dosimétrique des mineurs d'uranium a évolué dans le temps.

De 1947 à 1956, il n'existait pas de surveillance systématique individuelle des mineurs. Les mesures de concentration de radon dans l'air étaient rares. Aussi dans le cadre de cette enquête une commission d'experts a essayé d'attribuer à chaque siège minier un niveau de concentration en radon et une valeur du facteur d'équilibre entre le radon et ses descendants.

Les informations ayant servi à cette estimation sont :

- quelques résultats de mesures de concentration de radon dans l'air
- les conditions d'aération des mines
- les conditions du gisement
- la méthode d'exploitation.

Ces informations sont complétées par l'interrogatoire des personnes ayant connu les conditions de travail de cette époque. A l'aide de cette méthode de travail, pour la période 1947-1956, à chaque agent ayant travaillé dans une mine donnée est affectée une exposition. Son application conduit à une estimation de la concentration moyenne des descendants à vie courte du radon variant entre 20 et 200 $\mu J/m^3$ (1-10WL) suivant le siège minier considéré (tableau 1).

Tableau récapitulatif des doses mensuelles moyennes (WLM)**
à affecter aux agents pour chaque siège
dans le cadre de l'enquête épidémiologique

ANNEE \ SIEGE	Henriette	Brugeaud	Margnac	Fanay-Sagnes	Ecarpière	Chapelle	Commanderie	Ementruère Edrillère Goriandière	Lachaux	Bauzot-Grury	La Faye Les Brosses	B.N.L.
De 1947 à 1950	10								1			
1951	10	2		2					1			
1952	10	2		5					1			
1953	10	5	5	5	5	5		5	1	1	1	1
1954	10	5	5	5	5	5		5	1	1	1	1*
1955	10	5	5	5	5	5		5	1	1	1	1
1956	1						début					

* Pour le personnel ayant travaillé au BN du 18.2.1954 au 5.3.1954, il faut ajouter 50 WLM

** Un ouvrier qui passe un mois de travail avec une concentration en énergie α de 1 WL reçoit une dose de 1 WLM.

A partir de 1956 en France un contrôle systématique des expositions au radon des mineurs d'uranium a été mis en place dans toutes les exploitations minières. Ce contrôle systématique des concentrations en radon dans toutes les mines permet d'obtenir, pour chaque individu présent dans la mine à un moment donné, un relevé individuel de l'exposition. A cette surveillance de l'exposition au

radon s'ajoute une surveillance systématique de l'irradiation externe et il est permis de penser que pour chaque mineur une surveillance dosimétrique relativement précise, tant pour l'exposition au radon que pour l'exposition externe, est disponible pour tous les agents ayant travaillé dans les mines après 1956. La surveillance systématique a entraîné une amélioration des conditions de travail. De plus les mines exploitées pendant les dix dernières années sont en moyenne moins riche en uranium que ne l'étaient celles exploitées pendant les années 1950 à 1956. A titre de comparaison, au cours des dernières années, la concentration en énergie potentielle α pour l'ensemble des mines souterraines variait entre 2 et 10 $\mu J/m^3$ (0,1 à 0,5 WL).

Toutes les données dosimétriques sont notées, année par année pour chaque mineur, sur une fiche épidémiologique individuelle. Cette fiche renferme également tous les renseignements administratifs nécessaires pour l'identification de l'agent et sa recherche ultérieure auprès d'organismes tels l'Institut de Statistiques et d'Etudes Economiques, la Caisse de Retraites des Mineurs... Elle indique également les antécédents professionnels éventuels du mineur.

RENSEIGNEMENTS CLINIQUES ET RECHERCHE DES CAUSES DE DECES

A partir des dossiers médicaux, le médecin du travail remplit une fiche clinique pour chaque mineur. Cette fiche indique, en dehors de quelques renseignements cliniques, surtout les habitudes tabagiques et la consommation en boissons alcoolisées des mineurs. Pour les mineurs encore en activité, ces renseignements seront obtenus par un interrogatoire direct. Pour les mineurs retraités ou déjà décédés, elles seront reprises à partir des dossiers médicaux.

La recherche des décédés a été possible grâce à une interrogation des fichiers de l'INSEE : sur un premier échantillon de 2.115 mineurs, 258 sont décédés, 1.547 sont vivants et 310 non retrouvés par l'INSEE. Les 15 % (310) de perdus de vue sont recherchés localement car une erreur dans la rédaction du nom ou dans le code de la commune de naissance suffit pour que le sujet soit rejeté. Nous pensons ainsi pouvoir identifier au moins 95 % des mineurs de cette première pré-étude.

La recherche de la cause de décès se révèle plus difficile. Si le décès intervient pendant la période d'activité professionnelle du mineur, la cause est relativement facile à retrouver. Si le décès intervient chez un mineur retraité (au-delà de 55 ans), certaines causes sont connues par le médecin du travail, notamment celles considérées comme maladies professionnelles. Les autres doivent être recherchées localement en interrogeant les médecins et les hôpitaux de la région. Il n'existe malheureusement en France aucune liste officielle permettant de retrouver, pour un individu donné, sa cause de décès.

DISCUSSION ET CONCLUSION

L'enquête épidémiologique française sur les mineurs d'uranium soulève un certain nombre de problèmes notamment celui de la recherche des causes de décès. Cette recherche se fera uniquement grâce à une organisation locale qui ne peut être secondée par des fichiers nationaux tels qu'ils existent dans la plupart des autres pays. Tous nos efforts se porteront sur la confrontation d'un maximum de données fiables, en contactant les médecins généralistes des 3 régions considérées, les hôpitaux correspondants et les caisses de retraite et de sécurité sociale concernées. Dans certains cas, l'interrogatoire de la famille du défunt sera peut-être nécessaire.

En comparant, pour une même époque (1947-1971) le nombre de mineurs français à celui des mineurs américains [1], tchèques [2] ou canadiens [3] nous sommes en présence d'une population de mineurs relativement faible (2.115 mineurs de fond) mais qui constituent un groupe très stable. En effet, beaucoup de mineurs d'uranium présentent plus de 10 à 15 années d'activité minière. L'ouverture des mines ayant eu lieu dès 1947, la surveillance de certains mineurs couvre maintenant une période de 25 à 30 ans. D'autre part, dès 1956 un relevé dosimétrique individuel a été établi. Bien que la dosimétrie antérieure à cette époque soit limitée à des examens ponctuels de la concentration en radon, la connaissance des doses reçues est suffisamment précise pour que l'on puisse espérer en tirer des conclusions. Le taux de cancer bronchopulmonaire de la population exposée sera comparé soit à celui d'une population témoin de même répartition âge, constituée par la population nationale ou départementale, soit, si l'effectif des mineurs de jour est suffisant, à celui de cette dernière population. D'autre part, en fonction des données disponibles, nous étudierons la relation dose-effet entre l'excès de décès par cancer bronchopulmonaire de la population des mineurs et l'exposition cumulée, en tenant compte des autres facteurs de risque du cancer bronchopulmonaire, tel notamment le tabagisme.

REFERENCES

[1] ARCHER V.E., WAGNER J.K.
Lung cancer among uranium miners in United States
Health Physics, 25 : 351-371, (1973)

[2] SEVC J., KUNZ E. and PLACEK V.
Lung cancer in uranium miners and long term exposure to radon daughter products
Health Physics, 30 : 433-437, (1976)

[3] MULLER J., WHEELER W.C., GENTLEMAN J.F., SURAMPI G., KUSIAK R.A.
Study of mortality of Ontario Miners. 1955-1977, Part I.
Toronto. Ontario Ministry of Labour

A FOLLOW-UP STUDY OF PERSONS WHO HAD IODINE-131 AND OTHER PROCEDURES DURING CHILDHOOD

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Various studies have associated exposure of the thyroid glands of children to relatively low levels of x-ray with development of both malignant and benign thyroid neoplasms and a a linear dose-response relationship between x-ray exposure and thyroid neoplasms has been proposed (1-4). However, limited results available of studies of children exposed to radioiodine do not seem to fit the x-ray dose response curve.

Two studies have shown an association between radioiodine and the development of thyroid neoplasm. In the Atomic Bomb Casualty Commission (ABCC) study persons who received 50 rads had a relative risk for clinically recognized cancer of 3.8 and 8.8 for males and females respectively, compared to persons exposed to less than one rad (5,6). In the Rongelap study of 19 children exposed to a mixture of radioactive isotopes of Iodine with a dose range of 700 to 1400 rads, one child developed a malignant neoplasm and 18 children developed benign adenomatous nodules after 16 years (7).

There have been no reported epidemiologic studies of persons exposed to diagnostic levels of Iodine-131 (I-131) during childhood and the subsequent development of disease. The radiosensitivity of the growing thyroid is of special interest because of the potential exposure of large segments of the population to radiation from nuclear medicine procedures as well as the possibility of exposure to radiation dispersed to the environment by nuclear power plants. Children who received diagnostic doses of I-131 comprise one of the few irradiated human populations available for study with adequate data for calculation of doses coupled with long enough latent periods to expect any radiation related lesions to appear. In view of concern over the levels of safety represented by current limits for exposure to the population at large, study of this population was begun to strengthen the scientific basis for national standards for radioactivity. The risk of the development of thyroid lesions following exposure to I-131 will be evaluated with special reference to the shape of the dose-response curve and the evaluation of the Relative Biological Effectiveness of I-131.

The children under study were chosen from the records of patients receiving I-131 for diagnosis at one of 20 participating U.S. health centers during the years 1946 through 1967. Using center records patients were selected for the exposed group who:

1. received I-131 before January 1, 1968;
2. were less than 20 years of age at the time of the test;
3. were United States residents at the time of the test;
4. had no record of exposure to other isotopes;

5. had no previous irradiation of the head or neck;

6. had no exposure to I-131 elsewhere;

7. did not have a diagnosis of thyroid cancer or hyperthyroidism at the time of I-131 exposure;

8. did not have I-131 for other than thyroid evaluation;

9. had adequate medical and location information; and

10. had not died within 3 years of the test date.

A comparison group of patients seen at the same centers who had thyroid function tests other than I-131 or had a similar diagnosis as the exposed group and who satisfied the same criteria as the exposed group were group matched to the exposed group by age, race, sex and date of entry. A second comparison group composed of siblings of the exposed group, was chosen by selecting one sibling of the same sex and nearest age to the exposed patient where possible.

Demographic, medical, dosimetry and location information was abstracted from medical records of the study centers and private physician for the exposed group and the hospital comparison group.

Each study person in each study group was sent a questionnaire on their present health status and a brief health history.

Various follow-up procedures are being employed in an effort to locate and obtain a questionnaire from each of the study patients. Death certificates are obtained for persons reported to have died and pathology reports and specimen are obtained for those persons who indicated on the questionnaire that they had had thyroid or neck surgery since entry into the study.

A total of 14,860 potential study patients have been identified at 20 hospital centers throughout the United States. Medical records were located and medical abstracts completed for 14,310 of these. From the medical records it was determined that 9,830 patients satisfied the criteria for and were entered into the questionnaire follow-up phase of the study. Of the patients eligible for follow-up, 5,568 were in the exposed group and 4,262 in the hospital comparison group. All of these were entered into the survey. During the course of the study approximately 1,200 sibling patients were identified and also entered into the survey. The overall response rate to date for the two hospital groups is 5,777 or 59%. This includes 342 refusals which yields a questionnaire response rate of 55%. Location and follow-up is continuing for those persons for whom a response has not been obtained. The response rate for the sibling group is approximately 30%. The anticipated overall response rate for those entering the follow-up survey is about 88%.

Assuming this response rate, the resulting study group will provide statistical power of .63 to detect a 3-fold increase in risk for a 5% significance test.

The results of preliminary analysis of the incomplete, unedited and unverified data are presented. The analysis is premarily to examine the selection variables (age at entry, sex, race, and year of entry) with regard to balance between the two hospital study groups. The distribution of religion and calculated dose to the thyroid are also examined and probable thyroid neoplasm cases are identified. The distribution of age, sex, race, and religion at entry are similar for the two study groups. The greatest imbalance occurs in the entry year.

The radiation dose to the thyroid from each I-131 uptake examination was tenatively calculated using the formula,

$$D = 90 \ \frac{U \ A}{M} \ \text{rad},$$

where U = percent uptake of I-131 by the gland, A = microcuries of I-131 administered, and M is the estimated mass of the gland (8).

Using this formula, the total dose to the thyroid ranged from 0 rads to 2,000 rads with a mean and median dose of less than 50 rads. The following twenty-seven thyroid surgeries were identified from questionnaires data;

1. 16 thyroid surgeries without diagnosis,
2. 3 operations for goiter,
3. 2 operations for Hashimotos Disease,
4. 2 operations for thyroid tumor,
5. 1 operation for thyroid cancer, and
6. 3 operations for a growth in the throat.

All reported thyroid surgeries will have the diagnosis verified and classified by a special panel to evaluate available pathology.

References:

1. Simpson C., Hempelmann L., Fuller L.: Neoplasia in children treated with x-rays in infancy for thymic enlargement. Radiol 64:840, 1955.

2. Pifer J., Toyooka E., Murray R., et al: Neoplasms in children treated with x-rays for thymic enlargement. 1. Neoplasms and mortality. J Nat Cancer Inst 31, 6:1333, 1963.

3. Hempelmann L., Pifer J., Burke G., et al: "Neoplasms in persons treated with x-rays in infancy for thymic enlargement. A report of the Third Follow-up Survey." J Nat Cancer Inst 38: 317, 1967.

4. Hempelmann L.H.: Risk of thyroid neoplasma after irradiation in childhood. Science 169:159, 1968.

5. Wood J.W., Tamagaki H., Neriishi S., et al: Thyroid Carcinoma in Atomic Bomb Survivors, Hirochima and Nagasaki. Amer J. Epidimiol 89:4, 1969.

6. Sampson R.J., Oka H., Key C.R., et al: Metastasis from Occult Thyroid Carcinoma. An Autopsy Study from Hiroshima and Nagasaki, Japan. Cancer 25:803, 1970.

7. Conrad R.A., Dobyns B.M., Sutlow W.W.: Thyroid Neoplasia as Late Effects of Exposure to Radioactive Iodine in Fallout. JAMA 14:316, 1972.

8. Hine G.J., Brownell G.L.; Radiation Dosimetry Academic Press Inc., New York, 1956.

THE USE OF DOSES FROM PERSONAL MONITORING SERVICES FOR EPIDEMIOLOGICAL PURPOSES

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Introduction

Epidemiological studies of groups of workers who are occupationally exposed to radiation may offer direct evidence on the effects of low doses of ionising radiation on human populations. Information on the doses to which these workers have been exposed will usually come from programmes of personal monitoring. One of the main aims of personal monitoring is to demonstrate compliance with dose limits and dose recording conventions are sometimes used which can seriously distort dose histories. This paper discusses some of these problems and describes ways in which estimates of dose can be improved.

The conventions which can distort dose histories are given below

Notional Doses If a dosemeter is lost or cannot be assessed for some other reason many dosimetry services assign a pro-rata notional dose. This is the fraction of the dose limit appropriate to the period for which the missing dosemeter should have been worn. The effect is to reduce the dose limit pro-rata to the period for which measured doses are available. Notional doses should not be confused with "estimated doses" which are realistic estimates of missing doses based, for example, on the experience of other workers.

Threshold Doses When the dosemeter assessment process does not indicate a dose above the threshold of detection, some personal monitoring services enter this "threshold dose" into the dose record.

Recording Levels It has been suggested, for example in ICRP publication 26, that it is unnecessary to record doses below some prespecified recording level. ICRP now suggest that the recording level should be 10% of the fraction of the dose limit for the period in question. 30% had also been suggested.

Illustration

The effect of notional doses, threshold doses and recording levels are illustrated in Table 1. The data refer to a group of 513 classified workers on the National Radiological Protection Board's dose record keeping service for the whole of 1980. More details of this group, will be found in Kendall et al, 1982, and Darby et al, 1982. The importance of the three dose recording conventions will of course depend on the particular group involved. Nevertheless we believe that data presented here are not atypical.

The importance of notional dose depends on the frequency with which dosemeters are lost or cannot be assessed for some other reason. Unless the loss rate is very small, it is likely that notional doses will be important. The data in Table 1 indicates that notional doses increased the recorded annual average dose by about 50%. In fact this underestimates the problem since one criterion for inclusion in the group was that the individual had lost no more than two dosemeters in the year. For classified workers as a whole, notional dose increased the annual dose by almost 80%.

	Number of individuals in each dose group (doses in mSv)								average dose mSv
	0.0-5.0	5.01-10.0	10.01-15.0	15.01-20.0	20.01-30.0	30.01-40.0	40.01-50.0	50.0+	
measured doses only	413	49	20	10	14	4	2	1	3.6
measured doses and notional doses	354	74	39	13	22	8	0	3	5.5
Threshold doses									
set to 0.1 mSv	391	59	30	9	16	5	1	1	5.7
set to 0.2 mSv	0	434	35	18	18	5	1	2	8.1
Recording level									
set to 0.2 mSv	421	43	18	13	11	4	2	1	3.3
set to 0.6 mSv	433	36	18	11	8	4	2	1	2.9

Table 1: The effect of notional doses, threshold doses and recording levels on the doses incurred by a group of 513 workers on two-weekly monitoirng.

Table 1 shows the effect of simulating threshold doses of 0.1 and 0.2 mSv. Even these low thresholds increase the recorded annual doses greatly. Most workers are in the lowest dose category and it is here that the effect is the greatest. Those individuals with large doses are relatively little affected. There will be a minimum recorded annual dose, in this case twenty-six times the dose threshold.

Table 1 also shows the effect of simulating recording levels by setting to zero doses which are below 10% and 30% of the appropriate fraction of the dose limit. The effects on dose histories of recording levels are smaller than those of notional doses or threshold doses.

It is worth noting that threshold doses or recording levels must be used in all personal monitoring services since there will always be doses which are below the threshold of detection, but are still non-zero. These must be set to zero or to the threshold dose. The evidence presented here suggests that the former will give much more accurate dose histories.

Discussion

While it is possible that notional doses may occasionally underestimate the dose incurred by the worker it is probable that their inclusion generally increases the annual recorded dose significantly. Moreover, the effect on individual dose histories is not systematic, being determined by the number of dosemeters which have not been assessed. This is in contrast to threshold doses and recording levels which lead to a systematic distortion of the dose scale. Fortunately separate records of notional dose are generally kept. Estimates of dose will almost certainly be improved by setting notional doses to zero, or to the appropriate fraction of the measured dose.

If the dose threshold is low, the presence of threshold doses in dose histories may not be serious for epidemiological studies. For example, if the essence of the method is to compare age specific death rates in low dose and high dose groups then the qualitative conclusions may be unaffected by an increase in the mean dose to the low dose group. However, large inaccuracies in doses are undesirable and it is easy to imagine that this kind of error might cause serious problems in trying to fit specific dose-response models. Unlike notional doses, separate records of the contribution of threshold doses to the recorded dose are not generally kept. Three methods for improving estimates of dose have been suggested (Darby et al 1982). The simplest of these was to subtract the minimum annual dose appropriate to the issue period from all recorded annual doses. This led to significant improvements in the estimation of doses and has the useful property of ensuring that no dose is overestimated. Moreover it requires a knowledge of only the dose threshold and the number of dosemeters worn in a year. The other two methods for improving estimates of dose lead to more accurate answers, but require considerably more data.

We now investigate threshold doses and recording levels by considering the distribution of dosemeter assessments to each individual. The distribution of all dosemeter assessments for the group of 513 workers was found to be very skew, with most of the results close to zero but with a long tail (Darby et al 1982). The Weibull distribution was found to model the data significantly better than the log-normal distribution. It was possible to apply rigorous statistical tests to the distribution of dosemeter assessments for all workers. It is much less easy to investigate our present interest, the distribution of dosemeter assessments for an individual, since there are only twenty-six doses and most of these were generally recorded as zero. However, a number of simple tests suggest that this distribution is also very skew. Thus for 90% of the workers the median dose was zero, almost as frequently the standard deviation exceeded the mean and the mean exceeded the median.

The effect of threshold doses and recording levels on the doses attributed to workers was modelled assuming that the distribution of doses to an individual was Weibull. Expected values of annual doses were calculated by summing the contribution from all parts of the distribution above the threshold. To simulate threshold doses, doses below 0.2 mSv were set to this value. Recording levels were simulated by setting to zero doses below 0.2 mSv or 0.6 mSv. Table 2 shows the consequences of the model for the expected annual doses of workers on two-weekly monitoring. These calculations are illustrative since we do not know the precise dose-distribution curve. However, they are broadly consistent with the observations on real dose histories described above. They demonstrate that threshold doses lead to very much larger errors than do recording levels but that the simple correction described above gives greatly improved estimates of dose. These corrected estimates are always lower than the true dose. The model suggests that the deviation is relatively constant. Recording levels lead to relatively small underestimates of annual doses. Again, doses are underestimated by a relatively constant amount. It is probable that epidemiological studies would not be seriously jeoparised if annual doses were all underestimated by a small quantity.

These features arise from the skewness of the distribution of dose assessments. The mean dose is largely determined by the tail of the distribution; the main bulk of the density is too close to zero to contribute greatly. If the contribution from the bulk of the distribution is set to zero, the mean is little affected (as in recording levels). On the other hand if the bulk of the distribution is deemed to fall at a non-negligible threshold dose, the average dose is significantly affected. The precise figures given in Table 2 depend on the details of the model. However, the main conclusions will be true whenever

True Average Dose mSv	Dose Recorded if Recording Level		Dose Recorded if Threshold Dose 0.2 mSv	
	0.2 mSv	0.6 mSv	uncorrected	corrected
1	0.5	0.1	5.5	0.3
5	4.6	4.1	9.2	4.0
10	9.6	9.1	14.1	8.9
15	14.6	14.1	19.1	13.9
20	19.7	19.1	24.1	18.9
30	29.7	29.1	34.0	28.8
40	39.7	39.1	44.0	38.8
50	49.7	49.1	54.0	48.8

Table 2: Predicted effect of recording levels and threshold doses on annual doses attributed to workers on two-weekly monitoring if the distribution of individual dosemeter assessments is very skew (Weibull).

the distribution of dosemeter assessments to individuals is of the general form described above. Thus for example, a log-normal model gave similar results to the Weibull distribution.

Summary

a) Notional doses can seriously distort dose histories. They should be removed or replaced by the appropriate fraction of the measured dose.

b) Dose histories which are based on the use of recording levels like those discussed in this paper can probably be accepted as they are.

c) If dose histories contain threshold doses then a correction should be considered. Simply subtracting the minimum annual dose may well lead to very significant improvement.

References

Patterns of dose incurred by workers on the National Radiological Protection Board's Dose Record Keeping Service. I Annual Doses G.M. Kendall, S.C. Darby, E. Greenslade J.SRP 2 (3) pp 20-25 (1982).

Patterns of dose incurred by workers on the National Radiological Protection Board's Dose Record Keeping Service, II Individual Dosemeter Assessments S.C. Darby, G.M. Kendall, E. Greenslade J.SRP 2 (4) pp 31-38 (1982).

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EPIDEMIOLOGY AND CANCER RISK FROM LOW DOSES OF RADIATION

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Introduction

Although ionizing radiation appears to be the most thoroughly investigated single carcinogenic agent, it still poses a problem: the quantitative risk of malignant development following low radiation doses is not known with sufficient accuracy. Most disturbingly, some recent epidemiological studies[3,4,6] seem to suggest much higher cancer risk values per unit dose than the current risk estimates accepted by international and national organizations for radiation protection.[1,5,12] It is commonly stated that the effects of low-dose radiation cannot be demonstrated, on statistical grounds, in small populations. However, this statement is valid only on the tacit assumption that the actual risk is small and constant.

The epidemiological approach has sometimes been criticized on the grounds that it does not prove much because, despite careful design of data collection and analysis, there are often confounding non-random variables that affect the final results. However, the mere existence of 'confounding' factors does not necessarily mean that they are irrelevant to the problem because they could also modify the expression of radiation-induced damage. Development of cancer, initiated by radiation, is a complex process and other factors such as promoters may drastically affect the outcome of the primary effect. Thus there is no good a priori reason to assume that all reliable studies must end up with some universally valid risk estimate consistent with the high-dose studies, expecially those concerning A-bomb survivors. Interestingly, the credibility of the Japanese data base itself has become somewhat questionable because of the uncertainties in the radiation dose estimates.[8]

Population Size and Risk Estimates

To show that ionizing radiation is carcinogenic at low doses is clearly difficult epidemiologically. However, merely the finding that 10 mGy of low-LET radiation can transform cells in vitro with a frequency of up to 10^{-4} (see ref. 2) should make us cautious; remember that there are some 10^{12} cells in the human body that may be targets for malignant transformation. Yet, if the cancer risk estimates of ICRP or UNSCEAR are correct, it is virtually impossible to demonstrate, on statistical grounds, any excess cancer cases in human populations exposed to doses below 10 mGy. On the other hand, if an effect is manifested within a population of some 10,000 people, the current risk estimates may not be universally valid.

The size of a study population must naturally be large enough to allow the effect to be detected, if it exists, with reasonable probability. The size requirement of the population can be estimated by power calculations.[7] In the following we present another simple but illustrative method. The reasoning is a slight modification of that presented by Pochin.[9] Let us consider a population with

the following characteristics:

X = number of persons
p = natural cancer risk per year
y = follow-up period (years)
D = radiation dose (mean value)
k = radiogenic cancer incidence per unit dose during the follow-up period.

The minimum latency time, if relevant, can be subtracted from y. The parameter p is a function of y; however, for our purposes it is sufficient to know the value of the natural cancer risk, py, during the study period. The total number of observed cases will be

$$Xpy + XkD. \tag{1}$$

If the radiation effect is considered to be detectable when the number of excess cases is double as compared with the variability of the expected number, then

$$XkD = 2\sqrt{Xpy}, \tag{2}$$

and consequently,

$$X = \frac{4py}{k^2D^2} \tag{3}$$

From this equation we see, as is often stated, that if a radiogenic effect is just observed at 1 Gy, a population 100 times larger is needed at 0.1 Gy and a population 10,000 times larger at 10 mGy. Note that similar relationship holds also for k, the radiogenic cancer incidence. The plot of equation (3) for different values of p, the normal cancer incidence, is shown in Fig. 1. The family of curves is calculated for a dose D = 10 mGy and for a follow-up period y = 20 yr.

As an example the upper curve refers to a hypothetical follow-up study of radiation-induced breast cancer among middle-aged US women (cf. ref. 7). The curve shows that with the current risk estimates ($(2\text{-}20) \times 10^{-6}$ mGy^{-1}) the low-level radiation effect is only seen in a very large population of women (5×10^6 - 10^8). However, if an increased risk is detected after a low-dose exposure in a population of more reasonable size ($<10^5$ women) this may mean that among this population the expression rate of radiation-induced malignant transformation is high.

The lowest curve is a simulation of the leukaemia risk. We see that leukaemia induced by radiation is much easier to detect than most other types of cancer, because of the high "signal-to-noise" ratio (low natural incidence of leukaemia). In the "Smoky" study[3,11] 3224 men participated in military maneuvers during a nuclear test explosion; the mean dose to the entire cohort was 4.7 mGy and 9 leukaemia cases were observed, whereas only 3.5 were expected. This statistically significant finding is possible if the radiation-induced leukaemia risk is over 100×10^{-6} mGy^{-1} (Fig. 1 or eq. (3)). Of course the "Smoky" study, by itself, does not prove that development of the radiation-induced primary damage to clinical leukaemia has been more common among these men than, say, among A-

bomb survivors, but such a possibility is not readily excluded either.

Discussion

The quantitative risk values extrapolated from high radiation doses vary greatly, depending on which kind of dose-response curve is assumed. The linear model is the one most commonly used, but the quadratic and linear-quadratic models also have their advocates. Further complications are introduced by the notion that the risk for radiation-induced cancer does not depend on dose only: biological variables - such as age, sex, genetic susceptibility, and exposure to other environmental carcinogens, co-carcinogens, promoters, etc. - may drastically modify the expression of the primary damage.

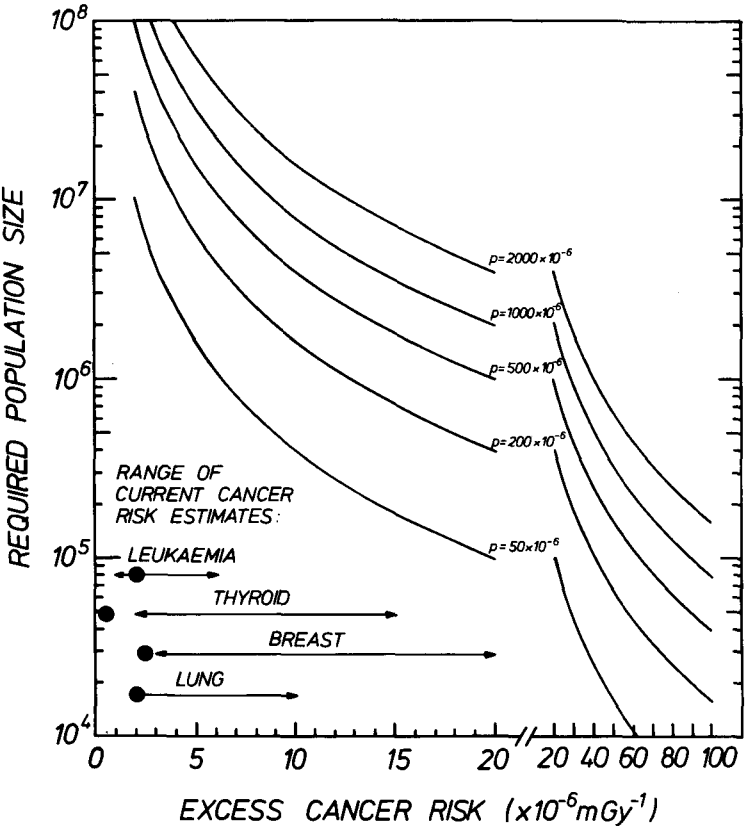

Fig. 1. Required population sizes in epidemiological studies as a function of excess cancer risk. The family of curves is calculated from eq. (3) with a radiation dose of 10 mGy and a follow-up period of 20 yr. The parameter p is the natural cancer incidence (see text). The black circles refer to the ICRP risk estimates;[5] the range of risk values is given by Pochin.[10] The curves may be used at all doses by properly selecting the ordinate; e.g. at 30 mGy the values quoted on the Y axis should be reduced by a factor of c. 10.

In conclusion, we suggest that the current risk estimates at low doses should be carefully reconsidered in the light of all epidemiological studies. It is not really justified to disregard one set of studies merely because the results are at variance with another set of data.

References

1. Committee on the Biological Effects of Ionizing Radiations, 1980, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation" (BEIR III).
2. Borek C., 1980, "X-ray-induced in vitro neoplastic transformation of human diploid cells", Nature 283, 776.
3. Caldwell G.G., Kelley D.B. and Heath C.W., 1980, "Leukemia Among Participants in Military Maneuvers at a Nuclear Bomb Test", JAMA 244, 1575.
4. Gofman J.W., 1979, "The Question of Radiation Causation of Cancer in Hanford Workers", Health Phys. 37, 617.
5. The International Commission on Radiological Protection, 1977, ICRP Publication 26.
6. Kneale G.W., Stewart A.M. and Mancuso T.F., 1978, "Re-analysis of Data Relating to the Hanford Study of the Cancer Risks of Radiation Workers", in "Late Biological Effects of Ionizing Radiation", IAEA, International Atomic Energy Agency, Vienna.
7. Land C.E., 1980, "Estimating Cancer Risks from Low Doses of Ionizing Radiation", Science 209, 1197.
8. Loewe W.E. and Mendelson E., 1981, "Revised dose estimates at Hiroshima and Nagasaki", Health Phys. 41, 663.
9. Pochin E.E., 1976, "Problems Involved in Detecting Increased Malignancy Rates in Areas of High Natural Radiation Background", Health Phys. 31, 148.
10. Pochin E.E., 1978, "Why be Quantitative in Radiation Risk Estimates", Lauriston S. Taylor Lectures in Radiation and Measurements, Lecture No. 2, NCRP, National Council on Radiation Protection and Measurements.
11. Toohey R.E., Rundo J., Essling M.A., Sha J.Y., Oldham R.D., Sedlet J. and Robinson J.J., 1981, "Radioactivity Measurements of Former Military Personnel Exposed to Weapon Debris", Science, 213, 767.
12. United Nations Scientific Committee on the Effects of Atomic Radiation, 1977, "Sources and Effects of Ionizing Radiation", UNSCEAR 1977 Report.

SOCIETAL AND ECONOMICAL CONSEQUENCES OF EVACUATION IN CASE OF ACCIDENTAL RELEASE: A POSSIBLE APPROACH

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The study presented is carried out with the objective of quantitative evaluation of the societal and economical risks of an evacuation after a radiological accident on a PWR.

Indeed, the cost of consequences on the economic activity of the region has to be added to the direct cost of the evacuation itself, correlated to transportation, relocation and protection of the populations and the real estates.

Our purpose is to obtain a complete investigation of the problems, and to define as realistically as possible a methodology, a nuclear plant was chosen : Gravelines in the North of France.

In this region around the nuclear plant, the major economic activities are encountered: agricultural areas, industrial activities in a specific area, sea-borne trade (Dunkerque harbour) and touristic business of the beaches between the site and Calais.

Taking into account the nature of the atmospheric release, the meteorological conditions and the dynamics of the accidents, the consequences of the counter measure (evacuation) are shown and their degree of importance are discussed as an important factor for decision making.

D. 28

THE VARIOUS WAYS OF INTEGRATING INDIVIDUAL DOSE EQUIVALENT OR TIME INTO THE OPTIMIZATION PROCESS (+)

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The dose limitation system recommanded by the ICRP in its Publication 26 /3/is based on three principles : justification, optimization and limitation of individual dose equivalents. This system has shown its validity in various ways since it was worked out. However many people involved in radiation protection wish to enlarge the framework of the optimization principle in order to have it better fit to practical needs. This wish is related to different problems such as : taking into account per caput dose distribution ; length of exposure integration ; uncertainty ; accidental exposure ; public or occupational exposure ; national or international exposure ... As a matter of fact the ICRP Publication 26 did not intend to take into account these differences. This is why the ICRP Publication 37 /4/ introduces a β term in the optimization formula. These β term is related to the introduction of per caput dose distribution but might be used for other dimensions. For the integration of time it is suggested to introduce a discounting function F(t). The aim of this paper is to analyse the advantages and drawbacks of these two possibilities.

THE INTEGRATION OF INDIVIDUAL DOSE EQUIVALENT INTO THE OPTIMIZATION PROCESS

If we analyze the differents levels of exposure of public or workers involved in the nuclear fuel cycle the results are contrasted. This is particularly right for occupational exposure where uranium miners are the most exposed group. For risk or equity consideration we may assume that it is desirable to focus on the protection of such exposed workers. It is therefore interesting to have an optimization procedure which allows to take into account the individual dose equivalent.

The NRPB has presented a procedure with alpha values depending on the per caput dose /5/. This method is quite comparable with the β formulation of the ICRP 37 (where the function f(Hj) can express the increase of the alpha values in function of the per caput dose) and we will use in our example this well known method.

To illustrate the interest of the integration of individual dose equivalent into the optimization procedure we will envisage the comparison of different protection strategies available to reduce the alpha contamination associated with short lived radon daughters in an underground uranium mine. The protection strategy i, its annualised cost Xi and the annual individual dose equivalent (Radon + Gamma + Ore dust) to the three groups of miners are given in the following table (data derived from another communication in this congress /6/).

(+) This paper is a summary of two studies carried out within the framework of two contracts with the "Euratom/CEA Association", Contracts n° SC 005 BIAF-423 F and SC 010 BIAF 423 F /1/ and /2/.

i	PROTECTION STRATEGY T	PV	SV	COST X_i (10^3 \$)	INDIVIDUAL DOSES (mSv/yr) $D1_i$	$D2_i$	$D3_i$
1	N	20	3	19.36	40.8	34.5	28.9
2	N	30	3	31.9	28.4	22.3	17.1
3	Y	30	3	34.24	26	21	16.3
4	Y	60	3	59.53	17.3	12.7	8.4
5	Y	60	5	65.61	15.8	11.3	7.7

T = implementation (Yes or Not) of small fans (Turbulator) in the stopes ; PV, SV = primary and secondary ventilation rates (in m^3/s).

A first cost-benefit analysis with an unique value of alpha as it was suggested in the ICRP 26 gives the following results (Number of workers 17 : 4 for the two first groups and 9 for the third) for an alpha value of 20, 000 \$ per man-Sievert (highest value taken in the ICRP 37 examples).

The "optimal" protection strategy, which minimizes the sum of financial X_i and detrimental Y_i cost, $Y_i = \alpha S_i$ with $S_i = 4(D1_i + D2_i) + 9\ D3_i$), is the first one. The maximum individual dose is about 80 % of the annual dose limit when this minimal protection strategy is implemented.

A second analysis taking into account the individual dose equivalent distribution gives the following results. In this example we assume that ALL RISKS MUST BE TAKEN INTO ACCOUNT (Radon, Gamma, Ore dust) though the strategies are assumed to reduce only radon exposure. Following, the NRPB, the alpha values retained as an example here are 6 000 \$/man.Sv for annual individual dose equivalents less than 5 mSv ; 30, 000 \$/man.Sv for annual individual dose equivalents between 5 and 15 mSv and 150,000 \$/man-Sv for annual individual dose equivalents greater than 15 mSv.

With these hypotheses the "optimal strategy" his now the forth-one. The maximum individual dose equivalent is then reduced to about 35 % of the annual dose limit.

The signifiant discrepancy between these two results is due to the rather large increase of the man-Sievert value in the calculation of the second analysis (individual dose equivalents are larger than 15 mSv for about all the options, hence the relevant value of alpha is 7.5 times higher than in the first analysis).

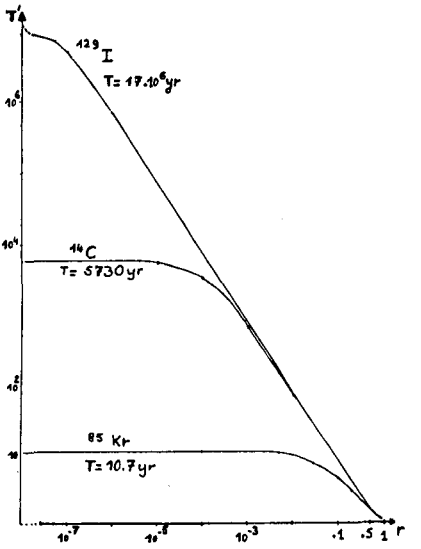
However it should be noted that the conclusions of the second analysis are probably more consistent with radiological protection of uranium miners.

THE INTEGRATION OF TIME INTO THE OPTIMIZATION PROCESS

The infinite time-integral of the collective dose, which is a measure of the total health detriment does not appear necessarily as being a realistic indicator within a decision aiding framework. Such a calculation has not been yet implemented for other risk than ionizing radiation, besides the prediction of events supposed to happen in millions of year can be questionable. A simple way of solving such a problem is often suggested by various authors. It consists of introducing a discounting exponential factor (e^{-rt} where t is time and r the discounting rate) which originates from the economic theory. However this manner of expressing the preference for present times vs the future is probably much more relevant when dealing with short or medium term decisions but not with very long term ones.

If one wishes to illustrate the impact of the discounting procedure (discounting rate r), it can be demonstrated that it is equivalent to the substitution of a radionuclide half-life T by a "pseudo" half life T' = T log2/(log2 + rT) inferior to T(*).

The following graph shows the variation of T' according to the value of the discounting rate r, for 3 radionuclides.



As can be seen, an annual discounting rate of 1 °/₀₀ shifts the half life of ^{129}I from 17.10^6 yr to about 690 yr (less than ^{14}C half life).

The use of a discounting procedure underweights significantly the long term consequences of present decisions. For example using an annual rate r = 1 % is equivalent to a reduction of the integration time interval P from infinite to 100 yr (in a first approximation P = 1/r see /2/).

The continuous attenuation of the health detriment cost with time tends to discard the future too heavily. Subsequently, an alternate way of tackling the problem of time distribution is to use various alpha values according to time or to adopt a step function in the ICRP 37 formula (which is equivalent to the former procedure). Various time intervals may be distinguished. Taking as an example the wastes disposal the first one can be identified to the life time of the storage center ($t \leq 25$ yr) the second one can be a period during which protection choices are kept reversible ($25 < t \leq 100$yr) the last one ($t > 100$ yr) may be cut so as to neglect time periods within which quantitative assessment present such a degree of uncertainty that it would be meaningless to take them into account (1 000 yr, 5 000 yr, 10 000 yr).

(*)
$$\int_0^\infty \frac{1}{(1+r)^t} \exp\left(-\frac{\log 2}{T}\right) t\, dt \simeq \int_0^\infty \exp -\left(\frac{\log 2 + rT}{T}\right) t\, dt$$

CONCLUSION

The use of various set of alpha values in order to take into account time and per caput dose distribution, (or similarly the introduction of β and F(t) in the equation of health detriment) is an appropriate way of expressing the multidimensionality of radiological protection choices.

The extension to other dimensions such as the need to distinguish between national and international exposure, the public and the workers etc.. might be envisaged in a further step of the elaboration of a practical decision aiding model for radiological protection.

However we should seek for a compromise between the adequation of the model to reality and the relative simplicity of this model which ought to be practicable and clearly understood by the various actors of the radiological protection system.

REFERENCES

/1/ J. LOMBARD, "La prise en compte de la distribution des individuelles dans l'optimisation de la protection radiologique", Rapport CEPN n° 65, Mars 1983.

/2/ J. LOMBARD, "La prise en compte du long terme dans l'optimisation de la protection radiologique", Rapport CEPN n° 73, Decembre 1983.

/3/ INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION - ICPR Publication 26, Pergamon Press 1977.

/4/ INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, ICRP Publication 37, Pergamon Press 1983.

/5/ NATIONAL RADIOLOGICAL PROTECTION BOARD, "Cost benefit analysis in the optimisation of protection of radiation workers : a consultative document", November 1982.

/6/ A. OUDIZ, J. LOMBARD, P. ZETTWOOG, "Radiological and economic assessment of various occupational protection options in uranium mines", 6th International Congress of the IRPA, Berlin - 7-12 May 1984.

ALARA- AND DE MINIMIS CONCEPTS IN RADIATION PROTECTION AND THEIR APPLICATION FOR THE NATURAL RADIATION ENVIRONMENT

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ABSTRACT

Normal levels of the natural radiation environment (NRE) are excluded from the dose equivalent limitation system, although they cause the largest contribution to the radiation burden of man. The significance from the different radon sources of the NRE and technical modifications of the NRE exposure are discussed with respect to their overall significance to the problem of applying the ICRP-ALARA-principle to man's exposure to the NRE. As an alternative graded de minimis-levels within a unified concept of radiation protection are proposed as a regulatory cut-off policy for "negligible" levels of radiation dose.

NRE AND TENR LEVELS

The natural radiation environment (NRE) and sources of technologically enhanced natural radiation (TENR) represent the largest sources of exposure to ionizing radiation for the general public. Although even "normal" NRE exposure exceeds levels from artificial sources by far, this type of radiation exposure is not intended to be subject of any system of dose limitation (IC77). Practical radiation protection problems arise from the overlapping definition of "normal" and "technologically enhanced" NRE sources:

a) Recycled industrial wastes, e.g. phosphogypsum from fertilizer industry or fly-ash from coal power plants, are increasingly used as construction material. The high ^{226}Ra concentration of some of these materials, up to 1220 Bq/kg (US75), also results in an increased equilibrium equivalent concentration (EEC) of radon inside such dwellings, typically 260 Bq/m³ (UN82).

b) Natural building materials, such as concrete based on alum shale, contain high ^{226}Ra-levels and represent a significant source for increased indoor radon daughter levels up to 770 Bq/m³ (Sw83).

c) Efforts to reduce the energy consumption for heating purposes consist mainly in a general reduction of the ventilation rate. This can increase the indoor radon concentration in such energy-efficient buildings up to 1000 Bq/m³ (Ne82).

Considerations about the mechanism of the effect of energy deposition at the cellular level prove the concept of a dose- or LET-threshold in radiobiology as logically invalid (Ka82). This is in agreement with the basic no-threshold-concept in the system of dose limitations adopted by the International Commission on Radiological Protection (ICRP). The logical consequence of this (admittedly cautious) approach is a unified concept of radiation protection, where every source of ionizing radiation is subject to control regardless of its origin, since the biological target "cell" cannot discriminate between energy deposition due to NRE, TENR or artificial radiation exposure. It appears particularly illogical to exclude voluntarily the most significant source of exposure of the general public, i.e. NRE, and to concentrate most efforts on artificial sources which deliver less than 2% of the total dose to the public (St83).

ALARA AND THE NRE

The objective of radiation protection should be primarily the protection of the individual against detrimental effects resulting from radiation regardless of source type. In the ICRP-system of dose limitation for artificial sources it is proposed that the limitation of stochastic effects is achieved by keeping all justifiable exposures as low as reasonably achievable (ALARA-principle). Two caveats are included, i.e. economic/societal factors have to be taken into account and the dose-equivalent limits shall not be exceeded (IC77). Risks from NRE can only be subject of regulation, if the sources can be controlled. However, it has to be considered that NRE-source control is not only dependent on the development of technical measures to control NRE exposure, but also on the public acceptance of risk and the willingness to invest in NRE-risk control. A decisive factor are the financial resources available for reduction of all environmental risks, of which "radiation" represents just one aspect.

In order to demonstrate the application of the ALARA-principle a "worst-case" has been assumed: average radon (Rn) EEC value indoors of 230 Bq/m³, i.e. 15 times above normal levels. Such levels can be found in dwellings built predominantely of materials with elevated radium content. The annual effective dose equivalent amounts to about 15 mSv for inhabitants of such houses. Technical remedial actions range from the application of wall-sealants to exchange of the subsoil with costs up to US $ 10 000 per dwelling. In Fig. 1 the standardized societal and protection costs are compared for this model dwelling. Assuming a total removal of the radon pollution down to "normal" levels of about 1 mSv by using soil replacement techniques, protection costs (X) increase non-linearly. Societal costs (Y) increase linear, assuming linear dose-response relationships and a no-threshold lung cancer risk.

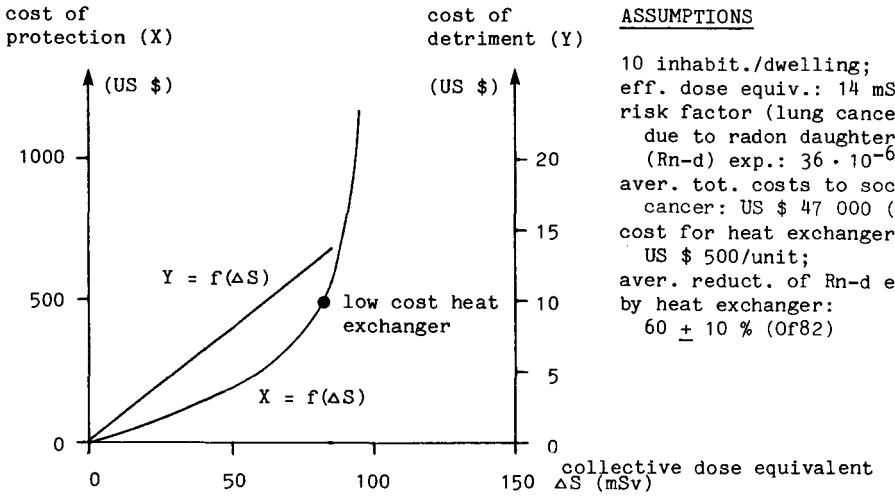

Fig. 1 Comparison of costs for technical remedial action against elevated radon daughter exposure and costs to society due to induced lung cancer cases, standardized for model dwelling

(figures rounded to indicate degree of uncertainty)

ALARA-optimization condition is fulfilled at a value S* under the following conditions:

$$\underbrace{(\Delta X/\Delta S)_{S^*}}_{A} = \underbrace{-(\Delta Y/\Delta S)_{S^*}}_{B}$$

It can be seen that even for a low-priced remedial action (heat exchanger) the above equation is not fulfilled (A >> B), i.e. application of the ALARA principle cannot be justified even for low-cost remedial actions at any NRE-exposure level.

DE MINIMIS AND NRE

From radiological studies it is known that deposition of energy from ionizing radiation at the cellular level results in the sterilization of a small proportion of the total cells, but also in the induction of various processes of repair and repopulation. The latter increases the threshold level of dose when irradiation is given over a long period (UN82). Therefore also NRE-exposure, which has been part of man's evolution and occurs through the total life span of any individual, must induce a very effective repair for a given level of detriment otherwise the human race would have been extinct by now. It has even been argued that a certain level of radiation exposure may not only be harmless, but is essential for the well-being of man (Lu80). As there is no zero-NRE-exposure for anybody and only a finite amount of financial resources available for general risk control, it has been suggested to use a certain cut-off level for artificial radiation sources, ranging from 10 to 100 μSv. For a comprehensive review of the de-minimis concept see ref. (Da81). This concept has the advantage that available financial resources and health physics manpower are used at an optimum. For a given dose frequency distribution in a population (about 2 - 5% of the people receive a dose an order of magnitude higher than the mean dose (St83a, Ho83)), it is the high dose group which should receive the largest efforts for reduction of their dose. The disadvantage of this approach is the possibility that summation of several "insignificant exposures" can eventually cause an exposure in excess of recommended limits. However, the probability for this to occur is low, if the cut-off level is set low enough to allow for a large number of such exposures to be required before a significant health hazard is associated with it.

In case of its application to NRE exposure it is essential to establish statistically significant NRE-dose-histograms for different population groups in various geographical areas. From this information the range of natural background doses can be determined and fractions thereof be used as de minimis-levels.

Defining a system of graded hierarchy for different de minimis levels, e.g. a "level of triviality" and "level of no regulatory concern" (0.1%, resp. 1% of the value of the detriment due to the collective dose equivalent of the NRE-exposure), provides a wide safety margin against any unwanted excess doses due to multiple de minimis exposures. Fig. 2 demonstrates the practical application of this scheme for the exposure conditions in a typical industrialized country. The total value of the detriment due to NRE exposure exceeds that for occupational exposure by a factor of over 200. Application of graded de minimis-levels within the framework of a unified radiation protection concept would enable the deregulation of many areas of occupational exposure presently requiring radiation protection practices, without imposing any significant health hazard to the worker. It is likely that implementation of such a scheme would improve the collective exposure conditions of the total population, since additional financial resources and manpower would be available to reduce the dose for those groups of the population, who at present receive high exposures in the general public as well as during occupational exposure.

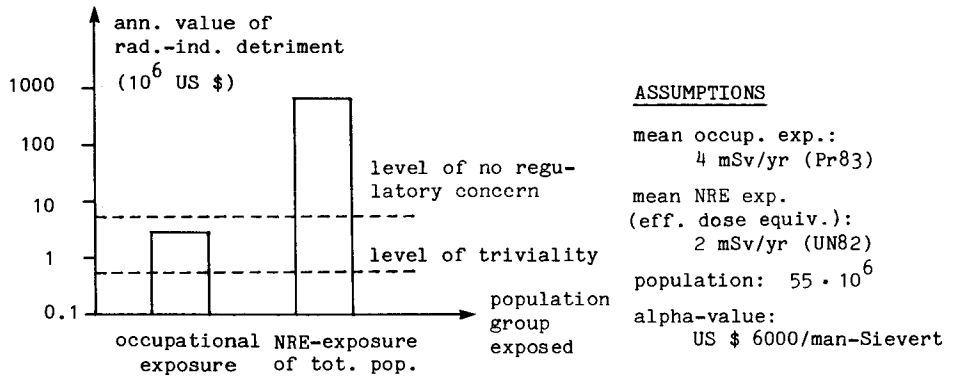

Fig. 2 Societal costs of detriment due to occupational and non-occupational radiation exposure in comparison to graded de minimis-levels

REFERENCES

Da81 Davis, J.P., 1981. General Physics Corp. Rep. No. GP-R-33040, Columbia, MA.

Ho83 Hofmann, W., Koblinger, L., Daschil, F. et al., 1983. Proc. XI. Regional Congr. of IRPA, Vienna, Austria (in press).

IC77 International Commission on Radiological Protection, 1977. Annals of the ICRP 1, No. 3.

Ka82 Katz, R. and Hofmann, W., 1982. Phys. Med. Biol. 27/9, pp. 1187-1192.

Lu80 Luckey, T.D., 1980. Hormesis with Ionizing Radiation. CRC-Press, USA.

Ne82 Nero, A.V., Berk, J.V., Boegel, M.L. et al., 1982. Proc. Int. Radon Spec. Meeting on "The Assessment of Radon Daughter Exposure and Related Biological Effects", Rome. RD Press, Univ. Utah, USA, pp. 144-152.

Of82 Offermann, F.J. et al., 1982. Lawrence Berkeley Lab., Univ. of California, Energy & Environment Div., Rep. No. LBL-13100 UC-95d.

Pr83 Price, L., 1983. Atom 317, pp. 54.

St83 Steinhäusler, F., Uzunov, I. and Pohl, E., 1983. Health Physics (in press).

St83a Steinhäusler, F., Hofmann, W., Pohl, E. and Pohl-Rüling, J., 1983. Health Physics 45, pp. 331-337.

Sw83 Swedjemark, G.A. and Mjönes, L., 1983. Book of Abstracts, Int. Seminar on Indoor Exposure to Natural Radiation and Related Risk Assessment, Anacapri, Naples, Italy, Abstract no. 71.

UN82 United Nations Scientific Committee on the Effects of Atomic Radiation, 1982. Report to the General Assembly, United Nations Publ. No. E.82.IX.8, New York.

US75 US-Environmental Protection Agency, 1975. ORP/CSD-75-3.

Vo82 Voillequé, P.G. and Pavlick, R.A., 1982. Health Physics 43, pp. 405.

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RADIOLOGICAL AND ECONOMIC ASSESSMENT OF VARIOUS OCCUPATIONAL PROTECTION OPTIONS IN URANIUM MINES

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INTRODUCTION

Object of this study is to present an example of a radiological and economic assessment of various protection options against alpha energy due to short lived Radon daughters in an underground non sedimentary mine /1/. The aim of the options is to reduce both the collective and individual effective dose equivalent received by the miners. In order to calculate these dose equivalents, a model has been prepared, which enables the simulation of the various protection strategies (+).

PRESENTATION OF THE MODEL

The mine model

One can imagine a mine as a more or less complex combination of a set of simple elements designated here by the term "branch". Each branch consists of a main gallery, an old stope and ten active ones.

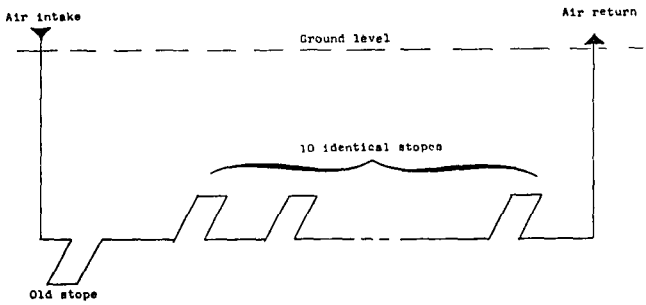

Fig. 1 : The considered mine model

The square section of the gallery and its stopes has an area of 16 m^2. All stopes are identical rectangular parallelepipedes with length, height and width of 12 m, 4 m and 4 m respectively.

(+) This paper is a summary of a study sponsored by the EURATOM/CEA-DPr Association, Contract n° SC 003 BIAF 423 F.

We assume a descending section mine. A figure of 0,1 °/₀₀ for uranium tenor in quasi-sterile zones is assumed and one of 2 °/₀₀ for tenor in the ore. Furthermore, the age of ^{222}Rn emanating from the old stope is assumed to be 25 mn.

The personnel employed in the branch consists of seventeen miners, distributed among the various zones of the mine according to their activity.

The alpha contamination model

The branch is considered as a sequence of ten "gallery-stope" pairs. The alpha energy in a gallery is calculated starting from the concentration of short lived ^{222}Rn daughters.

These concentrations are obtained by solving a system of differential equations relating the concentrations of ^{222}Rn and its daughters as functions of time. The concentration equations thus obtained involve the whole set of parameters related to the characteristics of the mine (volume, area, ^{222}Rn emanation flow, old stope), and to the protection options (primary ventilation rate, eventual presence of a parpen wall sealing the old stope. The modelling of alpha energy in a stope uses a compartment model, in order to properly account for the phenomenon of air stagnation caused by the working face that acts as a "cul de sac".

Here again, the equations that give the concentrations of short-lived ^{222}Rn daughters in both sections, involve the parameters related to the influence of the protection options (secondary ventilation rate, presence of filters, etc...).

Once the short-lived ^{222}Rn daughter concentrations are given, then it is possible to calculate the potential alpha energy inhaled by the various groups of miners and, subsequently, the associated collective and individual effective dose equivalent, using the factor proposed by the International Commission of Radiological Protection /2/for optimization purposes :

HE/IP = 2,5 Sv per Joule
HE : Effective dose equivalent per unit of potential alpha energy intake (IP)

DESCRIPTION OF THE PROTECTION STRATEGIES

A protection strategy is a combination of elementary protection options. The elementary options are the following :

a) Parpen wall (W) : putting up a wall isolating old stopes ;
b) Primary ventilation rate (PV) : choosing among four primary ventilation rates : 20, 30, 60 or 120 m^3/s ;
c) Secondary ventilation rate (SV) : choosing among three secondary ventilation rates : 3, 5 or 11 m^3/s ;
d) Turbulator (T) : introducing a small power rating fan (≃2 KW) into the working section in order to better ventilate the working face ;
e) Filters (F) : placing electrostatic filters (1, 2, 3 or 4) that would filter the primary air entering the stope and hold back the short-lived ^{222}Rn daughters.

The interdependence of the elementary options as far as effectiveness is concerned makes necessary the analysis in terms of strategies, that is combinations of elementary options. The effectiveness of a given elementary option depends, in fact, on the options already implemented.

THE ASSESSMENT OF THE PROTECTION STRATEGIES

In order to assess the various strategies, a cost-effectiveness analysis is carried out that compares the total cost and the reduction of the effective collective dose equivalent corresponding to each of the 240 available strategies for a given ten year period. Moreover, the maximum individual effective dose equivalent is calculated for each strategy in order to compare it to the occupational dose limit of 0,05 Sv/yr. Actually there is no need to carry out the analysis for the 240 strategies. It is only necessary to consider the lower part of the convex hull of the set of points corresponding to the 240 strategies.

Results and comments

The following table provides a summary of the results obtained.

	PROTECTION STRATEGY					Ci	Di	EIDEi	$\alpha i = \frac{Ci-Ci-1}{Di-1-Di}$
i	W	T	F	PV	SV	10^3 $	man-Sv	10^{-3} Sv/yr	(10^3 $/ Man.Sv)
R	N	N	0	20	3	151.2	7.04	55.9	-
1	Y	N	0	20	3	193.6	4.54	40.8	17
2	Y	N	0	30	3	319	2.50	28.4	61.5
3	Y	N	0	30	3	342.4	2.28	26	106.4
4	Y	Y	0	60	3	595.3	0.89	17.5	181.9
5	Y	Y	0	60	5	656.1	0.70	15.8	320
6	Y	Y	0	120	5	1 129.7	0.29	13.3	1 155
7	Y	Y	0	120	11	1 378.7	0.19	12.3	2 490
8	Y	Y	4	120	11	2 067.5	0.14	12	13 776

W,T : implementation (Y) or not (N) of wall or turbulator ; F : number of filters per stope ; PV, SV : primary and secondary ventilation rates in m^3/s ; Ci : total cost over a 10 year period expressed in 1981 US $; Di : radon daughters effective collective dose equivalent for 10 years ; EIDEi : annual effective individual dose equivalent due to radon, Gamma and ore dust exposure.

Table 1 : Summary of the results

It should be noted that the annual effective individual dose equivalent has been calculated for the most exposed category of miners, i.e. the drillers. Moreover, this calculation takes into account the contribution of gamma external irradiation ($4.5\ 10^{-3}$ Sv/yr) as well as the ore dust (a figure of $6.5\ 10^{-3}$ Sv/yr has been assumed, according to /3/).

The results obtained show that the reference strategy (R) cannot be considered as an acceptable one from the radiological protection standpoint because it does not comply with the third principle of the ICRP dose limitation system /4/. The cost-effectiveness analysis does not lead directly to the choice of a strategy among the eight available ones. For this it would be necessary to have a man-Sievert reference value α. If this value was fixed, then the "optimal" strategy (in the sense of ICRP /4/), would be the one having the highest man-Sievert cost figure α_i but inferior to this reference value α.

CONCLUSION

Object of this study is to make a contribution to the assessment of radiological protection strategies. The cost-effectiveness analysis allows in fact to compare the new techniques such as the turbulator or the electrostatic filters, to the more traditional ones (changing the ventilation rate).

REFERENCES

/1/ LOMBARD J., OUDIZ A., ZETTWOOG P., "Contribution à l'optimisation de la Protection radiologique du personnel dans une mine d'uranium", Rapport CEA (à paraître), Fontenay-Aux-Roses, France.

/2/ ICRP Publication 32, "Limits for Inhalation of radon daughters by workers", Pergamon Press 1981.

/3/ BERNHARD S., "L'irradiation du personnel dans les mines souterraines de la COGEMA 1979", Session d'étude de Vassivière, 22 au 24 Septembre 1980, France.

/4/ ICRP Publication 26, Pergamon Press 1977.

BALANCING NUCLEAR SAFETY AND WORKER PROTECTION REQUIREMENTS: A NEW CHALLENGE IN THE APPLICATION OF RADIOLOGICAL PROTECTION PRINCIPLES

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Introduction

In recent years, accelerated efforts in nuclear safety to minimise the potential frequency and consequences of accidents that could impact on the public have resulted in additional requirements affecting the design and operation of nuclear installations. These included in-service inspection, increased preventive maintenance on sensitive equipment, revised safety specifications necessitating equipment modification and backfit, and unscheduled special maintenance programmes. These requirements, formulated by designers and operators or imposed by regulatory authorities, involve exposing personnel to radiation doses which are additional to those originally expected to be associated with the normal running of the plants.

The trade-off to this is represented by a decrease of the potential exposure of the public, and possibly the workers, due to the possible decrease of the probabilities of accidents and/or the projected doses associated with these accidents. It is to be noted, however, that this decrease applies to an exposure of the public which is virtual rather than real.

There is a growing feeling among experts that this situation seems to show an imbalance, not only in the actual level of risk attributed to two different groups of people (workers and population), but also in the response to two different requirements (safety of plants and protection of workers) which both have a regulatory value. The result could be, in some cases, a deterioration, rather than an improvement, of the overall level of radiological protection associated with nuclear facilities.

One of the contributing factors to this imbalance may be the fact that many safety requirements appear to be established in isolation from the global context mentioned above, and do not seem to be based on an explicit analysis of the justification for the increase of radiation detriment to workers generated by the safety-related requirements in connection with their safety-related benefits, which may be expressed in terms of lower risk to the public and, under some circonstances, to the workers.

There is, therefore, a need for a rational approach which allows the finding of the optimum balance between the requirements of nuclear safety and those of radiological protection. In order to achieve such a result, there is a need to establish a dialogue between radiological protection experts and nuclear safety engineers for the purpose of developing a common language and methodological approach to the optimisation of safety.

For this purpose, the Committee on Radiation Protection and Public Health of the NEA set up, in 1982, a Group of Experts entrusted with the task of reviewing available data on occupational exposure in the nuclear industry, identifying those aspects which involve problems of balancing nuclear safety and radiological protection requirements, and discussing possible approaches and methodologies for the determination of an optimum balance. The work of the Expert Group is far from complete, and its concepts are still in the course of development. However, some preliminary considerations can be made on this work.

Occupational Exposure Implications

Undoubtedly, there appears to be a lack of published data on worker doses associated specifically with nuclear safety-related operations and on expected doses resulting from plant backfitting. It is, therefore, difficult to analyse these safety-related actions with respect to their dosimetric implications. Nevertheless, the few data available, almost entirely referred to nuclear power plants, seem to suggest that there are cases in which an imbalance is produced between the implementation of nuclear safety requirements and the corresponding levels of protection of workers. There is, therefore, scope for improvements in this area.

Such an improvement, however, should be sought through the search for an optimisation of the overall level of protection of the whole community represented by the workers and the public.

This provides an incentive to establish a generic methodology to define whether the nuclear safety-related dose expenditures are optimised, in the ICRP ALARA sense, and whether the increase in public safety, or other benefits thus achieved, warrant the resulting increase in occupational risk. The answer has specific implications for the required extent and methods of in-service inspection and maintenance, and the degree to which additional design efforts may be needed in catering for this facet of nuclear safety.

Also, although the ICRP system of dose limitation, of which optimisation is a part, has been initially applied explicitly only to the case of sources under control (normal conditions of operation), there are increasing indications that the protection of people should be optimised in all the aspects of the performance of a given practice, including those involving events of a stochastic nature, such as the accidents.

Conceptual Approach to an Optimised Balance

The Expert Group is studying a tentative conceptual approach to this optimisation, which is similar to the cost-benefit analysis approach suggested by the ICRP for the application of the principle of optimisation of protection.

The relevant parameters for the definition of the suggested balance approach are financial costs required for the implementation of safety measures, potential financial losses associated with nuclear accidents, collective doses to workers due to safety-

related activities, potential collective doses to workers and the public saved through the decrease of accident probability and/or mitigation of their consequence due to the implementation of safety measures.

However, the application of this balance involves two orders of problems. One is the question of the balance between workers and population detriment in the application of the optimisation process. The other, more complex problem is the fact that to evaluate detriments in the case of probabilistic events it is necessary to consider not only the magnitude and distribution of doses that may arise from these events, but also their probabilities of occurrence.

The question of balance between workers and population detriment in the optimisation procedure can be dealt with, for example, in the framework of the methodology suggested in ICRP Publication 37. In particular, with reference to the equation of the cost of detriment given in that publication (ICRP Pub. 37, paragraph 87), the same value of α could be applied to the collective doses to workers and the public, ignoring possible minor differences in the objective health detriment per unit collective dose between workers and the public. On the contrary, both the value of β and the form of the function $f_j(H_j)$ in the "β term" of the equation can be expected to be different for the two groups of people. This is due to the different degress of voluntariness in the acceptation of risk and the different levels and types of risk perception in the two groups.

As far as the other question of the treatment of probabilistic events in optimisation is concerned, the only possible solution seems to be the shift from the concepts of dose and detriment to the concept of risk, defined here as the probability of a health-effect for an individual. It is to be recognised that the risk associated with a given event is, in fact, a combination of the probability of this event and that of its potential consequences. This can be expressed as the product of the probability of people receiving a given dose and the probability of detrimental health effects associated with that dose.

The use of the concept of risk in an optimisation process taking into account the whole spectrum of accidents potentially associated with a nuclear plant, requires the assessment and the costing of the total collective risk associated with this spectrum of accidents. A combination of probabilities and consequences for this purpose, based purely on the product of these quantities, would give an "expectation value" of the objective collective risk. This may be a too simplistic view of the total collective detriment, when a large spectrum of probabilities and consequences is involved. The response of people and decision makers to accidents having different consequences is, in fact, different and not necessarily linearly related to their probabilities. It would be necessary, therefore, to introduce in the calculation of the total collective risk appropriate subjective factors giving a variable weight to different consequences in relation to their probabilities of occurrence. Only then would it be possible to assign a cost to the total collective risk and balance it against the costs of protection and prevention.

In the specific case which is the object of the present study, the various issues mentioned above are complicated by the fact that a part of the risk - that to the workers - is associated with a real dose and is therefore represented by a "risk of the first order" of stochasticity, namely the risk of health effects due to the dose actually received. On the other hand, the part of risk which concerns the public in connection with the accidents is only associated with a potential dose, that is it corresponds to a "risk of the second order" of stochasticity.

Conclusions

The degree of quantification of the parameters involved is far from being satisfactory at present. In order to overcome this difficulty, an organised data base correlating measured occupational doses with specific nuclear safety-related measures and activities and their monetary costs should be set up. Moreover, objective calculations of the occupational and public risk variations resulting from the past or present introduction of safety-related measures and procedures should be carried out.

An additional difficulty of a less technical nature, which can complicate or delay the already technically difficult solution of this issue, is the concern expressed by the nuclear operators, designers and safety specialists about the methodological complications associated with the introduction of a further constraint, represented by the requirement of optimisation, into their decision-making process.

The task assigned to the NEA Expert Group is obviously very difficult and the state of scientific and technical development, as well as that of doctrine and policy are probably not mature yet for the solution of this complex issue. However, the debate among nuclear safety and radiological protection experts is now actively engaged, and this represents in itself a successful achievement. The future will tell us which will be the outcome of these efforts.

DERIVED LIMITS FOR APPLICATION TO THE CONTROL OF ROUTINE RELEASES

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INTRODUCTION

The National Radiological Protection Board has undertaken a programme of work in which derived quantities for use in the control of routine discharges of radioactivity to the environment have been calculated for a range of commonly occurring radionuclides. The quantities, which are intended for general application within the UK, are referred to as Generalised Derived Limits (GDLs) and are convenient reference levels against which the results of environmental monitoring can be compared and atmospheric discharges assessed. Compliance with Derived Limits should ensure that the appropriate dose equivalent limits are not exceeded. Due to their general nature, GDLs are intended for application only when the environmental contamination or discharge to atmosphere is less than a small fraction, say 5%, of the GDL; above this level a more realistic site specific assessment may be necessary to take account of possible variations or uncertainties in parameter values and to identify more accurately the location and habits of the critical group.

GDLs may be calculated for any environmental materials liable to be subject to routine sampling, such as water, soil, grass, sediments and various foodstuffs derived from the terrestrial and aquatic environments, and they may also be specified for discharge to atmosphere. In this paper a selection of GDLs is presented for several commonly occurring nuclides, namely ^{90}Sr, ^{137}Cs, ^{131}I, ^{239}Pu and ^{241}Am. A full range of GDLs have been reported elsewhere for these and other nuclides of strontium, caesium, iodine, plutonium, americium and curium[1] and the calculation of GDLs for nuclides of other elements will be undertaken in the future.

PRINÇIPLES FOR THE CALCULATION OF DERIVED LIMITS

To ensure virtual certainty of compliance with the dose equivalent limits for members of the public, assumptions of a deliberately pessimistic nature are incorporated in the GDL calculations. For example, the exposures considered are those of a hypothetical population group, the critical group, whose members are subject to above average exposures by virtue of their location, age or habits, such as elevated consumption rates. The intake rates used in GDL calculations represent higher than average consumption rates but not extreme values which would be applicable only to a few individuals in the population. They thus provide an appropriately conservative basis for the estimation of GDLs in foodstuffs.

For a particular radionuclide, GDLs are determined by considering the concentration in environmental materials which lead to the critical group receiving the annual dose equivalent limit for members of the public (either 5 mSv effective dose equivalent or 50 mSv to a single organ or tissue if this is the more restrictive). When the exposure route is by ingestion or inhalation of activity, annual limits of intake (ALIs) for members of the public are used in GDL calculations. ALIs are related to the annual dose equivalent limits by means of defined metabolic and dosimetric models for man. ALIs may be determined by either stochastic or non-stochastic effects; the more restrictive gives the ALI which is used in GDL calculations. The principles and method of calculation of ALIs are those given in ICRP Publication 30[2]. For adult members of the public, the ALI's are taken to be one tenth of those given for workers in ICRP 30, except in certain

cases where it is considered that, for the form of elements likely to be found in the environment, alternative metabolic parameter values are more appropriate[3]. For children, ALIs are evaluated using a model which predicts the variation of dose per unit intake with age[3]. In addition an extended integration time following intake, up to a mean lifetime of 70 years is used. The values of the ALIs used in GDL calculations for adults and children for the nuclides considered are given in Reference 1.

GDL calculations are performed for three age groups - adults, children aged 10 years and infants - taking into account the variation with age of the ALI and dietary and other habits. The most restrictive value of those obtained for the three age groups is taken as the GDL.

GENERALISED DERIVED LIMITS IN ENVIRONMENTAL MATERIALS

The GDL in an environmental material m, for a particular pathway p is given for activity taken into the body by

$$GDL_{m,p} = \frac{A}{I_m} \quad (Bq\ kg^{-1} \text{ or } Bq\ l^{-1}) \qquad \ldots\ldots\ldots (1)$$

where A is the appropriate ALI ($Bq\ y^{-1}$)
and I_m is the annual intake of material, m ($kg\ y^{-1}$ or $l\ y^{-1}$).

When the exposure route is external irradiation the GDL is given by

$$GDL_{m,p} = \frac{H_L}{H_m^E} \quad (Bq\ kg^{-1}) \qquad \ldots\ldots\ldots (2)$$

where H_L is the annual dose equivalent limit ($Sv\ y^{-1}$)
and H_m^E is the annual dose equivalent from external exposure from unit concentration of the nuclide in material m ($Sv\ y^{-1}$ per $Bq\ kg^{-1}$)

For each environmental material the GDL_m is then calculated by summing the contributions from all significant pathways of exposure, p, as follows

$$\frac{1}{GDL_m} = \sum_p \frac{1}{GDL_{m,p}} \qquad \ldots\ldots\ldots (3)$$

For example, to calculate a GDL for activity in soil account must be taken of all relevant pathways to man which may include inhalation of activity resuspended from the ground, external irradiation from the ground, and the ingestion of contaminated crops grown in the soil and of products from grazing animals.

GENERALISED DERIVED LIMITS FOR DISCHARGE TO ATMOSPHERE

The discharge of radionuclides to atmosphere may lead to members of the public being exposed by a number of pathways, including

external exposure from the cloud
internal exposure from inhalation of the cloud
external exposure from deposited activity
internal exposure from inhalation of activity resuspended from the ground

and internal exposure from ingestion of contaminated foodstuffs.

The GDL for the discharge of a radionuclide to the atmosphere, GDL_A, is given by

$$GDL_A = \frac{H_L}{H^E} + \frac{H_L}{H^I} \quad (Bq\ y^{-1}) \qquad \ldots\ldots\ldots (4)$$

where H_L is the annual dose equivalent limit (Sv y^{-1})

H^E is the annual dose equivalent from external radiation at the appropriate distance from the discharge point per unit annual discharge of a nuclide (Sv y^{-1} per Bq y^{-1})

and H^I is the time integrated dose equivalent due to the annual intake of a nuclide by inhalation and ingestion at the appropriate distance from the discharge point per unit annual discharge (Sv y^{-1} per Bq y^{-1})

For the purpose of calculating GDL_A, reference values are used for the parameters which influence dispersion in, and deposition from the atmosphere[4]. The critical group is assumed to be located at a distance of 100 m from the point of discharge but to derive its food uniformly from within 1 km of the discharge point. The average concentration in food is approximated by taking the concentration in food at a distance of 500 m from the discharge point. The GDLs for discharge are evaluated assuming that the discharge continues for a period of 50 years, to allow build up of activity to occur in soils and other environmental materials.

When the discharge is greater than about 5% of the GDL, a more realistic site specific estimate of the derived limit for discharge to atmosphere may be warranted, taking into account the conditions of the release and the location and habits of the exposed population. A procedure for this is described in Reference 5.

RESULTS OF GENERALISED DERIVED LIMIT CALCULATIONS

A selection of GDLs for five commonly occurring nuclides are given in Table 1. It can be seen from Table 1 that for these nuclides the critical age group is usually infants or adults. The adult age category is, with the exception of the GDL for milk, the most restrictive age group throughout for ^{239}Pu and ^{241}Am, whereas for ^{90}Sr, ^{137}Cs and ^{131}I the most restrictive age group is infants. The relative significance of the pathways which contribute to the GDL for stack discharge depends upon the nuclide. For ^{239}Pu and ^{241}Am the most important route of exposure during conditions of continuous discharge to atmosphere is inhalation in the initial cloud, while for ^{90}Sr and ^{131}I the major contributor is the dose from the consumption of contaminated food, mostly milk and milk products. For ^{137}Cs, the doses from the consumption of foodstuffs and external irradiation from deposited activity are of equal importance.

A full range of GDLs for these and other nuclides are reported in Reference 1, including GDLs in aquatic materials such as water (fresh water, sea water and drinking water) and marine foodstuffs.

REFERENCES

1. Generalised derived limits for radioisotopes of plutonium (NRPB-DL5), americium and curium (NRPB-DL6), caesium (NRPB-DL7), strontium (NRPB-DL8) and iodine (NRPB-DL9). Chilton, NRPB-DL series, 1982-1983 (London, HMSO).

2. ICRP, Limits for intake of radionuclides by workers. Oxford, Pergamon Press, ICRP Publication 30, Part 1. Ann. ICRP, 2, 3/4 (1979), Part 2, Ann. ICRP, 4, 3/4 (1980), Part 3, Ann. ICRP 6, 2/3 (1981).

3. NRPB, Metabolic and dosimetric models for application to members of the public. Chilton, NRPB-GS3 (to be published).

4. Harrison, N T and Simmonds, J R, Dosimetric quantities and basic data for the evaluation of generalised derived limits. Chilton, NRPB-DL3 (1980) (London, HMSO).

5. Hallam, J, Harrison, N T and Jones, J A, A procedure for estimating site specific derived limits for the discharge of radioactive material to the atmosphere. Chilton, NRPB-DL4 (1983) (London, HMSO).

TABLE 1

GENERALISED DERIVED LIMITS

GDL	Unit	Nuclide				
		^{90}Sr	^{137}Cs	^{131}I	^{239}Pu	^{241}Am
Milk[a]	Bq l^{-1}	$2\ 10^{2}$*	$3\ 10^{2}$*	$5\ 10^{1}$*	1‡	$8\ 10^{-1}$‡
Beef and veal[a]	Bq kg^{-1}	$2\ 10^{3}$	$5\ 10^{3}$**	$1\ 10^{3}$**	$8\ 10^{1}$	$8\ 10^{1}$
Green vegetables[a]	Bq kg^{-1}	$1\ 10^{3}$*	$2\ 10^{3}$*	$5\ 10^{2}$*	$6\ 10^{1}$	$6\ 10^{1}$
Grain products[a]	Bq kg^{-1}	$8\ 10^{2}$*	$1\ 10^{3}$*	$3\ 10^{2}$*	$4\ 10^{1}$	$4\ 10^{1}$
Well-mixed soil[b]	Bq kg^{-1}	$2\ 10^{3}$*	$3\ 10^{3}$*	– [c]	$3\ 10^{3}$	$3\ 10^{3}$
Surface contamination[d]						
1) soil	Bq m^{-2}	–	–	–	$2\ 10^{3}$	$2\ 10^{3}$
2) urban	Bq m^{-2}	–	–	–	$5\ 10^{1}$	$5\ 10^{1}$
Discharge to[e] atmosphere	Bq y^{-1}	$2\ 10^{12}$*	$1\ 10^{12}$*	$3\ 10^{11}$*	$1\ 10^{10}$[f]	$1\ 10^{10}$

Notes:
a All food products as fresh weight.
b Soil evenly contaminated down to 30 cm, no deposition.
c The half-life of ^{131}I is too short for this GDL to be significant.
d These GDLs are only relevant for those nuclides for which inhalation related pathways are important. They are calculated for surface deposits in the absence of continuous deposition and are based on the resuspension pathway only.
e Using reference discharge parameters - see text.
f This value may be relaxed by a factor of three if the plutonium is discharged as PuO_2.

The critical group is adults except where indicated by
* for infants aged 1 year
** for children aged 10 years
‡ for infants drinking milk in the first year of life (average age 6 months). See Reference 1 (NRPB-DL5).

RECORDING AND INVESTIGATION LEVELS: EFFECTS ON INDIVIDUAL AND COLLECTIVE DOSE EQUIVALENTS

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INTRODUCTION

The purpose of personnel monitoring of radiation workers is two-fold: (i) to demonstrate compliance with the prescribed dose equivalent (DE) limits, (ii) to maintain DE records which may be of use in the study of long term effects of low levels of exposures. The recommendations of the International Commission on Radiological Protection (ICRP) are used by national and international institutions to formulate their system of dose limitation. The latest publication of ICRP (1) has given a concept of a recording level (RL) as one above which Dose Equivalent and Intake are of sufficient interest to be worth recording and keeping. Results below the RL may be treated as zero for assessing DE or intake for radiation protection. In a subsequent publication (2) a value of 10% of DE corresponding to the period of monitoring has been suggested for the recording level. There has been apprehension in the minds of radiation protection people regarding the application of this concept due to resultant loss of information in individual and collective doses. The effect of 5%, 10% or 15% recording level on the individual and collective DE (CDE) using the data generated at this Research Centre are reported here. The role of fixing an investigation level on the control of personal exposures is also brought out.

DATA SOURCE

The data source is the monthly personnel monitoring records of about 2450 radiation workers at the Research Centre for 1978. Personnel monitoring was then carried out at the Research Centre, employing the film badge and all doses measurable by the system were recorded (3). This is being gradually replaced by the thermoluminescent dosimeter badge.

DATA ANALYSIS

Radiation workers at the Research Centre were grouped into five categories, viz. (i) fuel fabrication, (ii) reactor operation, (iii) fuel reprocessing, (iv) radioactive waste management and (v) research. Fuel fabrication includes processes starting from the uranium concentrate to the final fuel bundles. The monthly DE data of the workers of each category and the Research Centre as a whole was analysed for (i) DE distribution, (ii) effect of RL on the annual collective DE and (iii) effect of RL on the individual annual average DE. Part of the data was also analysed to determine, (iv) the effect of RL on individual annual DE, (v) the distribution of individual monthly DE values for same annual DE, (vi) the effect of investigation level. Recording levels of 5, 10 or 15% of the monthly dose limit were considered. Thus personnel monitoring dose records were scanned for doses of $\leqslant 0.2$, $\leqslant 0.4$ and $\leqslant 0.6$ mSv.

To evaluate the effect of recording level on individual DE, persons in the same category receiving nearly the same annual DE in the year 1978 and 1979 were selected. Thus there were 15 persons receiving annual DE of 3.5, 8.0, 12.0 and 20.0 mSv, in each case. The distribution of monthly DE values for each category was tested for log-normal distribution.

RESULTS

In Table I annual collective DE and its distribution for each of the five categories separately as well as total for the Research Centre as a whole is given in four DE intervals. Fuel fabrication and research contributions to the CDE are relatively small. The maximum contribution to the CDE is in the range of 5 to 30 mSv except for the fuel fabrication and research groups. In Table II the

Table I

DE Distribution

	Category	Collective DE (mSv)	% of the Centre	% Collective DE between (mSv)			
				0-5	5-15	15-30	>30
(A)	Fuel Fabrication	172.50	2.83	93.49	6.51	-	-
(B)	Reactor Operation	2381.40	39.0	20.56	46.97	31.20	1.26
(C)	Fuel Reprocessing	2405.0	39.4	14.24	25.85	53.25	6.66
(D)	Waste Management	766.25	12.55	14.74	28.05	29.87	27.44
(E)	Research	380.35	6.23	60.55	25.0	14.44	-
(F)	Research Centre	6105.50	100	21.88	33.83	37.75	6.54

effect of RL on the collective DE is given. Dose reduction in collective DE depends on its absolute value and on the DE distribution. The percentage reduction in collective DE for different values of RL is as follows: 5% - negligible to 4, 10% - 4 to 33, 15% - 9 to 57. When the evaluation is done for the Research Centre as a whole the effect is 1.4%, 8.7% and 16.5% respectively. In Table III the effect on the individual annual average DE is given. It is expected that the

Table II

Effect of Recording Level on Collective DE

Category	Collective DE (mSv)	5%		10%		15%	
		Reduction in CDE (mSv)	%	Reduction in CDE (mSv)	%	Reduction in CDE (mSv)	%
A	172.50	3.0	1.74	57.70	33.45	98.55	57.13
B	2381.40	57.30	2.41	265.60	11.15	478.0	28.07
C	2405.0	1.93	0.08	92.98	3.87	213.35	8.87
D	766.25	8.25	1.09	59.95	7.82	119.4	15.58
E	380.35	16.55	4.35	56.70	14.91	99.6	26.19
F	6105.5	87.03	1.43	532.93	8.73	1008.9	16.52

higher the DE the lower would be the effect of RL. But there is some dependence of actual reduction on the distribution of DE. Analysis of the effect of RL, particularly 10 and 15%, on the rangewise DE distribution of collective DE shows that the percentage reduction is very high at low DE and decreases with increasing DE.

Table III

Effect of RL on Average Individual DE

Category	No. of radiation workers	Av. DE (mSv)	Av. DE (mSv) after applying RL of		
			5%	10%	15%
A	391	0.44	0.43	0.29	0.19
B	566	4.21	4.11	3.74	3.36
C	497	4.85	4.83	4.65	4.41
D	152	4.99	4.93	4.59	4.20
E	848	0.45	0.43	0.38	0.33
F	2455	2.49	2.45	2.27	2.08

In Fig. 1 the effect of RL on individual annual DE is shown for two categories of workers. Same trend as seen for the influence of RL on individual annual average DE (Cf table 3) is observed. The effect of 10 and 15% RL decreases rather steeply with increasing DE.

In Fig. 2 the distribution of monthly individual doses for groups with nearly same annual DE is plotted. From the figure it is seen that the data fits generally a log-normal distribution. Such graphs can be used to estimate the probability of monthly dose being less than a certain RL and thus assess the reduction in DE. Such calculations showed general agreement with the data presented earlier in tables 2 and 3.

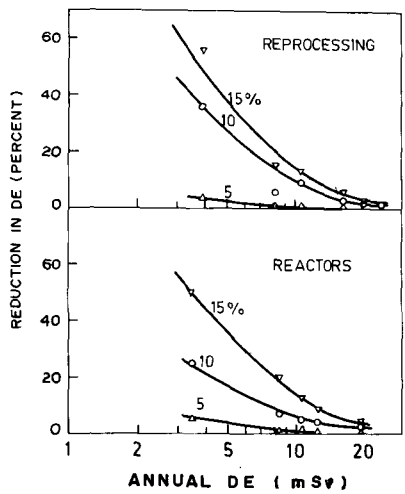

Fig.1 Effect of RL on individual annual DE

INVESTIGATION LEVEL

It has been the practice at the Centre to treat DE exceeding twice the limit corresponding to the period of monitoring as subject to investigation. In the year under analysis 104 cases were investigated. With this procedure there was no case of individual DE exceeding 50 mSv.

CONCLUSIONS

1. The effect of RL on the collective DE is dependent on its distribution. For example the CDE for reactor operations and spent fuel reprocessing is very nearly the same. But the influence of RL at all levels, 5, 10 or 15%, is different for these two categories. Similar considerations hold good for other categories too.

2. The annual average DE for the Research Centre i.e. for all the five categories together is not very sensitive to the RL level selected. However, in the

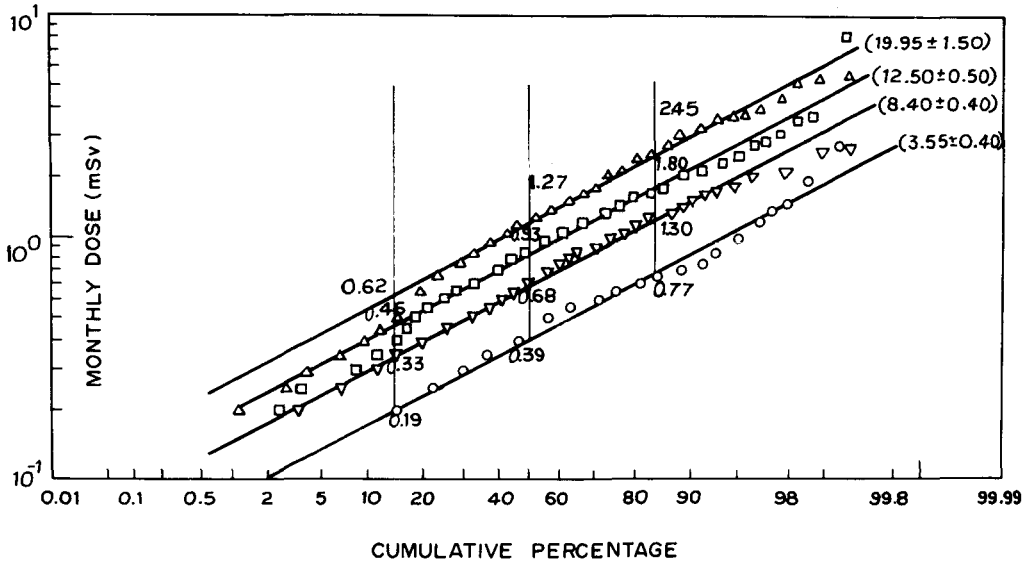

Fig.2 Distribution of monthly DE for different annual DE

5 categories separately, it is different depending again on the dose distribution.

3. The effect of 10% RL on individual and collective DE seems acceptable on radiation protection considerations.

References:

1. Radiation Protection, International Commission of Radiological Protection Publication 26 (1977). Pergamon Press.

2. General Principles of Monitoring of Radiation Protection Workers, ICRP Publication 35 (1982), Pergamon Press.

3. Nayyar S.S., Jain V.K., Singh Nagpal J.B. and Shenoy K.S., Personnel Monitoring in India, in Radiation Protection Monitoring I.A.E.A., (1968) Vienna, 393.

COST-BENEFIT ANALYSIS FOR IODINE FILTER INSTALLATION IN THE AUXILIARY BUILDING IN AN OPERATING PWR

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BACKGROUND

Following the TMI2 accident the installation of an iodine filter in the Ringhals unit 2 auxiliary building was considered by the Swedish State Power Board. Such a filter should as far as practically possible meet the requirements in Reg. Guide 1.52. Later on a filter was also demanded by the Swedish National Institute of Radiation Protection.

The preliminary design of a filter system was based on a reactor coolant activity level equal to the one obtained at TMI2. The leakage in the auxiliary building was assumed to be in accordance with the Technical Specifications for largest permissible leakage in isolation valves. Calculations indicated a cost of about 15 MSEK which raised the question whether an iodine filter would be cost-effective or not from a safety point of view.

At present the air from the auxiliary building is vented during normal operation and accidents through absolute filters out to the environment. These filters do not prevent the release of elemental iodine or organic iodide. Four alternatives were therefore considered:

1. Accept the present state of no iodine filter
2. Exchange existing absolut filters to iodine filter, which the ventilation air would by-pass during normal operation
3. Add a simplified iodine filter for accident conditions
4. Add an iodine filter largely according to R.G. 1.52 for accident conditions.

In alternative 2 the prevention of aerosol release (like caesium) is probably negligible. These four alternatives were investigated regarding cost and consequences of an accident of TMI-type. The methods and results are given in this paper.

ACTIVITY RELEASES OF A LARGE ACCIDENT

The conditions and assumptions of the TMI-type accident are as follows. The activity level in the reactor coolant is, according to TMI2 experience, for the important nuclides in this context:

Noble gases	100 % of core inventory
Iodine and caesium	50 % of core inventory

The largest permissable leakage flow according to Technical Specification is 45 cm^3/s. The time before isolation of the leakage is achieved is assumed to be 60 minutes. The activity leaked in this way to the auxiliary building is vented to the filters. However, due to steam condensation and deposition on walls, ventilation drums, etc. it is assumed that only 10 % of the iodine resp. 1 % of the

caesium will reach the filter(s).

With these assumptions the activity to the filter(s) will be:

Noble gases:	4700 TBq (13000 Ci)	Kr-87 equivalents
Iodine:	109 TBq (2900 Ci)	I-131 equivalents
Caesium:	10 TBq (270 Ci)	of which 1.5 TBq from the isotopes Cs-134 and Cs-137

The dose calculations based on above releases (without regard to filter attenuation) showed that no acute effects were obtained. The dominating individual dose was as expected the thyroid dose to children, which was 1.5 Sv at 0.5 km distance. The results of the collective dose calculations (ref 1) are shown in Table 1. The thyroid dose from iodine is transformed to a weighted whole-body dose by multiplying the thyroid dose by a factor of 0.03 according to ICRP 26.

Table 1

Collective weighted whole-body doses from iodine (no iodine filter) and caesium (no absolute filter) with an accident of TMI-type and a simultaneous reactor coolant water leakage. 0-150 km.

	5% cumulative frequency (weather conditions)		50% cumulative frequency (weather conditions)	
	iodine	caesium	iodine	caesium
External gamma radiation and inhalation dose, mmanSv	9.3	0.17	1.1	0.020
Ground dose (30 years), mmanSv	1.2	3.2	0.14	0.39
Dose from foodstuffs (30 years), mmanSv	-	1.7	-	1.1
Total, mmanSv	10.5	5.1	1.2	1.5

At 5% cumulative frequency, there is a small gain if the absolute filter is exchanged for an iodine filter. This gain is uncertain, since it depends on a comparison between two nuclides which behaviour in the auxiliary building is different and uncertain. At 50% cumulative frequency, the caesium gives greater concequences.

IS THE IODINE FILTER COST-EFFECTIVE FROM A SAFETY POINT OF VIEW?

To estimate the cost-effectiveness of "back-fitting" an iodine filter the method recommended in NUREG-0880 (ref 2) was used. The collective dose due to the event is first calculated, and then multiplied by the probability of the event in order to obtain an

expected value for the collective dose. The recommended cost is $ 100 000/manSv saved.

In this case, an accident of the TMI-type is assumed to occur once in every 1500 reactor years and the probability of an on-shore wind is c. 0.5. It is assumed that a leakage of 45 cm^3/sec exists from the beginning. Over 30 reactor years this gives a probability that the event will occur of 0.01 (=0.5 · 30/1500). Since this study a PRA of Ringhals unit 2 has indicated a much lower probability for an accident of TMI-type. Table 2 gives information on costs and consequences for the different alternatives.

DISCUSSION OF UNCERTAINTIES AND CONSERVATISM

The most important uncertainties are as follows:

- The assumptions that 10% of the iodine and 1% of the caesium activity leaking out into the auxiliary building will reach the filters are probably conservative.

- For the weather type, 5% cumulative frequency has been assumed. This means that the collective dose is normally lower.

- As regards collective dose calculations, the greatest uncertainties are in food intake and ground dose. As regards iodine, it is assumed that the milk in the contaminated area has been dumped.

- The probability of an accident of TMI-type is overestimated.

CONCLUSIONS

The conclusions can be summarized as follows:

- Alternative no 2 is excluded since in most cases this exchange gives a higher collective dose (see Table 1, 50% cumulative frequency).

- Alternative 4 involves a very large marginal cost by comparison with a simplified iodine filter according to the note in Table 2.

From a cost-benefit point of view, alternative no 1, no iodine filter is acceptable.

REFERENCES

1. The Collective Dose from an Accident of TMI-type at Ringhals 2. Investigation of the Importance of Different Filter Types (in Swedish).
Studsvik Energiteknik AB NW-81/153 (1982)

2. NUREG-0880. Safety Goals for Nuclear Plants: A Discussion Paper, NRC (1982).

Table 2. Cost-benefit analysis of iodine filter plant from a safety point of view for different alternatives

	Cost Million SEK	Collective weighted whole-body dose (manSv)	Probability /30 years	Cost per saved manSv x probability (Million SEK/manSv)
Value recommended by NRC	-	-	-	0.8
1 No change		10.5	0.01	-
2 Exchange existing absolute filter for coal filter; by-pass during normal operation	2.5	5.1	0.01	46
3 Simplified iodine filter. No redundance	10	0.01	0.01	95
4 Complete iodine filter, largely according to NRC standards	15	0.01	0.01	140

Note Cost-benefit for alternative 4 by comparison with alternative 3 has also been investigated. Since there is no redundance in alternative 3, the probability of a loss of filter functions is $5 \cdot 10^{-3}$. The marginal cost/manSv saved will then be

$$\frac{15-10}{(10.5 - 0.01) \times 5 \cdot 10^{-3} \times 0.01} = 9500 \text{ million SEK}$$

CALCULATION CODES FOR THE OPTIMIZATION OF RADIOLOGICAL PROTECTION SYSTEMS

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INTRODUCTION

One of the requirements of the dose limitation system recommended by the International Commission on Radiological Protection (ICRP)(1) is that exposures to ionizing radiation be kept "as low as reasonably achievable, social and economical factors being taken into account". This requirement implies that radiological protection, when applied to any practice or operation, must be optimized so as to increase protection to a such a level that any aditional increase would imply a social effort greater than the decrease obtained in the detriment. At the dose levels at which optimization is performed, collective effective dose equivalent commitment (further on "collective dose") provides an adequate measurement of the detriment; on the other hand, the cost of protection provides a measure of the social effort involved in the protection. An appropriate balance between changes in collective dose increases and the cost of protection allows for identifying the optimized protection levels.

In some practical cases, the protection systems used in the control of doses incurred by occupationally exposed workers and by the public are independent from eachother, considering that the control of one of them has an influence upon the collective dose due to the other.(2)(3) The objective of optimization is decreasing the collective dose as far as it is reasonable, without any a-priori assessment of the groups of people among which such reduction will take place. Therefore, the optimization of radiological protection must be performed by means of a simultaneous analysis of the various systems under consideration and of the incidence of each one of them upon the reduction of the collective dose.

The optimization of radiological protection, within the framework of the recommendations issued by the ICRP, implies minimizing the summation of the costs of protection, X, and of those of detriment, Y. Both costs depend on the protection level and, therefore, on the value of the exposure control parameters. The protection and detriment costs are connected with eachother by the values of those parameters through functions that constrain the optimization process. The latter may also be carried out within specific limitations. One of these is that individual doses may by no means exceed appropriate limits that must be necessarily lower than the individual dose limits established by radiological protection regulations.

A calculation code is introduced herewith for the optimization of radiological protection in the design of nuclear installations.

CONTROL PARAMETERS

The protection systems that were considered for the development of the calculation code were the following: shields, ventilation, confinement and decontamination.

Shields were aimed exclusively at decreasing the dose rates at which workers, and -to a lower extent- the public, are exposed. The remaining systems are oriented preferently at reducing internal contamination and its resulting collective dose commitment. The purpose of ventilation is to control the concentration of radiactive materials in the working rooms and has no direct incidence upon the doses incurred by the public. Opposedly, the systems for effluent decontamination are aimed exclusively at decreasing the collective dose in the public. Confinement systems, such as glove boxes or leakproof rooms, allow for decreasing the rates of emanation to both the working rooms and the outside environment, consequently decreasing the collective dose in both groups.

The protection parameters involved in the design of the above mentioned systems are the thickness of the shielding, the ventilation flow rate, the retention factors of the confinement systems and the efficiency of the decontamination units. Changes in the thickness of a shielding or in a ventilation flow rate allow for adjusting the protection levels in a more or less continuous manner. On the contrary, the retention factors of the confinement systems vary as per finite increases, depending on the retention alternatives under analysis.

COST EVALUATION

The cost of protection is composed by a capital cost and an operation cost. In some protection systems, the capital cost is predominant (i.e., shields); in other systems, the capital cost is regardless if compared with the operation cost (i.e., certain ventilation and filtering systems). The use of crude costs usually shown in bibliography(2) makes no distinction between those two different situations and produces a distortion in the optimization analysis. For instance, the use of crude costs -for the same reduction of the collective dose- would result in preference for a system in which only capital cost has an incidence against another that -although slightly more costly- would require an investment that may be distributed all along the useful life of the installation. In the code introduced herewith a present worth evaluation is made of all the investments.

The evaluation of investments for a given project involves the analysis of a basically finantial problem. Consideration must not only be given to the fund outflow but also to the time at which those outflows will occur. That is, amounts are not only valid for their nominal value but for their present finantial value. This may be mathematically expressed as follows:

$$VA_{(x)} = \frac{G\ (x)}{(1+r)^{x}}$$

where:

$VA_{(x)}$ is the present value of expenses incurred in period n;
$G(x)$ is the nominal value of expenses incurred in period n;
r is the finantial rate; and,
x is the period under analysis.

For the project as a whole, the total present expenses (TPE) will be:

$$TPE = I_{(0)} + \sum_{x=1}^{n} \frac{G(x)}{(1+r)^x}$$

where $I_{(0)}$ is the initial investment.

The finantial rate applied was calculated, taking into account the finantial yield and the inflationary detriment, as per the following equation:

$$r = \frac{(1 + r_{av})}{(1 + i)} - 1$$

where:

r_{av} is the due yearly yield rate.

Considering that only those decisions taken in the present will modify the total collective dose, no discount rates are considered in the calculation of the cost of detriment. The latter is simply obtained from the product between α and S, where α is worth 10^4 US$/man Sv. The calculation of the marginal collective dose resulting from the application of each alternative protection system is performed as shown in previous papers.(3)

CALCULATION METHOD

The calculation may be divided into three steps: optimization, verification of the accomplishment of dose limits in the public and in the workers.

The code proceeds initially to the calculation of the investment increase elements corresponding with each one of the alternatives under consideration (Δx_i), as from an investment increase element (Δx) used as a basis and in accordance with the finantial analysis described above.

Δx_i is calculated by means of the following equation:

$$\Delta x_i = \left(\frac{\Delta x}{TPE_i} \right) \Delta x$$

Then, the reduction of the effective dose (RED) is calculated, resulting from the application of the Δx_i to the corresponding protection systems.

The code selects the system that allows for a greater decrease of the collective dose and establishes the ratio Δx/RED.

Then, it compares the value of such ratio with the monetary value, α, assigned to the collective dose unit. If the former is lower, it applies a new investment increase Δx and proceeds as described above until the $\Delta x/RED$ ratio approaches α as close as it is possible.

When the calculation process is started, investment increases, Δx, fall upon the protection systems that show continuous variations, such as the ventilation and shielding systems. When the product between the number of increases (n_i) and the cost of each one of them (Δx) reaches the value of a confinement (or decontamination) unit, the programme calculates which would be the reduction of the collective dose if the investment $n_i \Delta x$ would be made on the confinement system. If this reduction is greater than that obtained from the rest of the systems adopted until such time, investment in confinement is decided and the rest of the protection parameters go back to the initial point. If the reduction is lower, the previous procedure is continued until the new value $n_i \Delta x$ reaches the cost of a decontamination unit or that of another confinement alternative.

Later on, the code starts a new step consisting in proving whether the doses absorbed by the most-exposed individuals in the public (e.g., the critical group) do not surpass the established limits. If such condition is not verified, it keeps increasing the investment in finite elements (Δx), as per the maximum efficiency criterion, until it the accomplishment of the corresponding dose limits is reached.

During the last step of the calculation, the accomplishment of dose limits for occupationally-exposed workers is verified. If this is not verified, the code proceeds as in the previous step and selects the system on which the investment is to be made as per the maximum efficiency criterion.

Finally, the parameters that serve for the optimization of protection are obtained from the application of the programme (3): the value of the air flow, the thickness of the shielding, the number of decontamination steps and the adopted confinement alternatives. In those cases in which the accomplishment of the dose limits is most restrictive, the programme provides the value of the protection parameters to be adopted.

BIBLIOGRAPHY

(1) ICRP Publication 26. Oxford, Pergamon Press, 1977. Annals of the ICRP v.1: No. 3, 1977.
(2) ICRP Publication 37. Oxford, Pergamon Press, 1983. Annals of the ICRP v.10: No. 2, 1983.
(3) Segado, R., Lopez Lizana, F. and Reyes, R. IAEA-SM-258/56.

COST-BENEFIT EVALUATION IN A QUALITY CONTROL PROGRAM FOR THE CONVENTIONAL RADIODIAGNOSTIC

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The goal of this research is the comparison between the staff and equipment cost necessary in the quality control program of conventional radiodiagnostic and the benefit obtained in the reduction of the retake of the films and in reduction of the dose to the patients and personell.

For over two years our control program has followed a protocol verified on about 50 X-ray tubes and 25 radiological devices. Its results have already been issued (1,2). RMI test tools and BRH perspex phantoms with 1015 X-ray monitor of MDH Industries Inc. are employed for physical and geometrical procedures.

The present research, based on previous data, works on a limitated number of radiological X-ray tube and accessory (10 radiological equipment), selected as representative sample of type and function; the control procedures are periodically carried out, to improve and make costant the efficiency of the radiological devices. An evaluation of the cost of these procedures is done, too. In the meantime the dose to the patients for radiodiagnostic examinations is evaluated with a trasmission ionization chamber. The comparison between the values obtained before and after the control give the possibility to evaluate the benefits. Moreover, the retake of radiographic films is evaluated before and after the control in order to obtain the benefit of the cost reduction. During the control time, wich lasts a year, there are no variations in the technical characteristic of the personell, in the operational procedures and in he workload.

1) Belletti, S., Catolla Cavalcanti, R., Gallini, R., Giugni, U.: Risultati di un programma di controlli di qualità in radiodiagnostica. La Radiologia Medica (in press)
2) Gallini, R.: Programmi operativi di controlli di qualità per apparecchiature radiologiche. Atti XXX Congresso Nationale SIRMN, Milano 1982.

A MAMMOGRAPHY RISK-BENEFIT ANALYSIS

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INTRODUCTION

A cost-benefit analysis, performed according to ICRP recommendations[1], raises serious problems because it entails comparison of protection levels and non-homogeneous quantities such as detriments. A risk-risk approach would seem to be simpler because comparison is between the detriment incurred by introduction of a specific procedure and the detriment avoided by adoption of precisely that procedure. Mammography is a field in which the risk-risk assessment concept can be applied, as the probability of tumour formation induced by the examination process can be compared with that of finding already-formed tumours. Research aimed at evaluating the advisability and limitations of a generalized mammographic screening has therefore been carried out in Umbria, a region in central Italy.

METHODS

Methods used in a Japanese benefit-vs.-risk analysis of stomach cancer mass screening[2] were adapted to mammography in generalized and selective screening respectively. Loses in life-expectancy for screened and unscreened subjects were compared.

Now, when screening is performed, loss A in life-expectancy, as a function of age, can be expressed as:

$$A = P\left\{S + D\left[f(1-w) + (1-f)(1-w')\right]T\right\} \qquad (1)$$

where: P is the number of people screened; S is the loss due to mammography induction of tumours; D is the incidence rate (supposed constant) of breast cancer per year; f is the ratio of true positive mammographies: true breast cancer cases; w is the five-year survival probability of patients with breast cancer found in mass screening; w' is the five-year survival probability of non-screened patients with breast cancer, and T is average life-expectancy. Where cancer patients die, life-expectancy is assumed to be zero at the time of cancer detection.

Likewise, when screening is not performed, loss B in life-expectancy, as a function of age, can be expressed as:

$$B = PD\left[KS + (1-w')T\right] = N\left[KS + (1-w')\ T\right] \qquad (2)$$

where: K is the ratio between the number of patients with breast diseases and the number with breast cancer in hospitals and N=PD is the number of tumours found in the screened population.

As far as S is concerned, leukaemia induction risk being negligible with respect to breast cancer induction risk, (1) may be expressed as:

$$S = E \cdot R \cdot \Delta M \qquad (3)$$

where: E is the average dose equivalent to the breast; R is the incidence rate of breast cancer per unit dose; ΔM is the shortening of life-expectancy, T, due to induction of breast cancer and is a function of latent period L_1 and incidence period L_2: $\Delta M=0$ if $T<L_1$, $\Delta M=(T-L_1)^2/2L_2$ if $L_1<T<L_1+L_2$, $\Delta M=T-L_1-L_2/2$ if $T>L_1+L_2$.

Different rescreening intervals are taken into account by repeated use of Eq.(3). Therefore, screening is justified when B>A.

DATA SOURCES

Data relative to population resident in Umbria and average life-expectancy as a function of age group were derived from[3,4]. The incidence rate per year of breast cancer in the whole population was estimated by a statistical model based on breast cancer mortality in Umbria[5].

The selected population consisted of about 7 000 spontaneous women clients of a mammographic centre in Perugia who were then selected for mammography on the basis of physical examination results and common risk factors. From clinical reports of the mammographic centre, numerical values were derived for: incidence of breast cancer in the selected population as a function of age group; ratio of true positive mammographies to true breast cancer cases (f=0.92); breast volume and density conditioning mammographic exposure, as a function of age group; five-year survival probability of patients with breast cancer found in the screening (w=0.72); five-year survival probability of non-screened patients (w'=0.60) as derived from follow-up data on patients with breast cancer detected by physical examination alone.

Average dose equivalent to the breast, as a function of age group, was evaluated by measurements obtained from Rando female phantom with TL dosimeters, under different technical conditions depending both on X-ray films used (with or without screens) and on volume of breast. Results showed breast dose per view ranging from 2 to 23 mSv (for non-screen films) or 0.4 to 4 mSv (for screen films), a negligible gonadal dose and a red bone marrow dose ranging from 0.25 to 1% of incident dose. The risk for breast cancer induction has been estimated to be $5 \cdot 10^{-3}$ Sv^{-1}, calculated according to ICRP No.26[1].

Latent period L_1 of radiation-induced breast cancer was taken to be 15 y and incidence period L_2 to be 30 y[2].

RESULTS

Different screening strategies, for breast cancer prevention by mammographic examination, are compared herebelow in terms of estimated cumulative benefit, defined as the sum of age-specific benefits over all the n age classes considered:

$$C = \sum_i (B_i - A_i) \qquad i = 1, \ldots, n. \qquad (4)$$

Strategies differ by the age of the first examination (16 to 76 years), periodicity of rescreenings (1-5 indicates years between two subsequent examinations; 0 indicates single examination with no repetition), X-ray dose used and by target population (selected high-risk population, general population).

To give an idea of how losses in life-expectancy combine to provide benefit (B-A), A and B, as defined in Eqs.(1) and (2), are reported in figs.1 and 2 as functions of general population age when screen films are used.

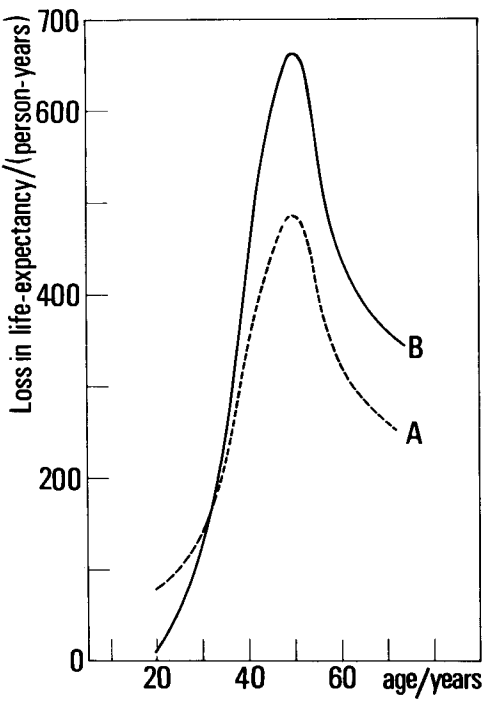

Fig.1 - Loss of life-expectancy in general population (A: single screening at age 16 y; B: no screening).

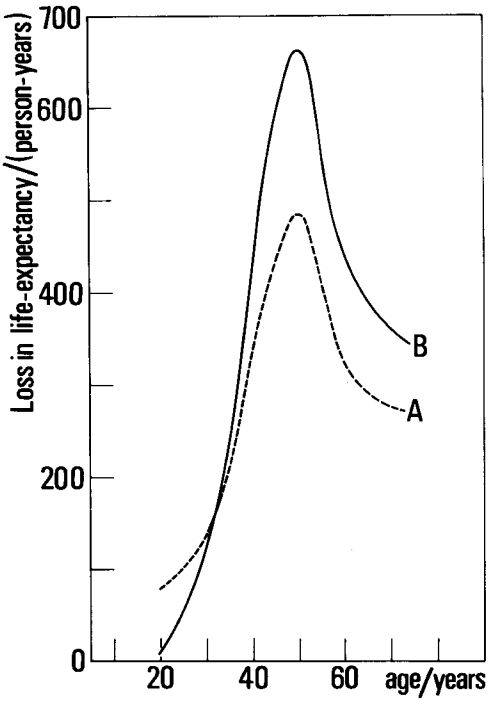

Fig.2 - Loss of life-expectancy in general population (A: screening with two-year interval, starting age 56 y; B: no screening).

For periodical screenings it has been assumed that a single examination also occurred in the population at age 16.

Table 1 show the estimated values of benefit for the different screening strategies considered for high-risk groups and the general population. Note that for single examination with no repetition it has been assumed that no other individual mammographic examination might ever occur in the population.

When the high-risk population is considered, maximum benefit is obtained for single examination at age 31, no matter what dose is used. As concerns the general population, maximum benefit is obtained again for single examination screening only at age 36 with screen films and at age 46 with non-screen films.

TABLE 1
Cumulative benefit/(person-years) of mammography screenings as a function of age at first examination and periodicity of rescreenings for two different commonly used X-ray doses.

age at 1st examination/years	Cumulative benefit/(person-years)							
	screen films				non-screen films			
	periodicity/years				periodicity/years			
	1	2	5	0	1	2	5	0
	a) selected high-risk population							
16	270.3	266.5	264.9	276.1	256.5	242.6	236.7	275.9
21	260.5	257.7	256.3	276.1	221.2	209.8	204.8	275.9
26	253.8	251.6	273.1	276.1	195.3	186.3	182.3	275.9
31	274.5	272.4	269.4	276.2	268.9	262.5	187.1	276.0
36	272.0	256.9	256.6	265.7	259.4	200.8	199.2	265.5
41	272.4	261.2	261.0	195.8	260.4	217.2	216.0	195.7
46	267.3	267.2	267.2	158.8	240.9	240.4	240.2	158.7
51	270.2	270.2	270.0	122.9	252.2	252.1	252.0	122.9
57	274.1	274.1	274.1	92.3	267.7	267.7	267.7	92.2
61	275.2	276.2	276.1	43.7	272.0	271.7	271.7	43.7
66	276.1	276.1	276.1	18.7	273.3	273.3	273.3	18.7
71	276.1	276.1	276.1	5.6	275.8	275.9	275.9	5.6
76	276.1	276.1	276.1	1.8	275.9	275.9	275.9	1.8
	b) general population							
16	-10900	-6244	-2194	492	-45568	-22659	-8914	250
26	-5598	-2514	-706	492	-22879	-11314	-4376	250
36	-1460	-487	-729	500	-9560	-4655	-1712	353
46	82	287	410	404	-1784	-767	-157	361
56	458	475	485	226	92	171	219	220
>65	492	492	492	103	250	250	250	102

CONCLUSIONS

Periodical generalized mammographic screenings seem to be justified when started at age 45 or over if screen films are used, and at age 55 or over if non-screen films are used. Single examination without repetition is always justified giving the absolute maximum benefit at ages 36 and 46, for screen and non-screen films, respectively. When a high-risk population is selected, any kind of screening appears to be justifiable, even though the absolute maximum benefit is reached when single, no-repetition examination is performed at age 31 for both doses considered.

REFERENCES

(1) ICRP Publication No.26, 1977.
(2) T.A.Iinuma et al., Proc.6th Int.Conf.Rad.Res., Tokyo, 1979.
(3) G.Caselli and V.Egidi, Istituto Demografico, Università di Roma, 1980.
(4) Suppl.Boll.Mensile di Statistica, No.11, 1981.
(5) A.A.Klementiev, IIASA Report RM-77-43, 1977.

METHODS FOR ASSESSING THE ECONOMIC COST OF EMERGENCY COUNTERMEASURES FOLLOWING A NUCLEAR ACCIDENT

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The National Radiological Protection Board has developed a series of interlinked models to predict the transfer through the environment of radionuclides released in an accident, and subsequent doses to and effects on man. The Methodology for Assessing Radiological Consequences (MARC) from atmospheric releases predicts the resulting dose distribution and health consequences in the population, and the economic impact of restrictions which may need to be made on the use of contaminated land[1,2]. Techniques for estimating the economic cost of emergency countermeasures are described in this paper, and some examples of their potential application given.

Countermeasures which could have significant economic effects include the banning of consumption of foodstuffs grown in an affected area, and a temporary or more permanent ban on living and working in the area. It is therefore useful to establish the cost of applying these countermeasures, and the corresponding reductions in radiation detriment, as an input to decisions on criteria used for invoking countermeasures. A method for assessing costs incurred by countermeasures is described below.

THE ECONOMIC COST OF COUNTERMEASURES

The economic definition of cost is given as a lost benefit[3] and, on this basis, the cost will be the lost opportunity to use land for agricultural, industrial and residential purposes. A measure of this loss would be the contribution made to the nation's Gross Domestic Product (GDP) in the area affected by the accident. The model assumes that the contribution to GDP will be lost when countermeasures involving relocation are necessary, or when a food ban is imposed. The costs of actually implementing these countermeasures, such as transporting people away from an area, have to be estimated separately.

Gross Domestic Product is a measure of the value of goods and services in an economy; UK statistics are compiled annually by region according to a standard industrial classification. GDP can be measured as output, income or expenditure. A rise in GDP is usually an indication of an overall increase in living standards in a country, but the statistics used do not include all activities having an economic benefit. A loss of GDP cannot therefore be used as a complete measure of the cost of countermeasures; it is only a measure of the loss of production.

Estimation of Costs

The economic cost of countermeasures can be calculated from the end points evaluated in MARC, that is the area and number of people evacuated or relocated, and any agricultural restrictions. An annual average contribution to GDP by each person has been deduced from national statistics (Table 1). This is then multiplied by the number of people who have to be evacuated and relocated following an accident, and the length of time for which the restriction is in force. It is sometimes necessary to estimate the impact on agriculture and housing separately and these contributions are therefore not included in the GDP per person value.

Agricultural losses are evaluated using information on the number of animals and area of crops in each 5 x 5 km square of the UK. From national statistics, it is possible to estimate the contribution to GDP from animal and crop production, and

therefore assign GDP per animal and GDP per hectare of crop for the purpose of estimating losses. The value of milk production is calculated separately but in a similar manner, because for some contamination events milk may be the major foodstuff banned (Table 1).

Although the occupation of houses does not contribute to production in the obvious way that agriculture does, a significant proportion of income may be spent on rent or mortgage repayments, and therefore appears in GDP statistics. These figures are applied to a data base which contains the number of households in each 1 x 1 km square in the UK. The loss of housing stock for the period of an entry restriction is hence calculated.

In order to calculate the present value of the cost of countermeasures, the loss to GDP should be discounted over the period that the countermeasures will be in force. In the UK, HM Treasury recommend a discount rate of 5% per year.

The method described can be used to calculate the economic costs of emergency countermeasures taken to reduce radiation exposure. There are a number of potential applications of this method for aiding decisions such as those on emergency planning criteria. A technique which can be used to indicate the annual individual dose levels at which to introduce food restrictions following an accident is described below.

AN EXAMPLE OF APPLICATION IN EMERGENCY PLANNING

In this example, the accidental release chosen to illustrate results is typical of a large degraded core accident in which approximately 1% of the volatile fission products are released. The wind is blowing 300°N from a UK nuclear site on the east coast in average UK weather conditions and no rain. Food restrictions are based on postulated food intakes by adult critical group members. The results given are presented solely to illustrate the potential use of the method described. Undue significance should not be attached to the particular numbers presented as they relate to an assumed release in defined circumstances.

Figure 1 illustrates a typical curve for the cost of banning agricultural produce, in this case crops, and the corresponding collective dose at a range of annual individual dose limits. The general principle of diminishing returns is illustrated with a progressive rise in cost as the collective dose decreases. A measure of the cost effectiveness of each dose criterion may be calculated by comparing the ratios of cost to collective dose saved between dose criteria. The gradient dX/dS of the cost curve, describes the marginal cost effectiveness in £ per man Sv, where X is the cost of the crop ban, and S is the collective dose. This is illustrated in Figure 2 as a function of collective dose. If a value for avoided health detriment has been pre-determined, the dose level at which to introduce the crop restriction could be obtained from this graph. For instance, for a value of £20,000 per man Sv, banning milk around 0.5 or 1 mSv y^{-1} could be advocated. The curve also shows that the most collective dose is saved for the least cost at the least restrictive dose criterion, 50 mSv y^{-1}. This implies a cost of only £500 per man Sv avoided.

The information obtained from MARC could also be used to draw curves from which the optimum dose level at which to restrict a particular foodstuff would be indicated for a specified value of health detriment avoided. Figure 3 shows X, the cost of the food ban and Y, the cost of the health detriment at £20,000 per man Sv. X + Y reach a minimum at an individual dose criterion of 1 mSv y^{-1}. A similar result was obtained from Figure 2, but in a less direct way.

There are, however, difficulties in using these graphs to select a dose criterion at which to introduce food restrictions, not least in deciding on a cost

per unit collective dose to use in accident situations. However, for the purposes of an example here, a range of valuations have been used for the health detriment avoided, incorporating those recommended by NRPB for routine releases[4]. The optimum dose level at which to ban each foodstuff, as illustrated in Figure 3, was calculated for each valuation of £ per man Sv and the results plotted in Figure 4. In all the cases considered, the optimum level at which to ban milk is less than 1 mSv y^{-1}, while the corresponding levels for crops and meat are higher. However it should be stressed that this is just an example of application, which has been simplified for presentational purposes. In emergency planning, the results for the same release in different meteorological conditions would have to be considered, along with the consequences of other releases, in a defined methodological framework. Needless to say, there will be other important factors to consider when banning foodstuffs, and decisions are unlikely to be made purely on the basis of individual dose levels predicted. Even if they were, the methods described here would still have a role to play, by indicating the cost per health effect avoided which would be implied by such decisions.

CONCLUSIONS

A method has been described to estimate the economic cost of emergency countermeasures taken to reduce the radiation exposure of a population after an accident. There are a number of potential applications. An example in emergency planning has been given; estimates of cost could assist decision making about the levels of contamination at which countermeasures should be introduced. The method could also be used in the design and planning of nuclear installations, where the cost of safety features required to mitigate the effect of accidents could be compared with the cost of any potential accident. Finally, the method could be used to estimate the relative impact of countermeasures at different sites, providing input to siting decisions.

Table 1 Contributions to UK GDP, 1978 UK prices

GDP (£ per person)*	2374	* excludes contributions
Crops (£ per hectare)	18,400	from housing and
Animals (£ per animal)	41	agriculture
Milk (£ per litre)	0.04	

ACKNOWLEDGEMENTS

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REFERENCES

1. Kelly G N and Clarke R H. An Assessment of the Radiological Consequences of Releases from Degraded Core Accidents from The Sizewell PWR, NRPB-R137 (1982).

2. Clark M J and Dionian J. ECONO-MARC : A Method for Assessing the Cost of Emergency Countermeasures after an Accident, Chilton, NRPB-M85 (1982).

3. Lipsey R G. An Introduction to Positive Economics, 5th Edition London, Weidenfeld and Nicolson (1979).

4. Cost Benefit Analysis in Optimising the Radiological Protection of the Public. A Provisional Framework. Chilton, NRPB, ASP4 (1981).

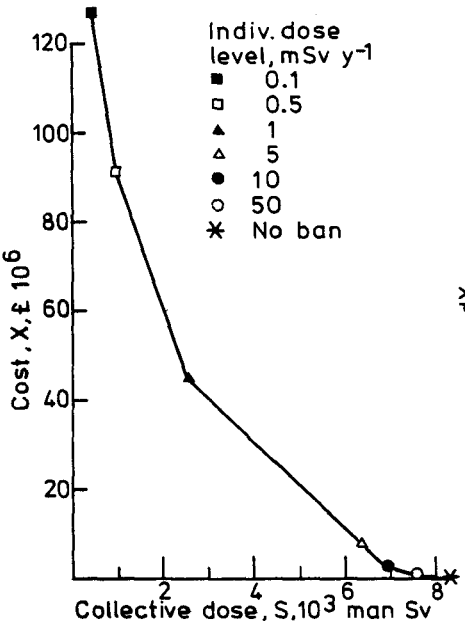

Fig 1 Cost of a crop ban

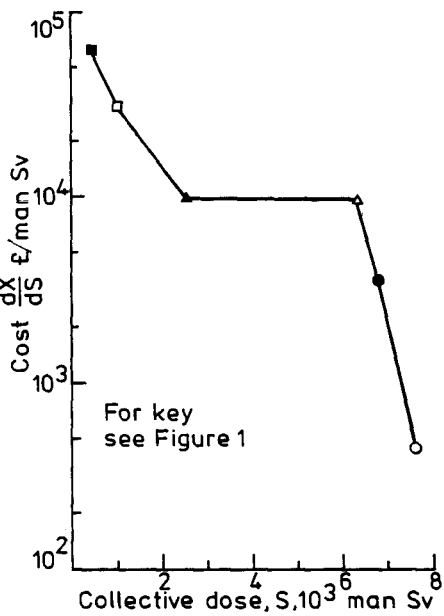

Fig 2 Cost effectiveness of a crop ban

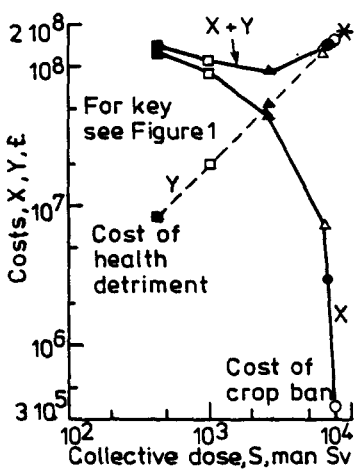

Fig 3 Optimisation of crop ban using £20,000 per man Sv

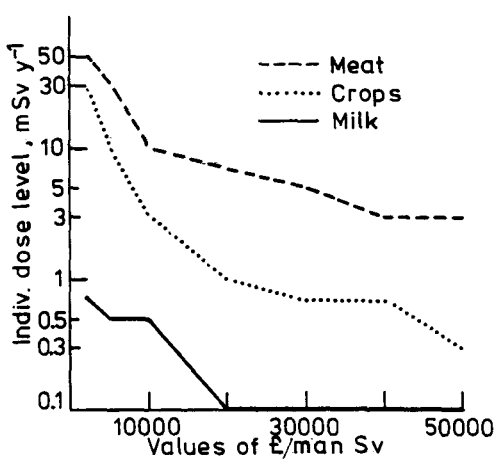

Fig 4 Optimum values for food bans

OPTIMISATION OF PROTECTION OF RADIATION WORKERS

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INTRODUCTION

Under the current ICRP system of dose limitation, the principle of optimisation of protection has become established as a major objective of practical radiological protection; requiring all doses to be reduced to levels which are As Low As Reasonably Achievable or "ALARA" within the constraint of dose limits. Application of ICRP recommendations is the responsibility of national authorities and, in the UK, the National Radiological Protection Board has already advised on a provisional framework for optimising the exposure of the public, using the technique of cost benefit analysis[1]. This framework contains recommendations in a number of areas, including a costing scheme for the monetary valuation of radiation-induced health detriment.

In presenting this advice on the optimisation of protection of the public, it was stressed that the recommendations, and particularly the numerical suggestions on the monetary values of collective dose, should not be applied to occupational exposure[1]. The Board nonetheless recognised that for full implementation of cost benefit analysis as an input to decisions on ALARA, the framework would need extending to include radiation workers, and proposals for achieving this objective have recently been outlined in a consultative document[2]. As with the Board advice for members of the public, this concerns only routine exposures during normal operations, but it is intended to apply to all those occupationally exposed, whether in the nuclear industry, medicine, general industry, research or elsewhere. This paper describes some of the factors which influenced the proposals and their implications for practical optimisation studies.

THE ROLE OF COST BENEFIT ANALYSIS

The application of cost benefit analysis to radiological protection as described in ICRP publications[3,4] has been couched in considerable mathematical formalism. While this presentation can provide a precise analytical framework for the implementation of ALARA, in other respects it can give the wrong impression. It may, for example, imply a degree of complexity such that the technique appears to be relevant for only major radiological problems. It may also encourage the idea that quantitative cost benefit analysis is all that is required for decision-making. To dispel such misconceptions, it is important to outline the role which we envisage for cost benefit analysis in the optimisation of protection of radiation workers.

Cost benefit analysis provides a straight-forward framework for optimisation. The necessary inputs are assessments of the cost of radiological protection and radiation-induced health detriment for alternative courses of action, so that the choice which minimises the sum of these costs (or equates 'marginal' costs) can be identified. Most of the effort in providing these inputs is in the assessment of protection costs and dose reductions associated with each control option considered; these assessments should be carried out in any case before coming to a decision. For large scale problems, perhaps involving complex or novel technologies, this assessment can be a specialised exercise embracing alternative plant designs and the use of mathematical models for dose calculations. But it must be emphasised that the level of sophistication and corresponding degree of effort given to the assessment should reflect the scale of the radiological problem being considered. For small-

scale operational problems, rough estimates of costs and dose reductions by professional health physicists are entirely adequate.

The principal advantage in applying decision-aiding techniques, such as cost benefit analysis, is not that the need for judgements is removed but that the judgements are clarified and made consciously. This feature is particularly relevant to the evaluation of health detriment costs in the Board framework, which reflects broad socio-economic considerations in addition to matters of science. The inherent subjectivity in quantifying such considerations should always be kept in mind, to counter the impression of numerical accuracy that may otherwise be implied by these valuations. Furthermore, it must be emphasised that cost benefit analysis will not be the sole means by which specific decisions on radiological protection are taken, if only for the reason that it will rarely encompass all of the judgements involved. For example, the monetary valuations of health detriment put forward by NRPB will be generic, relating to 'average' radiation workers in 'typical' working environments. But the practical management of occupational exposure is concerned with the protection of specific individuals and may need to account for such factors as their age, sex and cumulative exposure. The Board framework for optimisation should therefore be regarded as more relevant to the design of installations involving notional workforces and to broad operational decisions concerning working practices. It will be less applicable to day-to-day decisions, although even here it may have a role to play, allowing health physicists and others to examine the implications of their decisions on ALARA. In all these contexts, the valuations of health detriment will provide a guideline against which actual expenditures on dose reductions may be compared.

PRINCIPLES UNDERLYING THE VALUATION OF COLLECTIVE DOSE

It is clear from ICRP publications that the assumption made for protection purposes is that the health detriment in irradiated populations will be directly proportional to the collective dose received. It is then tempting to go on to assume a linear relationship between the cost of this health detriment and the collective dose, but this simple costing procedure (implying a constant cost of unit collective dose) would ignore some very important features of practical radiological protection. These include the existence of dose limits and equity considerations which have led health physicists in general to attach considerable importance to high individual doses, even where few individuals are involved so that the collective dose is small. While it is certainly possible to carry out optimisations using a constant cost of unit collective dose, judgements on the relative significance of the individual doses involved are still likely to be necessary before reaching decisions on ALARA. We believe that more consistent decision-making will result by incorporating such judgements within the numerical costing scheme for health detriment. This may be achieved by assigning a higher cost to unit collective dose made up of higher individual doses, particularly those approaching the dose limit, as recommended by the Board for exposure of the public[1]. The effect of this scheme is to concentrate resources on the protection of those individuals who are most highly exposed, thereby achieving compatibility with the practical emphasis in radiological protection on critical groups.

This type of costing scheme appears even more appropriate for radiation workers, as in general, occupational exposures are closer to the dose limits. Furthermore, in the working environment, judgements on the relative importance of individual and collective doses are often reflected in decisions taken by management. Cost benefit analyses of these practical situations will be significantly influenced by the manner in which collective doses are converted into monetary terms. A constant conversion factor (i.e. a single value for the cost of unit collective dose) will reduce the cost benefit procedure to the minimising of collective dose. This can be shown in

many cases to lead to solutions in which the "optimum" is to expose the smallest number of people as close to the dose limit as possible. In contrast to this, a conversion factor which increases with increasing individual dose levels will discriminate to some extent against high individual exposure and lead to a more equitable distribution of risk; under operational conditions where say, individual doses can be controlled by varying the number of workers assigned to a specific task, it will allow reasonable trade-offs to be made between the sometimes conflicting objectives of reducing the individual and collective doses.

COSTING SCHEMES FOR OCCUPATIONAL EXPOSURE

In order to proceed from these general principles to a system for practical application, it is necessary to assign monetary values to unit collective dose as a function of the individual doses involved. In theory, this relationship should take the form of a progressively increasing or convex curve. However, the use of such a continuous function in assessing health detriment costs would require rather precise knowledge of the individual dose distribution. This knowledge might be available for existing operations based on monitoring results for individual workers. But optimisation studies are concerned with changes in radiation exposure arising from changes in the level of protection and this requires prediction of the associated dose distributions.

A requirement for detailed knowledge of potential dose distributions is regarded as impractical, especially for design studies involving new installations. The costing scheme put forward by the Board therefore assigns values to unit collective dose within various annual individual dose bands, providing a 'step' function as an approximation to a smooth curve. This requires a number of judgements to be made on both monetary valuations and the range of dose bands selected. A principal purpose of the consultative document was to solicit comments on the numerical values suggested, so that these judgements could be refined in the development of formal Board advice. In the event, responses were obtained from a wide range of organisations and individuals, including some from outside the UK, and the proposed scheme is currently being reviewed in the light of these comments.

Among other factors influencing these proposals was the recognition that for some decisions, such as whether to install additional radioactive waste treatment at nuclear sites, reductions in doses to the public may be accompanied by increases in doses to radiation workers, so that comprehensive optimisation studies would involve trade-offs between the two groups. To address these practical situations within a common framework implied a need for broad compatibility between the costing scheme for occupational exposure and previous Board advice for members of the public, and these two costing schemes are shown in Tables 1 and 2, respectively.

It can be seen from these tables that both schemes incorporate three annual individual dose bands, with changes of a factor of 5 in monetary values between adjacent bands. However for radiation workers, the choice of individual dose bands reflects the fact that those occupational exposures which are measured or assessed fall within a range of about two orders of magnitude. The bands are therefore more tightly grouped than were the corresponding bands for members of the public, whose predicted doses may be spread over many orders of magnitude and seldom approach the dose limit. Furthermore, to account for the higher levels of individual dose received by radiation workers greater relative weight is assigned to occupational doses approaching dose limits. In contrast to this, public exposures are generally given greater relative emphasis at individual dose levels below 5mSv, except at very low dose levels. This reflects considerations of both risk aversion and equity, since not only are workers better informed and more familiar with the nature of

radiation risks, but unlike members of the public, they have an element of choice in accepting such risks for the personal benefits attained through employment.

CONCLUSIONS

The proposals outlined in this paper represent a step in the development of formal Board advice on the use of cost benefit analysis in the optimisation of protection for radiation workers. They include suggestions on the costs to be assigned to collective dose received by workers for comparison with costs of protective measures and, in some cases, the costs associated with public exposures. It should again be stressed that the cost benefit process is intended as an input to decisions on ALARA and not as a prescription for making such decisions. Nonetheless, by structuring the application of judgements in this quantified manner, it is hoped that the Board framework for optimisation will assist those with responsibilities for the radiological protection of workers to more consistently implement the ALARA requirement.

REFERENCES

1. Cost benefit analysis in optimising the radiological protection of the public. A provisional framework. Chilton, NRPB, ASP4 (1981).

2. Cost benefit analysis in the optimisation of protection of radiation workers: A consultative document. Chilton, NRPB, (1982).

3. Recommendations of the ICRP. Oxford, Pergamon Press, ICRP Publication 26. Ann. ICRP, 1, No. 3 (1977).

4. Cost benefit analysis in the optimisation of radiation protection. Oxford, Pergamon Press, ICRP Publication 37. Ann. ICRP, 10, No. 2/3 (1983).

Table 1 Proposed costs of unit collective dose for radiation workers[2]

Individual annual dose band (mSv)	Percentage of annual dose limit for radiation workers, 50 mSv	Cost* of unit collective dose made up of individual doses within the band (£/man Sv)
< 5	< 10	4,000
5 - 15	10 - 30	20,000
15 - 50	30 - 100	100,000

* In 1980 prices

Table 2 Recommended costs of unit collective dose for members of the public[1]

Individual annual dose band (mSv)	Percentage of annual dose limit for members of the public, 5 mSv	Cost* of unit collective dose made up of individual doses within the band (£/man Sv)
< 0.05	< 1	2,000
0.05 - 0.5	1 - 10	10,000
0.5 - 5	10 - 100	50,000

* In 1980 prices

ON THE IMPLEMENTATION OF THE SYSTEM OF DOSE LIMITATION FOR THE SAFETY OF WORKERS AROUND RADIATION SOURCES AND REACTORS

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ABSTRACT

The system of dose limitation for the safety of workers around radioactive sealed and unsealed sources and nuclear in-radiological protection (comptant authority).

In the present work, general description of the Egyptian comptant authorities, the need of health physicists to implement the nuclear energy programme and how to attract new ones, the annual dose equivalent for occupational workers, the monetry value for unit dose equivalent and setting release limit are discussed.

INTRODUCTION

In 1977 the system of dose limitation was developed by the International Commission on Radiological Protection (1). When the International Atomic Energy Agency and International Organization decided to implement the system of dose limitation two Radiation Protection experts from Egypt took part in the expert group meetings.

When the basic safety standards for radiation protection was published by IAEA in 1982 (2), the system of dose limitation was proposed by GOMAA to be implemented at the Egyptian Atomic Energy Authority. In May 1983, the Atomic Energy Authority of Egypt recommended the implementation of the system of dose limitation to the central Authority for Radiological Protection of Egypt.

In June 1983, The Central Authority for Radiological Protection of Egypt (3) approved the implementation of the system of dose limitation in Egypt in principal.

The Egyptian low for radiological protection (1960) (4) indicate that the all radiological protection activities should be controlled by the Ministry of Health or the Atomic Energy Authority of Egypt. Along side with the radiological protection low the Egyptian regulations for radiological protection (1963) (5) was based on IAEA and ICRP publications. Four working groups are

formed in July 1983 to rewrite the Egyptian regulations according the system of dose limitations. The working groups are grouped according to the type of exposure as occupational, medical, members of the public and emergency and accidental.

THE EGYPTIAN COMPTENT AUTHORITY

During 1982, the nuclear safety commission of Egypt is formed. In 1983 the commission is renamed as the nuclear regulatory and safety commission of Egypt (6).

The commission is responsible from the control of open sources and reactors as well. The research centres of the Atomic Energy Authority are responsible on radiological protection of its workes.

The radiation protection committee of the Nuclear Regulatory and Safety Commission are responsible on the formulating regulations, codes and guides for radiation protections and to ensure that the regulation are implemented within the research centres the Atomic Energy Establishment and outside. The AEE is issuing liscence for open sources and research and power reactors after recommendations by the NRSC.

The radiation protection Exactive Bureau of the Ministry of Health is controlling the use of sealed sources, X-ray machines and accelerators. The technical committee for radiation protection of the same Ministry is issuing Liscences for persons and the working places.

The national comptant Authority of Egypt is the Central Authority for Radiological Protection. The main objective of the Central Authority is to lay down the national radiation protection policy and to suggest the up dating the radiological protection law. During the June Meeting the CARP approved the use of Becquerel as a unit of activity, the Gray as a unit of absorbed dose and the seivert as a unit of dose equivalent along side with the special units of Curie, rad. and rem.

THE NEED FOR HEALTH PHYSICISTS

The future plans for the nuclear energy in Egypt includes the following activities:

(1) Eight nuclear power plants by the year 2000, and
(2) Uranium mining,

Health physicists are needed for participating in:

(a) Review and assessment of the nuclear safety reports for nuclear installations.

(b) Liscensing of nuclear installations.
(c) Operation of the nuclear installations as inspectors.

Health physicists are also needed for radiation protection work at Nuclear Power Plant and the Research Centres of AEE and for controlling the use of open sources for medical and other purposes. For the growth use of sealed sources and accelerates for medical and radiographic purposes the Ministry of Health is also in need for health physicists.

In Egypt, Radiation Protection work is done by radiation protection experts and health physicists.

Radiation protection experts in Egypt are physicists granted Ph.D. in Physics with two years of experience in the field of radiation protection or health physicists having five years of radiation protection experience.

While health physicists are physicists with B.Sc. in Physics with one year diploma in radiation physics with two years experience in radiation protection.

In order to attract physicists to be health physicists a new diploma in health physics is now recognized at Cairo University. Furthermore, a new department is now recognized at the Faculty of Science of Cairo University for Biophysics. In these courses the system of dose limitation is part of the teaching programme.

Alongside with the previous courses training programmes in radiation protection and the use of ionizing radiation for various purposes are organized by the Atomic Energy Establishment Centres and the Ministry of Health for techaians and scientists.

ANNUAL DOSE EQUIVALENT

For the last 20 years the average annual dose equivalent for occupational workes of the Atomic Energy Establishment in Egypt is 2.50 mSv (7). The system in use is the film badge and nuclear emulsion. Future plans is aimed to reduce cases of exposures greater than 10 mSv by appling the ALARA principal and for personnel dosimetry the systems of TLD and SSNTD which were experimented for more than ten years shall be in use for occupational exposure.

OPTIMIZATION FOR PROTECTION

In order to apply the ALARA principal, a monetary value for unit of collectine effective dose equivalent (∝) should be assessed.

Following Webb's (8) defination:

$$\alpha = 0.4\ L + C \qquad (1)$$

where ICRP estimated a total loss of life expectancy of 0.4 man year per man Sv, and L is the annual mean per Caput income, and C is the cost of additional medical care per man Sv.

$$\text{In Egypt } I = \text{L.E. } 810 \qquad \text{US\$ } 1000 \qquad (2)$$

where C = cost of medical care x induction rate for cancer, i.e.,

$$C = \$\ 10{,}000 \times 10^{-2}\ Sv^{-1} = \$\ 10^{2}\ Sv^{-1} \qquad (3)$$

The value of α = 500 US$ per sievert.

In practice this value is not enough to buy radiation protection equipments since these equipment are usually made at the developing countries, so, the value I recommended should be ten times the suggested one.

SETTING RELEASE LIMITS

For setting release limits (9), the Egyptian recommendations are still under study and for the critical group the following relation is developed:

Annual exposure = $\leftarrow$ annual dose equal limit,

where $\leftarrow$ is 1/100,
where the annual exposure is due to external and internal exposures.

REFERENCES

(1) ICRP-26, 1977. Oxford, Pergamon Press.
(2) IAEA, Safety Series No. 9, 1982.
(3) Minutes of the meeting of Egyptian Central Authority for radiological protection, 1983.
(4) Egyptian Low No. 59, 1960.
(5) Egyptian Regulation for Radiological Protection, 1963.
(6) Egyptian Atomic Energy Authority, order No. 4, 1983.
(7) M.A. Gomaa, First National Conference on Biophysics, Cairo, Egypt, 1982, 145.
(8) M.A.M. Webb, IAEA Working Document (not edited) on the value of unit collective dose ..., Vienna, April, 1982.
(9) IAEA Safety Series No. 45B, 1982.

COMPARISON OF RADIATION PROTECTION PROGRAMS AT U.S. POWER REACTORS, URANIUM MILLS, AND LOW-LEVEL WASTE DISPOSAL SITES

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Abstract

The accident at the Three Mile Island nuclear power reactor in March 1979 and subsequent investigations identified serious concerns involving several aspects of radiation protection programs in general. Battelle, Pacific Northwest Laboratories was contracted by the United States Nuclear Regulatory Commission to characterize and evaluate radiation protection programs at power reactors, uranium mills and commercial low-level waste disposal sites in the United States. These evaluations were termed appraisals because they were structured to facilitate an integrated look at the total radiation protection programs, delve into areas for which explicit regulatory requirements did not exist, and emphasize evaluation of capability and performance rather than compliance with regulations.

This paper contains some of the results of 48 power reactor appraisals, 10 uranium mill appraisals and 3 commercial low-level waste disposal site appraisals. The appraisal scope and methodology as well as summary findings and conclusions will be discussed. It was observed from this effort that there is a difference in the adequacy of radiation protection programs as compared between the three types of nuclear facilities. It was observed, based on the risks involved, that the program elements at low-level waste disposal sites and power reactors were substantially better than at the uranium mills.

Introduction

On March 28, 1979, Unit 2 of the Three Mile Island (TMI) Nuclear Power Plant experienced the most severe accident in the operating history of commercial nuclear power plants in the United States. Preparation for such an event by the station staff and the radiation protection group was deficient in several respects that led to a less-than-satisfactory response to a real radiological emergency situation (NUREG-0600).

As a result of the Three Mile Isalnd accident and the resultant problems identified in the radiation protection program, the Nuclear Regulatory Commission (NRC) undertook a major effort to analyze the radiation protection programs at 48 commercially operated nuclear power plants and 10 NRC licensed uranium mills. This effort, called

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the Health Physics Appraisal Program (HPAP), was initiated to determine whether the nuclear power plants and uranium mills had adequate radiation protection programs and whether they had incorporated the lessons learned from the TMI accident in the area of radiation protection. A second objective was to identify generic radiation protection problems in order to make improvements in NRC regulations, requirements, and guidance. The NRC contracted with the Pacific Northwest Laboratory (PNL) to accomplish this task.

The concept in developing the Health Physics Appraisal Program was to institute a means for performing a comprehensive evaluation of the overall adequacy and effectiveness of the licensees' total health physics programs. Whereas the previous inspection program was more compliance oriented and led to the inspection of health physics programs by discrete subject areas, the appraisal program was structured to facilitate an integrated look at the total program. The criteria for evaluating the licensees' program elements were taken from technical specifications, NRC rules and regulations, and NRC regulatory guides, as well as ANSI* standards and ICRP/ICRU** recommendations, and in some cases where no published guidance was available, the professional judgment of the appraisal team members.

The NRC also contracted with PNL to ensure that current and future exposures were maintained "as low as reasonably achievable" (ALARA) at the three operating commercial low-level waste (LLW) disposal sites in the U.S. To accomplish this task PNL also used a HPAP approach to gather information and make an evaluation of program adequacy. The criteria for evaluating the licensee's program elements were again taken from the guides mentioned above and 10CFR61 (Licensing Requirements for Land Disposal of Radioactive Waste).

Health Physics Appraisal Program

The HPAP was structured using a systematic methodology that consisted of analytical trees with applicable questions for each tree. The analytical trees provided a graphic depiction that aided in the deductive analysis of a total system and provided a logic display of interrelationships. The questions were designed to define the scope of the appraisal and to ensure consideration of the essential elements of a radiation protection program. The questions were not an all-inclusive listing of significant items. Thus the HPAP teams were expected to use professional judgment and be flexible, as the need arose, in the application of the guidance and use of the analytical trees.

To implement the HPAP, eight power reactor appraisal teams were formed. The basic team was composed of three to five professional health physicists, including a senior NRC health physics inspector as a team leader, one contractor health physicist, and one PNL health physicist. On some of the appraisals, other NRC health physicists served as additional members. The inclusion of a contractor and PNL health physicist added an extra dimension of perspective and proved beneficial. The Mill Appraisal Program involved two appraisal teams each consisting of one NRC inspector as team leader, one health physicist from the NRC's Uranium Recovery Licensing Branch, and two

*American National Standards Institute
**International Commission on Radiological Protection/International Commission on Radiation Units and Measurements.

health physicists from PNL. Finally, the LLW disposal site appraisals involved one appraisal team consisting of three to four PNL health physicists.

A team approach was selected for several reasons. Because of the broad scope of the program, it would have taken too long for a single individual to perform the inspection and complete the appraisal schedule. Furthermore, the interaction between members was particularly desirable because many evaluations were necessarily based on professional judgments. Also, the interchange of concerns among team members and discussion of apparent weaknesses often helped clarify the real problem area or cause of the symptomatic deficiency. For purposes of the appraisal the areas of the health physics program evaluated were:

- radiation protection organization, and management;
- personnel selection, qualification and training;
- exposure control, external and internal;
- respiratory protection;
- surveillance;
- radioactive-waste management (tailings management for mills);
- ALARA program; and
- facilities and equipment.

One or more analytical trees with corresponding questions were developed for each of these major parts. The analytical trees start with a single desirable condition and systematically proceed through lower levels or tiers until all important factors, which produce the major conditions, are specified. The interfaces between areas are important in the evaluation process. To properly evaluate areas where transfers are noted, data collected from one area must be "transferred" to another and considered in the evaluation of both areas. The result is that, in a systematic way, one can assess the true impact of a particular event, and provide greater assurance that a given area is, in fact, adequate or inadequate.

Deficiencies or weaknesses were considered significant when the finding had a direct effect on the level of protection provided or was a critical element that was required for judging whether that portion of the program was acceptable. Isolated instances and minor items were not judged as representing a significant finding. However, if a number of deficiencies were found within a particular phase of the program, then a significant finding may have been warranted for that phase. In instances where the deficiency or weakness required immediate attention, the problem was discussed with licensee management and definitive corrective actions were agreed upon.

Findings

The HPAP inspections indicated that a number of weaknesses in the radiation protection programs, similar to those identified at TMI, did exist at many of the currently operating nuclear facilities. The most frequently encountered weaknesses are outlined below.

- inadequate health physics staffing levels;
- lack of adequate training for workers and health physics staff;

- lack of procedures;
- inadequate dose verification;
- inadequate calibration programs;
- failure to fully implement respiratory protection programs; and
- lack of formal ALARA programs.

Conclusions

In reviewing those programs considered to be good, three main ingredients were present. A management attitude existed that saw beyond merely meeting regulatory requirements but insisted upon and encouraged the development of a quality program with health and safety consciousness among all employees. Second, the qualifications and training of the radiation protection manager (RPM) and radiation safety officer (RSO) are of prime importance. A qualified RSO or RPM can often conduct a quality program even with a minimum of manpower, equipment, and full management support. Lastly, most licensees had implemented some portion of an ALARA program, usually not formally documented; and engineering modifications, procedural changes and housekeeping efforts had been used to provide contamination control and reduce personnel exposure.

In general the radiation protection programs at power reactors, uranium mills, and LLW disposal sites were found to be adequate. However, it was also observed that if any one of the three groupings needed more attention it was the uranium milling industry. This may be attributable to the fact that until recently it has been considered part of the chemical industry and not the nuclear industry. Therefore, in many cases health physics has taken a backseat to other aspects of the operations. More formalized additional efforts are desirable at mills. The weaknesses identified in ALARA and worker training programs indicate lack of management attention and commitment to these areas. Management attention and improvements in these programs could result in further exposure reduction, improved worker education and attitude, and a positive effect upon other mill programs. Licensee management is therefore responsible for and has a major role in the further development of radiation protection programs.

References

NUREG-0855-1981, "Health Physics Appraisal Program."

NUREG-0833-1982, "Uranium Mill Appraisal Program."

NUREG/CR-3125-1983, "Current Practices for Maintaining Occupational Exposures ALARA at Low-Level Waste Disposal Sites."

UNE ANALYSE DE L'EXPOSITION PROFESSIONNELLE DANS LES REACTEURS A EAU PRESSURISEE

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INTRODUCTION

La réduction de l'exposition professionnelle, tant individuelle que collective, est un souci permanent des exploitants des centrales nucléaires. Pour que ces efforts de protection soient entrepris d'une manière rationnelle, les exploitants doivent pouvoir s'appuyer sur une bonne connaissance des causes et des modalités de l'exposition des travailleurs pendant l'exploitation des installations. Les premiers travaux concernant l'exposition professionnelle dans les centrales nucléaires /1,2/ basés essentiellement sur les résultats concernant l'exposition collective globale, s'ils ont permis de mettre en évidence les caractéristiques principales de cette exposition n'ont cependant pas apportés de réponses très claires sur les paramètres explicatifs de la formation des doses dans les centrales. Des études plus récentes /3/, menées dans divers pays sur les possibilités de mise en oeuvre du principe ALARA et qui visaient à évaluer des actions de protection entreprises dans les centrales en vue de réduire les expositions, ont permis néanmoins de préciser les variables devant être prises en compte pour quantifier les niveaux d'exposition et d'autre part de démontrer l'intérêt de développer des systèmes de collecte des doses individuelles en relation avec les opérations effectuées pendant l'exploitation des centrales. L'objet de cette présentation est d'apporter quelques éléments de réflexions sur les déterminants de l'exposition collective dans les centrales à eau pressurisée et également de suggérer des voies de recherche nouvelles dans le domaine. Ce travail s'appuie essentiellement sur une analyse des données dosimétriques relatives aux réacteurs français en exploitation ainsi que sur quelques résultats enregistrés dans des centrales étrangères (+).

1. LES DOSES COLLECTIVES PAR OPERATION

L'identification des tâches les plus coûteuses en dose constitue l'étape préalable à la recherche et la définition des actions de protection envisageables pour réduire les expositions. La "dosimétrie d'intervention" mise en place dans les centrales françaises permet d'affecter systématiquement les doses reçues par chaque travailleur à une activité donnée. Compte-tenu de la standardisation des tranches, une typologie unique des principales opérations effectuées pendant l'exploitation a été mise au point pour l'ensemble des réacteurs. Cette typologie comprend une vingtaine d'opérations élémentaires regroupées, soit par sous-systèmes du réacteur (générateur de vapeur, circuit primaire...) soit par catégorie de tâches (inspections, chantiers...). Une première analyse de cette dosimétrie d'intervention montre que sur l'ensemble des tranches, la dose collective relative aux arrêts normaux pour rechargement du combustible représente environ 80 % de la dose totale (arrêt + exploitation). Le tableau 1 présente les résultats disponibles concernant 30 arrêts correspondant à 17 tranches réparties sur 5 sites, avec respectivement 17,9 et 6 arrêts pour le premier, deuxième et troisième cycle. On remarque la valeur relativement plus forte de la dose collective totale arrêt du premier cycle. Elle correspond au fait que pendant cet arrêt une révision générale de la centrale est effectuée, d'où un nombre important d'inspections et de travaux particuliers qui n'apparaissent plus sur les cycles suivants. On peut noter par ailleurs une légère augmentation entre le second et le troisième cycle (+ 10 %) mais seuls les résultats des cycles suivants permettront de dire s'il s'agit là d'une tendance significative. Du point de vue de la répartition de la dose arrêt selon les

(+) Cette étude a été effectuée dans le cadre d'un contrat supporté par la Commission des Communautés Européennes. Contrat CEE/CEPN BIO-444-F (S.D).

principales tâches, on note une certaine stabilité de l'importance relative de ces dernières et le rôle prépondérant joué par les travaux liés aux générateurs de vapeurs et aux chantiers (échafaudages, décontamination et nettoyage y compris BR, pose et dépose des calorifuges).

2. LES DEBITS DE DOSE

Le programme de mesures systématiques mis en place dans les centrales françaises comporte outre la collecte des doses collectives par opération, la mesure des débits de dose au contact des principaux composants du réacteur. Ces mesures sont effectuées pour un ensemble de points parfaitement définis et dans des conditions identiques d'une tranche à l'autre. La figure 1 récapitule l'évolution des "indices" de débit de dose pour les principaux composants. Ces indices représentent la moyenne arithmétique de l'ensemble des mesures effectuées au contact d'un composant donné. L'évolution des indices montre une tendance générale à l'accroissement des débits en fonction de l'âge du réacteur à l'exception de l'indice de débit pour le circuit de contrôle volumétrique et chimique (RCV). Dans le cas des générateurs de vapeur les valeurs de la figure 1 qui sont relatives à des mesures extérieures évoluent de façon analogue à celles des débits de dose moyens à l'intérieur des boîtes à eau (respectivement 44,94 et 87 mSv/h pour les 3 cycles).

Opérations	Dose collective par opération (homme-mSv)		
	1° cycle	2° cycle	3° cycle
1. Ouverture/fermeture de cuve et manutention du combustible	240 (15)	250 (18)	290 (20)
2. Générateur de vapeur	330 (21)	320 (23)	350 (24)
3. Circuit primaire	170 (11)	160 (12)	170 (11)
4. Circuit de refroidissement à l'arrêt (RRA)	120 (7)	90 (7)	80 (5)
5. Circuit de contrôle volumétrique et chimique (RCV)	70 (4)	50 (4)	60 (4)
6. Autres circuits et équipements	140 (9)	110 (8)	150 (10)
7. Inspections	120 (7)	40 (3)	80 (5)
8. Chantiers	410 (26)	340 (25)	310 (21)
Dose totale arrêt	1 600 (100 %)	1 360 (100 %)	1 490 (100 %)

Tableau 1 : Répartition de la dose collective totale arrêt

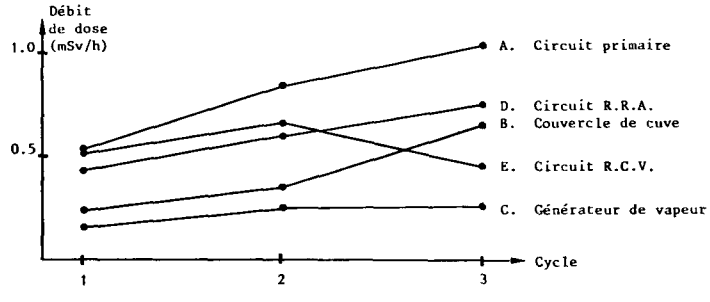

Figure 1 : Evolution des "indices" de débit de dose.

3. LA RECHERCHE DES FACTEURS EXPLICATIFS DES DOSES

La dose collective étant fonction du débit de dose ambiant et du temps d'intervention, la recherche des liaisons entre ces paramètres doit permettre de mieux préciser leur rôle et importance respectif.

3.1 - Le rôle des débits d'activité

Le tableau 2 représente la matrice de corrélation entre les doses par opérations, regroupées selon les 8 catégories principales et les débits de dose aux principaux composants. Y figurent également les corrélations entre la dose collective arrêt totale et les opérations ainsi que les corrélations entre les indices de débits de dose. En ce qui concerne des corrélations entre les débits de dose on remarque une assez bonne corrélation de l'indice relatif au circuit primaire avec les autres indices ce qui justifie que cet indice puisse être retenu pour caractériser le niveau général des débits pour un réacteur donné. Les corrélations entre la dose arrêt totale et les opérations montrent que ce sont les doses collectives associées aux opérations sur générateurs de vapeur et au circuit primaire et aux chantiers qui "expliquent" le mieux le niveau général de la dose arrêt. Quant aux relations entre les indices de débit de dose et les opérations on aurait pu s'attendre assez logiquement qu'à un débit de dose croissant devait correspondre une dose croissante, toute chose étant égale par ailleurs. Or on peut constater une mauvaise corrélation : les coefficients positifs sont généralement peu différents de zéro et la présence de nombreux coefficients négatifs tend en fait à révéler la présence d'autres facteurs déterminants.

3.2 - Le volume des travaux et le facteur humain

Dans la mesure où l'on ne dispose pas encore de données détaillées sur les facteurs déterminants les temps d'intervention du personnel, second élément fondamental de la formation des doses collectives, la révélation de l'importance de ces facteurs ne peut se faire qu'indirectement et de manière assez générale. L'analyse de l'évolution de l'exposition collective à la centrale Finlandaise de Loviisa I par exemple permet de mettre en évidence un certain nombre de tendances intéressantes (Voir figure 2). La distinction opérée ici entre dose "arrêt normal" et dose arrêt totale montre que cette dernière tend à décroître alors que l'indice de débit de dose relatif au circuit primaire est lui-même stable ou en légère croissance. On retrouve cette tendance pour la dose relative à l'ouverture et à la fermeture de la cuve ce qui tend à confirmer l'absence de corrélations déja constatées entre débits et doses. Par ailleurs on remarque que le niveau de la dose totale arrêt est fondamentalement dépendant du volume des travaux dénommés ici "exceptionnels" (défaillances du matériel, inspections particulières...) qui peuvent varier considérablement d'un cycle à un autre. Ces résultats très synthétiques tendraient à montrer que toute chose étant égale par ailleurs, en particulier le volume du travail à effectuer, le retour d'expérience ou en d'autres termes le facteur apprentissage semble jouer un rôle important dans l'évolution de l'exposition au fil des cycles.

		Doses collectives par opérations								Dose totale	Indices de débit				
		1	2	3	4	5	6	7	8		A	B	C	D	E
Dose totale		.43	.68	.68	.61	.65	.58	.59	.83	1.00					
Indices de débit	A	.21	.08	-.01	-.40	-.37	-.38	-.56	-.29	.26	1.00				
	B	.06	-.27	-.01	-.20	-.14	-.30	-.39	-.31	.32	.60	1.00			
	C	.33	-.21	-.27	-.55	-.36	-.31	-.62	-.26	.38	.72	.60	1.00		
	D	.14	.10	.19	-.11	-.21	.04	-.27	-.09	.09	.48	.35	.45	1.00	
	E	.66	.29	.20	.01	.18	.07	-.09	.5	.03	.15	-.07	.20	.13	1.00

Tableau 2 : Matrice des corrélations

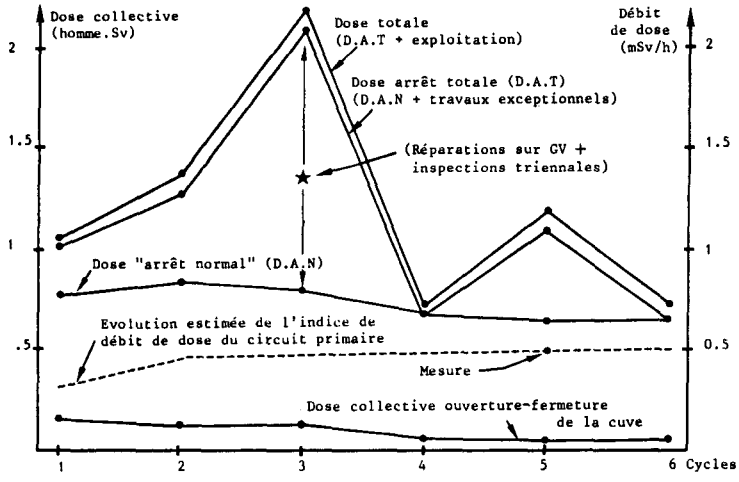

Figure 2 : Evolution des doses collectives de Loviisa I

CONCLUSION

L'étude présentée ci-dessus apporte quelques éclairages sur les modalités de l'exposition professionnelle dans les réacteurs à eau pressurisée. Il convient de noter que les extrapolations et les comparaisons avec des résultats d'autres centrales doivent cependant être maniés avec prudence dans la mesure où le recueil des données dosimétriques (découpage des opérations, procédures de mesure des débits de dose entre autre) sont loin d'être harmonisés.

Les quelques développements concernant la recherche des facteurs explicatifs des doses tendent cependant à montrer l'intérêt qu'il y aurait sûrement à étudier plus en détail le rôle des éléments constitutifs du volume de travail ainsi que ceux que l'on regroupe généralement sous le terme de facteurs humains à savoir entre autre : l'organisation du travail, les procédures d'intervention, la qualification et l'entraînement du personnel, les politiques de protection. Evidemment de telles études supposent qu'au préalable aient été mis au point une série d'indicateurs et qu'une collecte systématique des données adéquates se mette en place dans les centrales.

REFERENCES

/1/ S.F.R.P, Journées d'études sur les dispositions prises à la conception et en exploitation pour réduire la radio-exposition professionnelle dans les centrales françaises nucléaires à eau ordinaire, Paris 12-14 décembre 1979, SFRP, Fontenay-aux-Roses, France 1980.

/2/ I.A.E.A, Occupational Radiation Exposure in Nuclear Fuel Cycle Facilities, Proceedings of a Symposium, Los Angeles, 18-22 june 1979, IAEA, Vienna, 1980.

/3/ C.E.E, "As low as Reasonably Achievable", Second European Scientific Seminar on Radiation Protection Optimization", 8-9 Novembre 1983, Luxembourg (à paraître).

EFFORTS AND ACHIEVEMENTS IN DOSE REDUCTION AT CERN

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1. INTRODUCTION

The European Laboratory for Particle Physics, CERN, presently operates several high-energy proton accelerators, the largest having a maximum energy of 450 GeV. In addition, a large electron positron collider (LEP) is under construction.

There are two principal types of radiation fields encountered at CERN: the stray radiation outside the shielded beam enclosures when the accelerators are in operation, and the field due to radioactivity induced in the accelerator structures, which is of particular significance during periods of accelerator maintenance and repair. In spite of regular increases in the number of protons accelerated in the machines, it has been possible to maintain stray radiation levels in experimental halls, on the CERN site in general and in the environment around CERN at relatively low levels by constantly improving the efficacy of shielding configurations. To give an example: the 26 GeV proton synchrotron (PS) was designed in the early 1960's to operate with an intensity of 10^{10} protons per pulse with a pulse period of 2.4 seconds; now it routinely accelerates more than 10^{13} protons per pulse while the pulse repetition rate has doubled. The increase in annual doses at the CERN perimeter fences due to accelerator operation has hardly been noticeable.

Such an increase in intensity, however, has major consequences for the levels of induced radioactivity around the accelerators themselves and in target regions: it therefore has a direct impact on doses received by personnel during maintenace and repair. Both the collective dose and the personnel doses received during work in the vicinity of activated accelerator structures are about one order of magnitude higher than those due to stray radiation. It was therefore natural to concentrate the major dose-reduction efforts on this sector.

2. DOSE REDUCTION BY CORRECT DESIGN

The CERN Radiation Safety Manual (1971 edition) clearly stated that "as a general principle unnecessary exposure of any person to radiation must be avoided" and that "all precautions must be taken to reduce individual exposure to a minimum, i.e. restricted to the smallest number of people and to the lowest possible dose". In addition, "occupationally exposed persons shall not receive a dose to the whole body exceeding 5 rem/year".

The spirit of these three statements is already rather close to the basic principles of ICRP 26: justification, optimization, and limitation (Ref. 1). Despite the similarity in ideals, when the ICRP recommendations were published in 1976 most staff and CERN management still maintained an attitude based on pre-1960 philosophy: as long as a person stayed below the limit of a maximum permissible dose, there was nothing to worry about.

Thus early measures to reduce personnel doses were mostly efforts to ensure the smooth running of the accelerator complexes without exceeding any maximum permissible exposure limit. Just to mention three examples: the introduction of more reliable vacuum joints required fewer interventions by the vacuum group around the machines, thereby reducing accelerator down-time, and only incidentally radiation exposure. Rubber hoses for cooling water that tended to break in high-radiation areas were replaced by metal tubing mainly to reduce down-time but also requiring less repair work in hot areas. However, preoccupation with the high

dose rates in target stations allowed the development of quick-change mechanisms for the remote replacement of burned-out targets in extracted beams. The overall reduction in personnel dose by these and other similar measures was a beneficial side effect but, even so, certain machine specialists, mostly the same people, received high doses year after year.

With the construction of the 450 GeV SPS accelerator, serious efforts were made in the design phase to avoid or reduce personnel doses. These included the introduction of remote handling in target regions and of quick-connections of control and electricity cables and water hoses for beam elements likely to become highly radioactive. A system of beam loss monitors was introduced in order to optimize the operation of the machine.

Since radiation damage and induced radioactivity are closely related, improvements in older accelerators aimed at preventing proton losses around the machines and ejecting unwanted beam, preferably onto external, well shielded dumps. Both in the construction of the new accelerator and in improving the existing ones the Radiation Protection Group collaborated actively with the Machine Divisions with a view to applying the ALARA principle in the early design phases.

A rather unusual situation also existed within the Radiation Protection Group itself. Survey technicians were found to belong to the category of heavily irradiated personnel, and so efforts were necessary to reduce their doses. For example, in the past, maps of isodose curves were established in target regions or around hot splitter and septum magnets. As these surveys cost a great deal in dose received, the benefit of such an activity was seriously questioned. However, in order to continue to have information on dose levels in the vicinity of hot points where the probability of human intervention is high, very simple radiation monitor systems having the possibility of remote read-out were installed in these zones. It was however important to make sure that the installation (or later the repair) of such a system in a hot area did not cost more dose than saved by abolishing extended radiation surveys.

2. DOSE REDUCTION BY SENSITIZATION (OR THE IMPACT OF ICRP 26)

One of the basic changes introduced in Publication 26 of the ICRP compared with previous recommendations was the recognition that any exposure carries with it some detriment or risk. It was henceforth not "permissible" to irradiate persons at or near the dose-equivalent limits. This point was particularly emphasized by the Staff Association of CERN. The human attitude in the perception of risk here plays a decisive role. It did not impress people to see that the (calculated) detriment from professional radiation doses is much lower than from many commonly encountered hazards in daily life, or that other occupational hazards may be even less well understood than radiation. While more common risks are accepted (because accidents only happen to others), the argument of a stochastically unlikely detriment from a radiation exposure was too abstract to be convincing.

It was possible to identify within the highly exposed staff at CERN small groups who are regularly exposed at a higher level than the rest. These groups consisted primarily of accelerator maintenance staff (and the radiation protection technicians). The support of the CERN management was clearly needed (Ref. 2) to reduce their doses to a level lower than that achieved by the application of existing ALARA principles. This support was given in the form of a strong guideline introducing a reference dose of 15 mSv/year within the framework of a general CERN radiation policy (Ref. 3). The dose-equivalent limit of 50 mSv/year remains in line with the legislation of CERN's two host countries France and Switzerland, but it now needs the authorization of a division leader for a person to be exposed beyond the reference dose of 15 mSv/year. The value of 15 mSv was chosen rather arbitrarily but it was felt that most work in radioactive areas could well be

completed within such a dose. It was also expected that any authorization should be made the subject of a management review which should propose alternative procedures and/or expenditure of money to ensure that a similar authorization would not be necessary in subsequent years.

In order to achieve this, schemes involving closer dose control were set up for those groups in which the personnel doses of a majority of the workers were above one tenth of the dose limit set forward by the ICRP. For example, the activities of these persons were closely watched by radiation protection technicians. They also appeared on a special "dose limit" chart issued monthly by the film-badge service. Such a list is easier to read than the list containing the complete results of over 4500 film-badge wearers.

Special efforts were made to eliminate doses received unnecessarily while simply waiting in hot areas. The introduction of a small pocket monitor, giving an audible beep in proportion to the dose rate, was particularly successful in this respect. The result of these efforts is seen in Tables 1 and 2, which show that no member of either the CERN personnel or of outside firms working on the domain of the Organization has exceeded the reference dose in 1981 or 1982. The increased dose-consciousness among the staff supported by the strong policy statement of management has led to a decrease of percentages in all higher dose classes (Tables 1 and 2). The reason that doses for outside contractors show a less favourable tendency than CERN staff is that these persons are often hired specifically for special work during accelerator shut-down. It should be stressed, however, that personnel from outside contractors benefit from the same radiation policy as the CERN staff in that their exposure is subject to the reference dose of 15 mSv/year and furthermore limited to 1 mSv/week for the duration of their stay at CERN.

It is often stated that when efforts are undertaken to distribute the high personal doses of a few workers among more people, the collective dose for a given operation will increase. Such a trend is not seen at CERN. The development of the collective dose of CERN over the years 1979 to 1982 is presented in Table 3. It should be noted that the decrease by a factor of two during the last four years cannot entirely be attributed to the success of the ALARA programme. The profile of CERN activities in high-energy physics has changed: some old, dose-costly installations have been closed down and new ones brought into operation which were constructed according to the ALARA principle.

4. CONCLUSIONS

Constant efforts over the years by the Radiation Protection Group at CERN, in collaboration with the accelerator divisions, have led to an overall decrease in doses to personnel. Annual exposures in the most exposed groups are now below a level of 15 mSv/year, the figure used by the CERN management as a guideline for present and future research activities. Support at top management level has been beneficial in keeping an ALARA spirit alive throughout CERN, from the accelerator divisions as staff units down to the individually exposed person principally concerned.

REFERENCES

1. Recommendation of the International Commission on Radiological Protection, Publication 26, Pergamon Press, Oxford (1977).

2. A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA), U.S. Department of Energy, DOE/EV/1930-TS, UC-41 (1980).

3. CERN Radiation Protection Policy, Weekly Bulletin No. 29/81, July 1981.

Table 1

Percentage of CERN staff in different annual dose categories

Year	⩽ 2 mSv	2 mSv ⩽ 5 mSv	5 mSv ⩽ 15 mSv	> 15 mSv
1979	92.5	5.1	2.4	0.1
1980	93.9	3.9	2.2	0.1
1981	94.6	3.6	1.8	0
1982	95.7	2.9	1.4	0

Table 2

Percentage of contractors staff in different annual dose categories

Year	⩽ 2 mSv	2 mSv ⩽ 5 mSv	5 mSv ⩽ 15 mSv	> 15 mSv
1979	87.1	9.2	3.5	0.2
1980	88.8	8.5	2.7	0.1
1981	93.0	4.7	2.3	0
1982	90.6	7.0	2.4	0

Table 3

Collective dose in Sv of CERN staff and outside contractors for the years 1979-1982

Year	CERN staff		Outside contractors	Total
	Gamma dose	Neutron dose		
1979	3.12	0.22	1.25	4.59
1980	1.96	0.26	1.30	3.52
1981	1.36	0.26	1.16	2.78
1982	1.13	0.17	0.68	1.98

PROTECTION OF INDUSTRIAL RADIOGRAPHERS THROUGH FACILITIES DESIGN

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INTRODUCTION

Because Industrial Radiographers are involved in more radiation accidents, receive more overexposures, and are cited for more serious items of non-compliance by governmental agencies than any other group in the nuclear field, their protection facilities must be designed foolproof and fail-safe.

The Boeing Company has long strived to meet this requirement by full radiation protection built into every industrial radiographic facility. This includes both enclosed shielded rooms and open field radiography. Facilities and equipment are designed to contain the hazard, with anything from a four foot concrete shielded room for an industrial radiographic accelerator to a piece of rope that isolates an Iridium-192 source during a field radiographic inspection on an aircraft.

X-ray units, accelerators, and radioactive materials are used in our radiographic inspection program on aircraft, missiles, and seacraft. We presently employ seventy-five Radiographers, operate ninety radiation sources, and take ten-thousand radiographic exposures per month.

ANALYSIS OF RADIATION HAZARD

When a Quality Control Group within the Company wants to use radiation to perform a quality control test, they determine the type and amount of radiation required to do the job. Once the source of radiation is determined, the Quality Control Group and Radiation Health Protection must agree upon where and how the source will be used. We must decide if the radiography can be done in a completely shielded room, a partly shielded area, or an open locality.

Protection facilities are designed to solve specific radiation problems. These problems are defined by analyzing the hazards to determine their impact upon the working and public environments. The hazard analysis considers the type and size of the radiation source to be used, the expected exposure levels, any interaction between employees and the sources, and worst possible consequences of employee error or equipment failure.

Industrial radiographic operations using the radiation source are reviewed to accurately establish: (1) The amount of time the Radiographer and other employees are exposed to radiation; (2) The intensity of the exposures; and (3) The areas of their bodies that are exposed. The possible exposure dose will help scope the facility and equipment design.

FACILITY AND EQUIPMENT DESIGN

When the analysis is complete and the degree of hazard is known for the operation, the necessary facilities and equipment can be designed to contain the hazard. At Boeing, we try to design out all the hazards associated with industrial radiographic inspections.

Our industrial radiographic operations are classified in three general categories: (1) Shielded Room operations; (2) Portable Self-contained

Protection Systems for use outside of a shielded room; and (3) Open Field Radiographic Inspections.

SHIELDED ROOMS: In a Shielded Room design, we place enough shielding in the walls, floor, and ceiling to keep the radiation exposure outside the shielded area to less than 10 mRem per week. We shield the room for the maximum kV and mA to be used, Curie amount to be used, the maximum work load, and how often the areas outside the room are occupied per week. We reduce the cost of shielding by limiting the number of Primary Beam walls. Shielded Rooms may be made of Lead so they can be moved if necessary. Steel supports are used to hold the Lead (1.3 x 1.3 meter) sheets in place. The sheet is bonded to 19 mm Plywood, and tack-welded with lead strip overlay. This type of room can be taken apart and moved to another location. In some cases, Concrete is chosen as the shielding medium. Large, keyed, solid concrete blocks are sometimes used in the design of high energy shielded rooms.

All entrances to the radiation area within the room are provided with Interlocks that shut down the radiation production when opened. The Interlocks are all of a fail-safe nature, and must be reactivated by both closing the door and resetting manually at the door. The interior of the exposure room is posted with a sign, "DO NOT OCCUPY AREA WHEN DOOR IS CLOSED". A Flashing Warning Light is placed outside of each entrance to the room, and also inside the room. The Warning Light on the outside illuminates a sign which reads, "RADIATION ON". This sign is actuated by the radiation. The Warning Light inside (preferably a Rotating Beacon) is activated twenty seconds prior to the production of radiation, and remains activated during irradiation.

An Audible Warning Device is also used inside each shielded area. This audible signal is activated twenty seconds prior to the production of radiation and remains on for thirty seconds after the exposure begins.

Signs warning "CAUTION——ENTRANCE TO A HIGH RADIATION AREA" are placed at the entrance to each shielded room to warn personnel of the hazard within the room. There is also a "HIGH RADIATION AREA" Sign within the room.

Emergency Power Cutoff Switches (Scram Buttons) are placed in each exposure room, and are easily identifiable because of the large red square painted on the wall behind it. A Sign is located above the switch which reads "IF ALARM SOUNDS, PUSH SWITCH AND EVACUATE ROOM IMMEDIATELY".

Also placed in each room is a Continuous Radiation Monitoring Device which gives off an audible signal when radiation is present. These are placed in the room as a secondary protection system to ensure that the Radiographer knows if the X-ray beam is on, or a radioactive source is exposed within the enclosure. A Portable Instrument is also carried in the room to ensure that a radiation source is shielded.

IN-PLACE SHIELDING: In-place Shielding Devices are designed for special field applications. This may be a small lead enclosure used in radiographing small parts, or a lead shielding fixture which allows the radiographing of an in-place hydraulic tube on an aircraft. The radiation exposure levels on the exterior of these devices do not exceed 2 mRem per hour at 30 cm from the shields. Where possible, the shields are fitted with safety equipment such as Interlocks and Flashing Lights. Signs, instructing Non-Occupationally Exposed Individuals to stay at least one meter from these devices, are posted in the area. The Interlocks ensure that the in-place shielding device is closed or fits tightly to the working surface, to eliminate leakage. A Flashing Light is activated when the radiation source is turned on or exposed. Both X-ray

units and radioactive material sources are utilized in the in-place shielding devices. By use of this type of equipment, radiographs can be taken on an aircraft in the assembly line without having to remove any of the nearby workers (a great cost savings for the company).

FIELD RADIOGRAPHIC INSPECTIONS: For Field Radiographic Inspections, posted barriers such as ropes, fences, and barricades are used to designate the hazard area. Distance from the source of radiation is used as our primary protection method. However, portable lead shields may also be used during field radiographies. Shields generally are 1.2 meter by 2.4 meter sheets of 9.5 millimeter or 12.7 millimeter Lead. Either may be used for Primary and Secondary radiation shields, depending upon the energy of the source. Use of shields cuts down the size of the hazard area and allows more employees to continue working near the radiographic operation.

In addition to the portable shields, we also use Lead Cones on the Gamma and X-ray beams to limit the field size. The Cones are fitted to the X-ray tube or on the exposure tube of the radioactive material device. Both Audible and Visible Warning Alarms are used at the radiographic site. Flashing Warning Lights are activated twenty seconds before radiation is present and all the time during the exposures. The Audible Alarm is activated by the radiation source. The barriers, signs, and warning devices are put in place by a Radiation Monitor from the Radiation Health Protection organization. Often, the Radiographer aids in this process; the two work as a team. During the actual radiographic exposure, both members of the team patrol the hazard area to ensure that no one enters.

FACILITY CERTIFICATION

The Facility and Equipment Certification Program plays an important part in the Boeing success of reducing its employee radiation exposures. Each radiation facility is certified to operate only under specific conditions. These certifications are written by the Radiation Health Protection organization, and the contents approved by the Operating Groups.

The Certification Form contains the conditions under which an X-ray, Gamma source, or Accelerator may operate in the facility. It specifies the maximum operating power range in kiloVolts (kV) and milliAmps (mA), or Curies for radioactive material, and the maximum time the equipment may be operated in a given week. It tells the Operator at which walls the radiation source can be pointed, what power levels can be used, and the minimum distance the source must be from the wall whenever it is activated. These conditions are specified for the four walls, the ceiling, the floor, and any door within the facility. All operations are re-certified annually.

EVALUATION OF THE FACILITY DESIGN

The effectiveness of a facility design must be evaluated during the initial start-up of the operation. The start-up survey is quite detailed in order to assure adequate containment of the hazards. The shield is verified, interlocks/lights/panic button checked, and protection systems reviewed in case of abnormal operations.

Long-term evaluation is through personnel dosimetry. Personnel dosimetry is worn by all personnel working in radiation areas. These include Film Badges, Pocket Dosimeters, and Thermoluminescent Dosimeters (TLD's) for special monitoring programs. The Radiographers are required to wear the Film Badge and Pocket Dosimeter at all times. TLD's are used in special programs to

determine if certain areas of the body are receiving more radiation than others.

Personnel working under field conditions are issued Personnel Audible Warning Devices which alarm when in a radiation field. Of course, each field radiography is monitored by a Representative of the Radiation Health Protection organization. Periodic (quarterly) Radiation Surveys are also conducted on radiation facilities. These surveys include monitoring of the shielding, safety devices, and radiation source. Reports of all surveys are maintained in our records for review by governmental agencies.

The results of the Boeing program have been excellent. Only one individual in twenty years of operation was exposed to one Rem in a year. The average exposures for one year are below 100 mRem.

The potential for high acute or even low chronic exposure to radiation has essentially been eliminated at The Boeing Company by designing out the radiation hazards.

REDUCING RADIATION EXPOSURE IN THE OPERATING ROOM

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Advancements in radiologic imaging have led to the increased use of xrays in departments outside of radiology. A large number of cholecystectomies (gallbladders), orthopedic, endoscopic and pacemaker implants are now performed each year with the aid of fluoroscopic and/or radiographic systems. If the surgeon is unfamiliar with radiation safety precautions or does not require that proper procedures be adhered to, unnecessary exposure can result.

Very often it is impossible for the surgeon to move away from the tableside during exposure. The investigation began therefore with measuring exposure levels at the side of the operating room table with each of the four types of xray equipment available on the surgical floors. Table I indicates the radiation exposure at various distances from the xray tube. The exposure rates at the table side from the fluoroscopy equipment are such that anyone remaining in that location should be provided with a protective lead apron and a radiation monitor.

TABLE I TABLE SIDE EXPOSURE

Type	Technique	Output	Distance (m) 0.5	1.0	2.0
Portable Radiographic	80 kVp	500 mR	4.0 mR	1.0 mR	0.3 mR
Fixed Radiographic	80 kVp	100 mR	0.8 mR	0.2 mR	0.6 mR
C-arm Fluoroscopy	80 kVp	3 R/min	5 mR/min	2 mR/min	0.2 mR/min
Fixed Fluoroscopy	80 kVp	5 R/min	8 mR/min	3 mR/min	0.1 mR/min

During the course of the surgery the staff often moves about the OR suite, making it difficult to estimate their actual exposures using the information presented in Table I. To obtain actual exposure measurements, thermoluminescent dosimeters were attached to the staff during various procedures. Table II shows a summary of the range of exposures from various surgical procedures. The exposure values for both the surgeon and the anesthesiologist indicate that both types of physicians must be required to wear lead aprons as well as film badges. All other persons in the OR during these studies were able to either move at least two meters from the xray tube or out of the OR and therefore received non detectable doses.

TABLE II EXPOSURE TO PERSONNEL DURING SURGERY (mR)

Type	Surgeon	Anesthesiologist	Xray Tech	Other Staff
Cholangiogram	3-49	0-17	0-0	0
Orthopedic	10-31	1-11	0-4	0
Pacemaker	15-45	1-14	0-6	0

Discussion

All the factors which contribute to the exposure to the patient and staff must be considered before radiation exposure guidelines

can be developed. The type of equipment used for the procedure certainly is one of the most important factors. Fixed radiographic machines are usually equipped with phototimers that eliminate guesswork and retakes while keeping the dose to a minimum. Fixed fluoroscopy units, which may have a higher output than portable C-arms usually have shielding to protect the operator and therefore have lower scatter rates. Portable C-arms, however, are the type of fluoroscopy equipment most often used during surgery. Surgeons often are not familiar enough with the device to know which end is the image intensifier and which is the xray tube. Consequently, the equipment is often used in the wrong orientation. If the patient is positioned close to the image intensifier, the exposure is similar to stationary units. If the xray tube is placed next to the patient, however, the skin exposure will be much higher. Outputs as high as 40 R/min were measured at one institution. Gough (G070) has reported skin doses as high as 283 rads from pacemaker implants. The surgeon should be encouraged to place the tube under the table and as far away from the patient as possible.

Image storage devices are now available to use with portable C-arms. The device allows the surgeon to have still images on the video display. This is useful in procedures such as orthopedic surgery where the surgeon needs to look at the location of the pin and does not need continuous fluoroscopy. The image can be obtained in a fraction of a second reducing the fluoroscopy time from minutes to seconds and the skin exposure from thousands of mR to less than one mR. This device is also being used in emergency rooms for quick surveys of accident victims.

Other factors affecting the exposure to both the patient and staff include the patient size. Large or obese patients can increase the scatter by as much as 30%. This is due to a combination of the facts that larger patients require higher outputs and fat tends to scatter radiation farther than muscle tissue. The field size also influences exposure. The state of California requires that the field size be reduced to only the area of clinical interest. Surgeons often need to be reminded to close the collimator. Increasing distance is the easiest method to reduce the exposure to the staff. Personnel in nonsterile attire can leave the OR suite during exposures and eliminate all exposure. Other personnel must be provided with shielding. Portable lead shields can be used for staff who are able to leave the tableside. Lead aprons, however, must be used by surgeon and anesthesiologist. Special break away aprons have been designed to enable the surgeon to remove the apron without breaking sterility.

Finally, the surgeon must be encouraged to become acquainted with interrupted use of the fluoroscopic foot pedals. In order to bring this concept to the attention of the surgeons, a log was kept of the amount of fluoroscopy time used during each procedure. Shortly after beginning this program, the amount of fluoroscopy time for cholangiograms decreased from five minutes to two minutes, thereby reducing the patient exposure by 3 to 15 R. Personnel exposure dropped proportionately.

Table I indicates that it is possible for the surgeon to be exposed to 50 mR during a single procedure. Some surgeons may perform as many as five to ten procedures in a weeks time. It is possible

therefore for a busy surgeon to receive more exposure than the radiologists. In fact, the film badge exposures for surgeons who routinely wear a badge average between 100 to 200 mR per month. The film badges should be stored in an area the surgeons must pass on their way to surgery to encourage use.

Education is the most important aspect of any radiation safety program. The State of California no longer includes the use of xrays as part of the Medical Licence. Each physician must take an exam in order to receive a licence to use xrays. The two hour exam focuses on radiation safety and proper use of equipment. The radiation safety officer of this institution provides a review course to assist the physician in preparing for his exam. For most surgeons, this is the first time they have been exposed to radiation protection concepts as well as the availability of a radiation protection professional with whom they can discuss questions they and their patients have regarding radiation exposure.

In summary, the following list contains the major points that should be included in a radiation reduction program for the surgical staff.

1. Educate the surgeon to the correct orientation of equipment, proper collimation, availability of video storage devices, and especially, interrupted use of the fluoroscopy pedal.
2. Personnel in nonsterile attire should leave the room during exposures.
3. Personnel in sterile attire should stand behind a portable lead shield or move as far away from the xray tube as possible (2m).
4. The surgeon and anesthesiologist must wear lead aprons.
5. Film badges must be provided for the surgeon and the anesthesiologist and stored in a convenient location.
6. An educational program must be developed to address and alleviate the fear of radiation and to educate the staff on the simple methods that can be employed to keep the radiation exposure to the patient and themselves to a minimum.

REFERENCES

(1) California Radiation Control Regulations Title 17 California Administration Code Chapter 5, State of California Department of Health Services, 1980.

(2) (G068) Gough JH, Davis R, Stacey A.J. 1970 Br. J. Radial. 41,508

(3) (EA 80) Earley, D. Radiation Hazards in Operative Biliary Radiology edited by G. Berci pg 27-36 Williams & Wilkens, New York 1980

RADIATION SHIELDING FOR MEDICAL LINACS PRODUCING X RAYS OF 6 AND 15 MEV: COMPARISON OF CALCULATIONS WITH MEASUREMENTS

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Calculational models published by the U.S. National Council on Radiation Protection and Measurements (8,9) have been used to determine shielding design requirements. Calculation of x ray dose transmitted through a barrier makes use of measurements of "broad beam" transmission of x rays through various materials while transmitted neutron dose equivalent is based on calculations. For entry mazes, the model depends on empirically derived scattering coefficients.

We evaluated the accuracy of radiation shielding calculations for two medical linear accelerator facilities operating at 6 and 15 MV. We found a need for x ray transmission data for high density concrete, as well as a need to consider the primary and scattered photon energy spectra in dose estimations for scattered radiation.

Shielding design for the two facilities (Figs. 1 and 2) included consideration of unwanted neutrons produced by the 15 MV LINAC. Calculations were made to predict dose equivalent rates of x rays and neutrons at various points in the entry mazes and outside the shielding walls. Transmission of data for normal concrete shielding (8,9) was supplemented by transmission data for high density concrete (1,4,5). Radiation dose equivalent rates from both x rays and neutrons scattered in the entry mazes were estimated using published scattering coefficients (9).

The source terms for x ray dose rates were based on measured primary beam and leakage radiation, while neutron source data were taken from the literature (10). The primary x ray dose rate at 1 m from the target was:

For the 6 MV LINAC:

$$D = 2.25 \text{ Gy/min}$$

For the 15 MV LINAC:

$$D = 3.00 \text{ Gy/min}.$$

The source term for leakage x rays was based on measurements and was in the range of 0.05 to 0.1% of the primary beam dose rate at 1 m from the target. Transmission of leakage radiation was calculated in the same way as for the primary radiation which is likely to be an overestimate as Tochilin points out (11). The x ray transmission through walls as constructed with high density concrete (3.85 g/cm^3) was determined in two ways: 1) the wall thickness was converted to equivalent thickness of normal concrete (2.35 g/cm^3) using a ratio of the densities; 2) based on data presented by Maruyama (5) and Lokan (4).

The neutron fluence rate at 1 m from the 15 MV LINAC target was 4.4×10^5 $/cm^2 s$. The average energy of neutrons emerging from medical LINACs of this type is estimated to be 0.3 MeV (7). We assumed that high density concrete was as effective as normal concrete in attenuating neutrons.

X ray dose rate measurements were performed using an ionization chamber survey meter with calibration traceable to the U.S. National Bureau of Standards. Neutron dose equivalent rates were measured using a remmeter of the Andersson-Braun (2) design which had been calibrated against a PuBe source of known intensity. In the areas surveyed, neither instrument was affected by the pulsed nature of the radiation field. Ionization chamber readings were corrected for the neutron response of the instrument.

Tables 1 and 2 present data from both calculations and measurements of the dose equivalent rates of x rays and neutrons at the points designated in Figs. 1 and 2. The transmission results (Table 1) indicate for x rays that better agreement is achieved if transmission data for high density concrete are used. Neutron dose equivalent rates outside the shielding walls were extremely low. Data are not presented, but calculations predicted higher dose equivalent rates than were measured.

TABLE 1
Comparison of Calculated and Measured Dose Equivalent Rates from Primary Photon Radiation Transmitted through Shielding Walls at Medical Linear Accelerator Facilities Operating at 6 and 15 MV.

Meas. Pt. **	Beam Direct	Calculated Dose Equiv. Rate (mSv/h) Density Scaling	Refs. 4 and 5	Meas. Dose Equiv. Rate (mSv/h)
6 MV Photons				
D	North	0.037	0.009	0.008
E	North	0.039	0.011	0.015
F	South	0.026	0.007	0.008
G	South	0.015	0.004	0.006
15 MV Photons				
D	North	0.057	0.025	0.017
E	Up	0.0002	0.00008	<0.001
F	Up	0.0011	0.0007	<0.001

** measurement points refer to Figs. 1 and 2

Calculated dose equivalent rates from scattered radiation in the entry mazes were dominated by single scattered leakage and primary, and double scattered primary radiation. For the 6 MV room dose equivalent rates in the maze were overestimated, but were within a factor of two of the measured values (Table 2). For both x rays and neutrons in the entry maze of the 15 MV room, the dose equivalent rate was sometimes underestimated, but in general was within a factor of two of measured values (Table 2).

Uncertainty in the choice of scattering angle and photon energy when selecting scattering coefficients is one source of error in the calculations. The dose equivalent rate for x rays can be greatly underestimated if an inappropriately high energy is used since the scattering coefficients decrease rapidly with increasing energy. For the calculations presented in Table 2, we assumed an average energy of 2 MeV for the 6 MV beam. We tested this assumption by integrating the scattering over a published 6 MV x ray spectrum (3) and found that the calculated dose rate following the first scatter was within 5% of that calculated assuming an average energy of 2 MeV. When a second scatter was considered, we assumed

an average energy of 0.25 MeV for once-scattered photons (6). Similarly, for the 15 MV beam, we assumed an energy of 5 MeV for primary photons and an energy of 0.3 MeV for the scattered photons (6). Also, there is an uncertainty of about a factor of two in the neutron source term for the 15 MV machine.

TABLE 2
Comparison of Calculated and Measured Dose Rates from Transmitted and Scattered Photon and Neutron Radiation in the Entry Maze of 6 and 15 MV Medical Linear Accelerator Facilities

		15 MV		6 MV	
Meas. Pt. **	Beam Direct	Calc. Dose Eq. Rate (mSv/h)	Meas. Dose Eq. Rate (mSv/h)	Calc. Dose Eq. Rate (mSv/h)	Meas. Dose Eq. Rate (mSv/h)
Photons					
A	North	0.42	0.12-0.85	0.48	0.35
A	South	0.08	0.10	4.32	3.50
A	Down	0.09	0.08	0.62	0.60
B	North	0.04	0.05	-	-
B	South	0.04	0.03	0.52	0.25
C	South	0.02	0.02	0.07	0.06
Neutrons					
A	North	0.18	0.40	-	-
A	South	0.30	0.60	-	-
B	North	0.03	0.06	-	-
C	North	0.01	0.01	-	-

** measurement points refer to Figs. 1 and 2

Uncertainties in the x ray dose rate measurements were less than 10%. Uncertainties in the measurements of neutron dose equivalent rates were probably of the order of a factor of two because of the energy dependence of the remmeter.

In summary, calculations used to predict transmitted x ray dose rates at medical LINAC facilities shielded with high density concrete generally agree with measurements within 50% when published transmission data for the appropriate high density concrete are used. Transmitted neutron dose equivalent rates seemed to be overestimated. Data from Adams and Lokan (1) indicate that high density concrete may be more than 10 times as effective as normal concrete for neutron attenuation, and this may account for the observation. The dose equivalent rates from scattered x rays and neutrons were, in general, predicted within a factor of two for both machines. Differences could result because of the assumptions made regarding the radiation energy, the scattering geometry and the source terms, as well as inaccuracies in the measurements.

REFERENCES

1. Adams R.J. and Lokan K.H., 1979, Health Physics 36, 671-678.
2. Anderson I.O and Braun J., 1963, in Neutron Dosimetry, Vol. II, 87, International Atomic Energy Agency, Vienna.
3. Levy L.B., 1974 Doctoral Thesis, University of Texas, San Antonio, Texas.
4. Lokan K.H., Sherman N.K., Gellie W.H., Henry W.H., Levesque R., Nowack A., Teather G.G. and Lundquist J.R., 1972, Health Physics 23, 193-199.
5. Maruyama T., Kumamoto Y., Kato Y., Hashizume T. and Yamamoto M., 1971, Health Physics 20, 277-284.
6. Maruyama T., Sakata S., Kumamoto T., Hashizume T., Hattori H., Kanamori H. and Yamamoto M., 1975, Health Physics 28, 777-791.
7. McCall R.C. and Swanson W.P., 1979 in Proceedings of a Conference on Neutrons from Electron Medical Accelerators, 75-86, NBS Special Publication 554, U.S. Government Printing Office, Washington, D.C.
8. National Council on Radiation Protection and Measurements, 1976 NCRP Report 49, Washington, D.C.
9. National Council on Radiation Protection and Measurements, 1977 NCRP Report 51, Washington, D.C.
10. Swanson W.P., 1980, Medical Physics 7, 141-144.
11. Tochilin E. and LaRiviere P.D., 1979, Health Physics 36, 387-392.

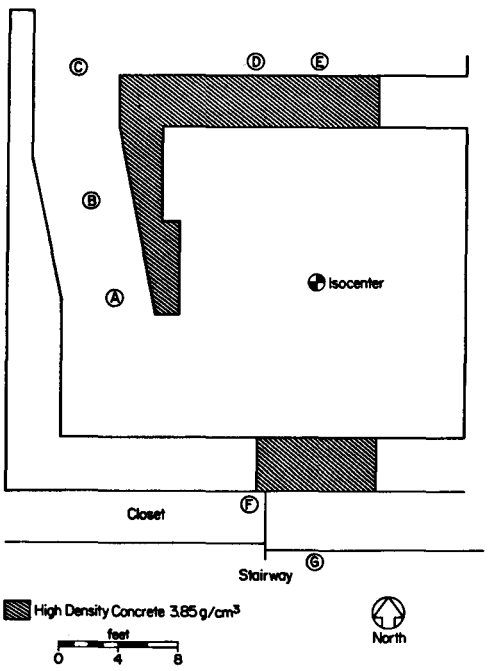

Fig. 1. Floor Plan of 6 MV Accelerator Room

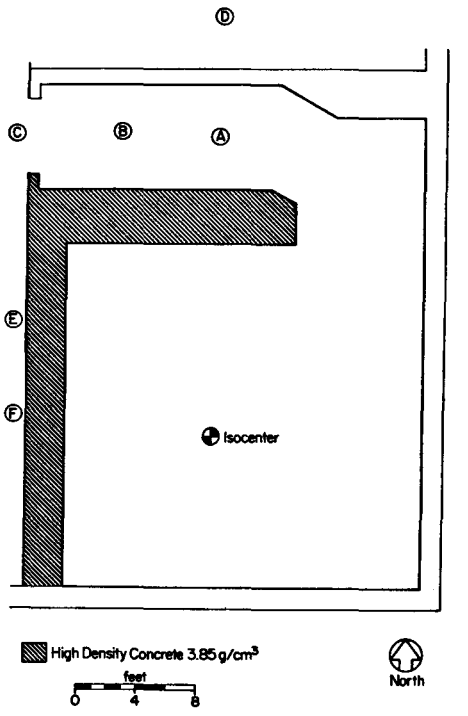

Fig. 2. Floor Plan of 15 MV Accelerator Room

AIRBORNE RADIOACTIVITY PRODUCED AT ELECTRON ACCELERATORS COMPARISON BETWEEN MEASUREMENT AND THEORY

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1.) Introduction

In the atmosphere around low energy (< 100 MeV) electron accelerators two radionuclides of concern are ^{13}N and ^{15}O which are produced by (γ,n) reactions from ^{14}N and ^{16}O. The half lives of these positron-emitters are 10 min and 2,03 min respectively. In order to obtain reliable data for estimating the radiation exposure caused by gaseous radionuclides within the accelerator installation and in the environment, the concentrations of the radioactive gases were measured on the axis of the primary X-ray beam and at various points in the accelerator room. These measurements were performed at a 35 MeV electron linear accelerator and a 45 MeV betatron.

As a detector for gaseous ß-emitters a conventional device consisting of a 70 l steel barrel with two large GM tubes inside (one of them shielded against ß-radiation) and a differential counter for background compensation was used. The radioactive gas produced in a small irradiation chamber on the axis of the X-ray beam was pumped into the measuring barrel located in a separate well shielded room. The concentration in the accellerator room outside the collimated X-ray beam was determined by directly (without an irradiation chamber) pumping the activated air into the measuring device. The contributions of ^{13}N and ^{15}O to the total gas activity were evaluated by decomposition of decay curves. The experimental values of the activity concentrations on the X-ray beam axis were compared with the effective cross sections of the reactions $^{14}N(\gamma,n)^{13}N$ and $^{16}O(\gamma,n)^{15}O$. Good agreement between measurement and theory was obtained.

2.) In Beam Concentration of Radioactivity

The saturated concentration of radioactivity C_{A_s} in forward direction at distance r from target is given by:

$$C_{A_s} = n \cdot \sigma_{eff} \cdot C_{eff} \cdot \dot{J}(r_0,o^0) \cdot \frac{r_0^2}{r^2} \qquad (1)$$

$$\sigma_{eff} = \int_0^{E_0} \sigma(E) f(E,E_0)\,dE \text{ mit } \int_0^{E_0} f(E,E_0)\,dE = 1 \qquad (2)$$

$$C_{eff} = \left[\int_0^{E_0} \frac{f(E,E_0)}{C(E)}\,dE \right]^{-1} \qquad (3)$$

n — number of atoms of the parent target element (^{14}N resp. ^{15}O) in 1 cm^3 of atmospheric air

σ_{eff} — effective cross section (fig. 4)

C_{eff} — effective conversion factor between exposure rate and flux density (fig.5)

$\dot{J}(r_0,o^0)$ — exposure in forward direction ($\Theta_0 = o^0$) at reference distance r_0 (fig. 6)

$f(E,E_0)$ — normalized X-ray spectrum in forward direction (fig. 3)

The activity concentrations C_{A_s} per unit exposure rate at electron energy E_0= 30 MeV are given in Table I.

3. Activity Concentration outside Beam

The activity concentration in the accelerator room depends on the total activity in the irradiated air volume and the distribution of the radioactive gas in the target room by diffusion and convection. Assuming that photons are emitted within a narrow cone of opening α around 0^0 and of length R the total activity is evaluated by integration of equation (1):

$$A_{S\ tot} = n \cdot \sigma_{eff} \cdot C_{eff} \cdot \dot{J}(r_0,0^0) \cdot r_0^2 \cdot 2\pi R \cdot (1 - \cos \alpha) \qquad (4)$$

For a broad X-ray beam the angular distribution $g(\Theta)$ of the photons must be considered:

$$A_{S\ tot} = n \cdot \sigma_{eff} \cdot C_{eff} \cdot \dot{J}(r_0,0^0) \cdot r_0^2 \cdot 2\pi R \cdot \int_0^{\alpha} g(\Theta) \sin\Theta d\Theta \qquad (5)$$

$$g(\Theta=0^0) = 1$$

In equation (5) the dependence of the bremsstrahlung spectrum from angel (Θ) of emergence has not been regarded. This is a conservative estimation, because the closer the angle of emergence is to 0^0, the harder is the spectrum. Assuming a homogeneous distribution of the total activity produced within the X-ray cone over the accelerator room with volume V_R yields the off-axis concentration:

$$\bar{C}_{A_S} = A_{S\ tot}/V_R \qquad (6)$$

4.) Results

In Table I. the experimental and calculated values are listed. The parameters are: E = 30 MeV, $\alpha = 4^0$, R= 350 cm, $V_R = 500\ m^3$

Table I. Saturation Activity per Exposure Rate $\dot{J}(r_0,0^0)$ $[Ci \cdot m^{-3} \cdot R^{-1} \cdot min]$

^{13}N	^{15}O		
$2,7 \cdot 10^{-6}$	$0,88 \cdot 10^{-6}$	experimental	on beam axis
$1,8 \cdot 10^{-6}$	$1,1 \cdot 10^{-6}$	calculated	
$6,2 \cdot 10^{-11}$	$2,5 \cdot 10^{-11}$	experimental	in centre of target room
$6,0 \cdot 10^{-11}$	$3,8 \cdot 10^{-11}$	calculated	

By comparison of the experimental and theoretical values the following conclusions can be drawn:

1. The activity concentration in the X-ray beam can be calculated with sufficient accuracy by folding the cross section with the normalized bremsstrahlung spectrum.

2. Assuming perfect mixing of the air the activity concentration in the room can be estimated by the ratio between the total activity produced within the X-ray cone and the room volume.

3. If there is no homogeneous distribution of the gaseous activity in the accelerator room, the assumption of a homogenous distribution yields a conservative estimation of the off-axis concentration and hence an overestimation of the dose equivalent as well to the public by exhaust radioactive gas as to the staff entering the target room after shut down of the accelerator.

5.) Dose Equivalent for Submersion

With the dosimetric data in ICRP Publication 30 dose factors for occupational exposure by submersion in a cloud of gas are derived. The dose factors are listed in Table II as a function of the volume of the radioactive cloud.

Table II Dose Factors [rem•h^{-1}•Ci^{-1}•m^3] for Different Volumes of Radioactive Cloud

Volume		∞/2	1000 m^3	500 m^3	100 m^3
^{13}N	γ-Subm.	590	27	22	13
	β-Subm.	370	370	370	370
^{15}O	γ-Subm.	590	28	23	14
	β-Subm.	630	630	630	630

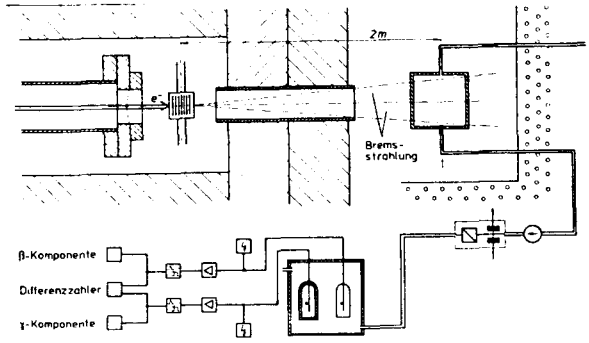

Fig. 1 Experimental set-up for in beam measurement of the activity concentration

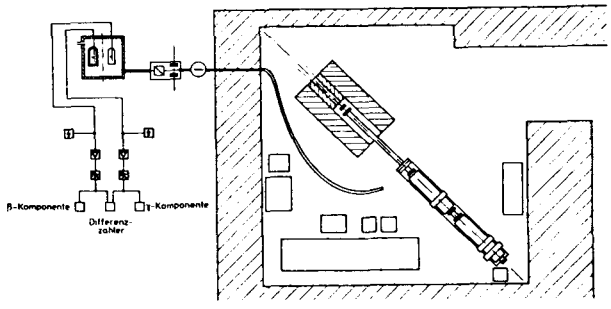

Fig. 2 Experimental set-up for measurement of the activity concentration at different locations in the target room

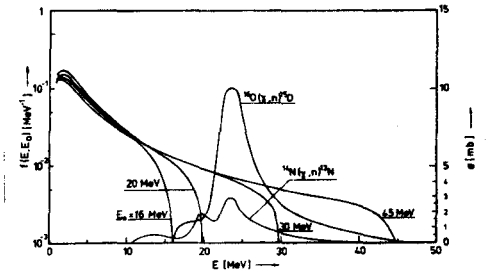

Fig. 3 Normalized Bremsstrahlung spectra and cross sections for (γ,n)-reactions

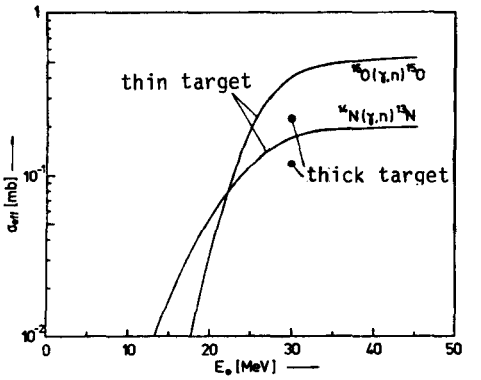

Fig. 4 Effective cross sections for (γ,n)-reactions

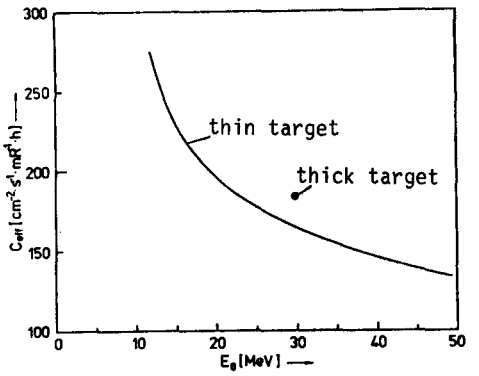

Fig. 5 Effective conversion factor for Bremsstrahlung

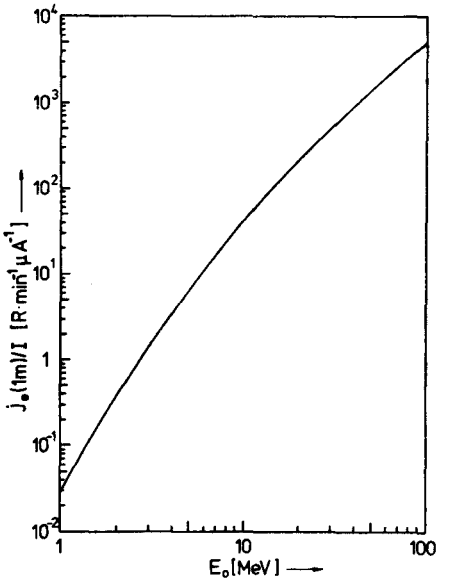

Fig. 6 X-ray emission rate from high-Z target of optimized thickness

PROTECTION AGAINST SCATTERED X-RAYS AT ELECTRON ACCELERATOR INSTALLATIONS

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INTRODUCTION

At accelerator installations those areas towards which the useful beam cannot be directed must be protected against scattered radiation. Shielding requirements for secondary X-rays may be calculated if data describing the intensity and the quality of scattered radiation are available. The intensity of scattered photons is conveniently described in terms of the differential albedo of the scattering material. The attenuation of secondary X-rays in shielding materials may be computed from the energy spectrum of the scattered photons using attenuation factors for monoenergetic photons. Albedo data of common shielding materials, photon spectra of scattered radiation and effective attenuation curves for shielding against scattered radiation were evaluated for 10 MeV to 35 MeV X-rays.

EXPERIMENTAL

The electron beam from the linear accelerator was completely absorbed in a tantalum target. The resulting X-ray beam was well collimated by a series of matched steel disks (total thickness : 1 m) inside a channel through the concrete wall (3 m thick) which shields the experimental area against the accelerator. The intensity of the primary and of the scattered photons was measured with ionisation chambers. A well shielded 5"x5"-NaI-scintillation detector was used for measuring the energy spectra of the scattered radiation.

The differential albedo and the spectra of scattered photons were measured as a function of the following parameters:

a.) scattering material: ordinary concrete, brick, barytes concrete, lead, water slab and a cylindrical water phantom
b.) scattering angle Θs
c.) angle of incidence: $\Theta_0 = 0^0$ and $\Theta_0 = 45^0$
d.) endpoint energy of the primary X-rays (electron energy E_e): 10 MeV to 35 MeV

RESULTS AND DISCUSSION

Differential Exposure Albedo

The differential exposure albedo can be expressed in terms of readily measurable quantities:

$$A_{jx}(E_0, \Theta_0, \Theta_s, \varphi) = \frac{\dot{j}_D(\Theta_s, \varphi)}{\dot{j}_0(E_0, \Theta_0)} \cdot \frac{r^2}{\Delta f \cdot \cos\Theta_0} \qquad (1)$$

$\dot{j}_D$ exposure rate of the scattered radiation at the distance r from the scattering surface

$\dot{j}_0$ exposure rate of the incident radiation at the scattering surface

$\Delta f \cdot \cos\Theta_0$ cross section of the incident beam

The definition of the angles is seen from Fig. 1. The differential albedo was calculated from the measured dose rates at the scatterer and at the detector according to equ. (1). The albedo data were corrected for the contribution of secondary electrons emitted by the irradiated scattering material. Some typical results are presented in Fig. 2 to Fig. 4.

Photon Spectra

The measured pulse height spectra were converted into photon energy spectra using the appropriate response matrix of the 5"x5" NaI-crystal. Four contributions to the total spectrum are clearly discernible:

a.) a broad continuum caused by Compton-scattering of the continuous X-ray photons,
b.) an isolated 511 keV line, due to the annihilation of positrons produced by high energy photons via pair formation,
c.) characteristic X-rays of the scattering material (lead),
d.) a high energy "tail" beyond the Compton-transformed maximum energy of the primary X-ray spectrum. This component is due to secondary bremsstrahlung produced by electrons and positrons from pair formation interactions.

Attenuation of Scattered Radiation

From the photon spectra effective attenuation factors of typical shielding materials for scattered X-rays were computed by folding the product of the flux-to-dose conversion factor and the photon spectrum with the energy dependent dose attenuation factor of the material for monoenergetic photons. Since in practice large areas of the shielding material are irradiated by scattered photons the attenuation factors for maximum build up must be used for a conservative shield design. In Tab. I some selected data for the shielding from a broad beam of scattered X-rays are presented. The results show that for all scattering materials the tenth value layer weakly increases with increasing X-ray energy and with decreasing scattering angle. The dependence on the scattering angle is most pronounced for scattering materials with low atomic number. For the same energy of the incident X-rays and the same scattering geometry the tenth value layer increases with the atomic number of the scatterer.

Tab. I Tenth value layers (TVL in cm) for the shielding from a broad beam of scattered X-rays. The scattering angle Θ_s varies between 140^0 and 40^0 in the case of the water phantom and between 140^0 and 60^0 for all other materials:

E_e in MeV	scattering material	shielding material			
		ordinary concrete		lead	
		TVL_1	TVL_2	TVL_1	TVL_2
10	water phantom	20---28	16---24	1,3---4,1	2,5---4,1
	ordinary concrete	20---23	17---20	1,4---2,6	2,5---3,0
	barytes concrete	24---26	21---23	2,8---3,4	3,8---3,9
	lead	27---28	24---25	3,5---4,0	4,3---4,4
35	water phantom	24---28	21---25	2,8---4,4	4,0---4,9
	ordinary concrete	21---24	17---20	1,6---2,8	2,6---3,1
	barytes concrete	27---27	24---25	3,6---3,9	4,0---4,8
	lead	29---30	26---28	4,3---4,6	4,3---4,9

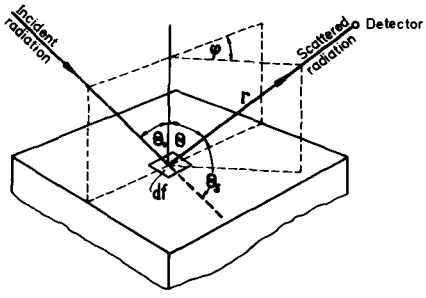

Fig. 1 Scattering geometry

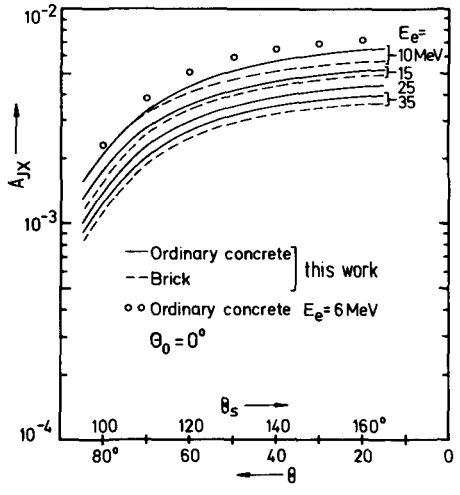

Fig. 2 Differential X-ray exposure albedo of ordinary concrete and brick. (E_e = 6 MeV:Karzmark et al. ; without secondary electrons from the scatterer)

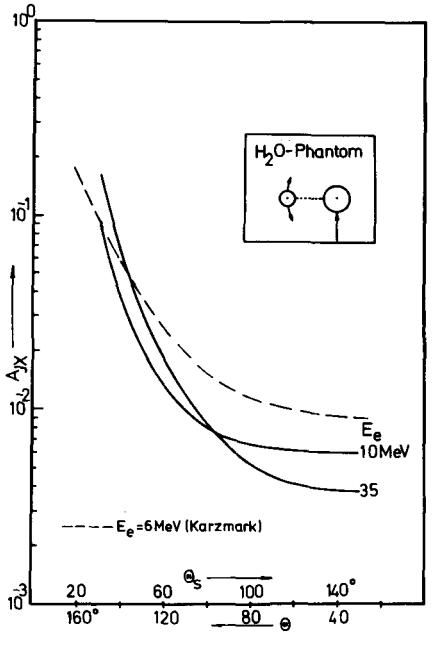

Fig. 3 Differential X-ray exposure albedo for a cylindrical 30 cm diam water phantom (without secondary electrons from the scatterer).

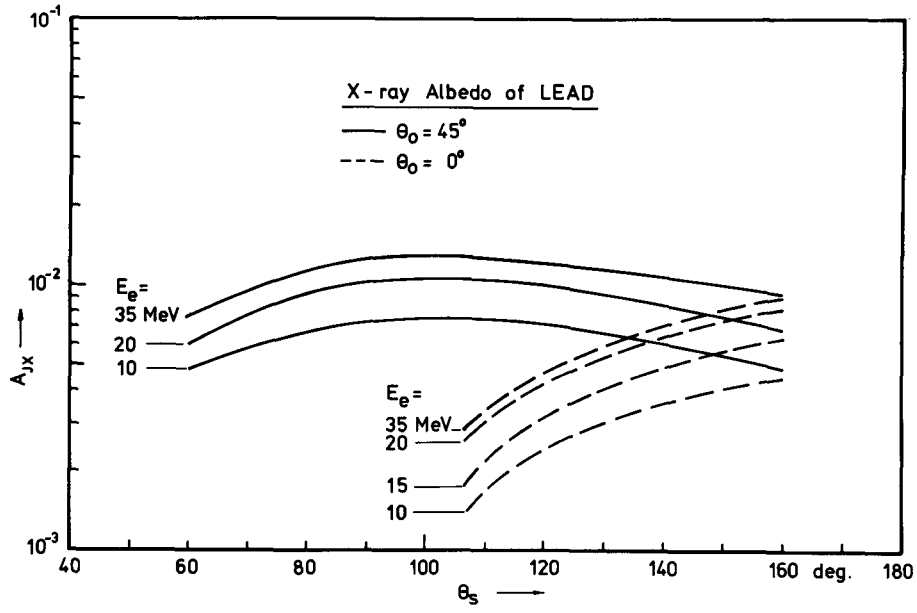

Fig. 4 Differential X-ray exposure albedo of lead (without the contribution of secondary electrons from the scatterer).

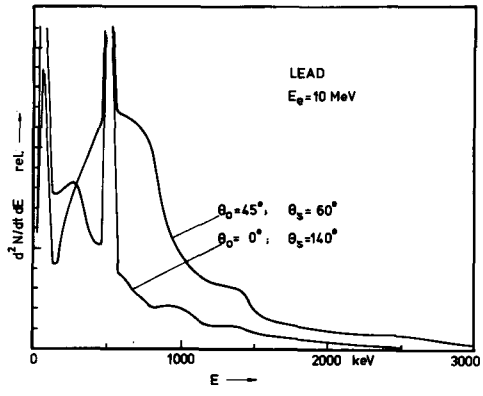

Fig. 5 Photon spectra of 10 MeV X-rays scattered from lead.

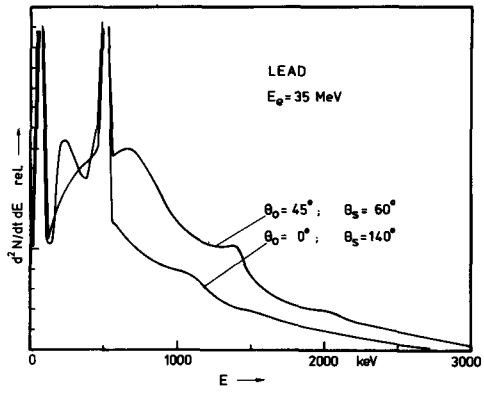

Fig. 6 Photon spectra of 35 MeV X-rays scattered from lead.

SOME RADIATION PROTECTION ASPECTS OF HEAVY ION ACCELERATION

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INTRODUCTION

The CERN Synchrocyclotron has been modified to accelerate heavy ions up to Ne. The present paper describes experiments performed with the most frequently used ^{12}C ions of 86 MeV/nucleon to assess some essential parameters for radiation protection, such as induced activity and secondary-particle production in various target materials. Moreover the attenuation of secondary neutrons in heavy concrete was studied. Finally it will be shown in this report that the experimental secondary-particle production is in good agreement with Monte Carlo calculations.

EXPERIMENTAL RESULTS

Fe and Al targets of 5 x 5 x 0.4 cm and 5 x 5 x 1.2 cm respectively were exposed to a beam of ^{12}C ions of 86 MeV/nucleon. The targets were thick enough to stop the primary ions. Various activation detectors were placed at 50 and 100 cm distance to measure the angular neutron distribution. The following reactions were employed: $^{12}C(n,2n)^{11}C$ for neutron >20 MeV, $^{32}S(n,p)^{32}P$ (E_n >3 MeV), $^{27}Al(n,\alpha)^{24}Na$ (E_n >7 MeV), and finally In foils inside 15 cm polyethylene spheres for neutrons between 0.4 and 10^7 eV.

The angular distributions were calculated using neutron spectra computed as a function of angle for ^{12}C ions of energy up to 900 MeV on iron by H.W. Bertini et al. (Ref. 1) using the HIC-1 code.

The calculated angular fast neutron distribution around the Fe target is presented in Fig. 1 and compared with the measured flux densities derived from the ^{24}Na and ^{32}P activities and In foils in moderator. The number of incident ^{12}C ions on the targets has been determined from the ^{24}Na activity of 0.1 mm thick Al foils placed in front of the target using a cross-section of 24.5 mb for the $^{27}Al(^{12}C,x)^{24}Na$ reaction. All flux densities have been normalized on 10^{11} ions/sec. As can be seen in Fig. 1 a very good agreement exists between the Al and S detector results. At large angles the indium-foil results are higher than the Al, S and theoretical predictions probably due to the contribution of room-scattered low-energy neutrons.

The results for the neutron flux density above 20 MeV are presented in Fig. 2. The ^{11}C-detector results are in good agreement with the calculated angular distribution except at small angles due to the ^{11}C activity produced by charged fragments. Using the ^{11}C data behind a 40 cm shielding block corrected for attenuation, the agreement at zero degrees becomes much better.

Some approximative data for the attenuation length in heavy concrete (density 3.4 g/cm^2) have been obtained from the detectors placed on 40 cm thick blocks under 0^o and 45^o. The attenuation length under 0^o derived from the S detectors is 78 ± 9 g/cm^2 while derived from the ^{11}C detectors an attenuation length of 104 ± 2 g/cm^2 is found as an average for all targets.

Under 45^o an attenuation length of 80 ± 4 g/cm^2 is found for the S detectors and 91 ± 12 g/cm^2 for the ^{11}C detectors. These values shall be considered as upper limits due to the unknown contribution of scattered neutrons.

The Fe target was exposed for 85 minutes with $3.7 \cdot 10^{11}$ ions/sec while the Al target was exposed for 45 minutes with $4.6 \cdot 10^{11}$ ions/sec.

The dose rate from induced radioactivity was measured at various distances from the target with tissue-equivalent ionization chambers. The dose rate a 1 metre from the Fe target versus decay time is presented in Fig. 3. An attempt was made to fit the experimental data with a relationship of the type

$$D = K \phi \log \left(\frac{T + t}{t}\right) ,$$

where:

ϕ is the flux density of ^{12}C-ions causing the activation,
K is a constant to be determined,
T is the irradiation time, and
t is the decay time.

This type of expression has shown to fit induced activity data in steel quite well for protons (see Ref. 2).

As can be seen the relation

$$D = \phi \cdot 10^{-15} \log \left(\frac{T + t}{t}\right) \text{ Gy/h} \qquad (1)$$

fits the experimental data between decay times from 1 to 10^4 h quite well with a maximum deviation of ∿50% for the dose rate at 1 m from the target. The result can be compared with the constant $K = 3.6 \cdot 10^{-15}$ found for high-energy protons (Ref. 2). Therefore it can be concluded that the induced activity in thick steel targets by 86 MeV/n ^{12}C ions is lower by a factor 3.6 per incident ion.

The dose rate as measured at 1 m of the Al target is presented in Fig. 2 together with the dose rate calculated for an Fe target using the formula presented above for the same irradiation time (45 minutes) and beam intensity. It can be concluded that the dose rates are quite similar up to a decay time of ∿70 hours, after which the Al target decays more rapidly as can be expected from the radionuclides produced in Al by ^{12}C ions due to target activation or projectile fragmentation (mainly ^{22}Na, ^{24}Na and ^{7}Be).

CONCLUSION

It can be concluded that the problem of induced activity around heavy ion accelerators is considerably smaller than for high-energy proton accelerators of equal intensity. It has also been shown in this report that the production of secondary neutrons is correctly predicted by the HIC-1 code (Ref. 1).

REFERENCES

1. H.W. Bertini, R.T. Santoro and O.W. Hermann, ORNL/TM-5161 (1976).
2. A.H. Sullivan and T.R. Overton, Health Physics, 11, 1101 (1971).

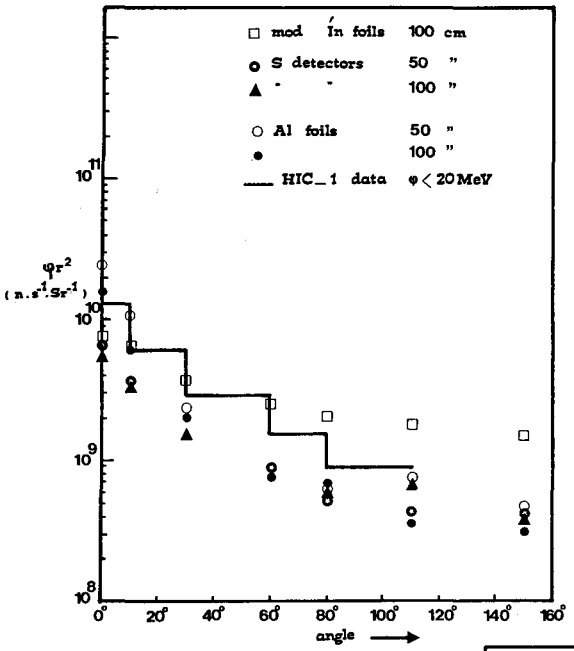

Fig. 1.

Comparison between measured data and HIC-1 calculations for the neutron flux density below 20 MeV around a thick Fe target.

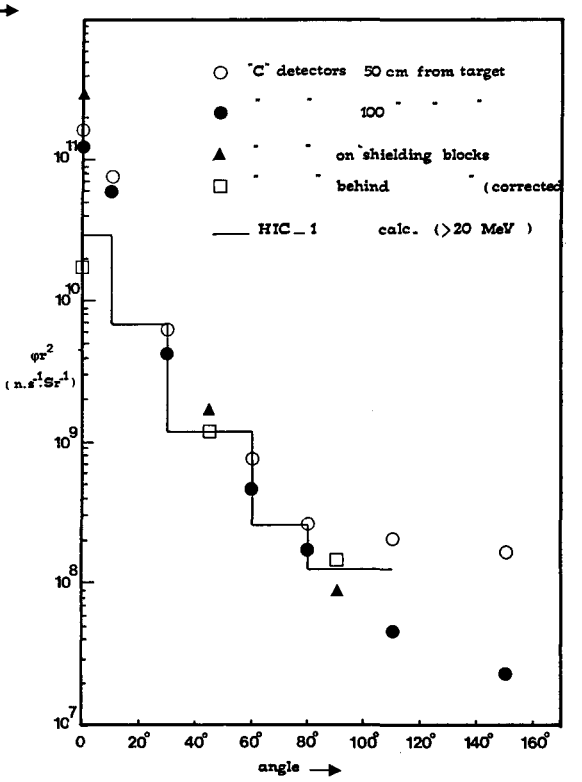

Fig. 2.

Comparison between measured data and HIC-1 calculations for the neutron flux density above 20 MeV around a thick Fe target.

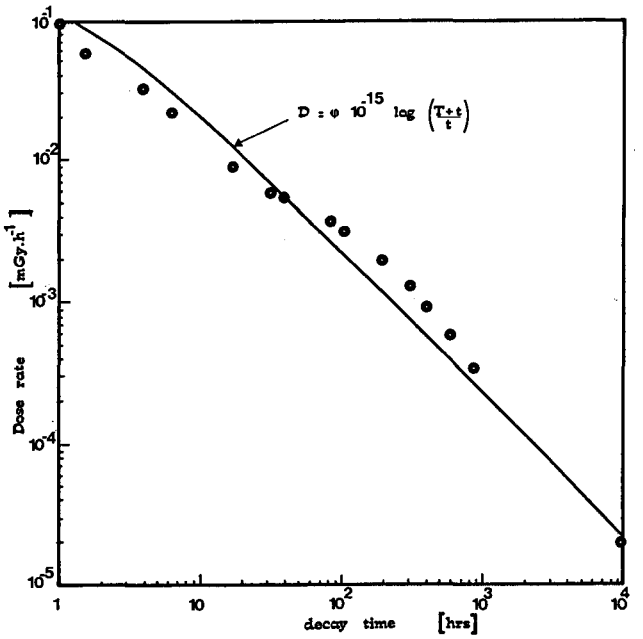

Fig. 3.

Measured decay of induced activity in a 4 mm thick Fe target irradiated with 86 MeV/n C ions for 85 minutes at $3.67 \cdot 10^{11}$ ions/sec.

Dose rates measured at 1 m from target with a TE ionization chamber.

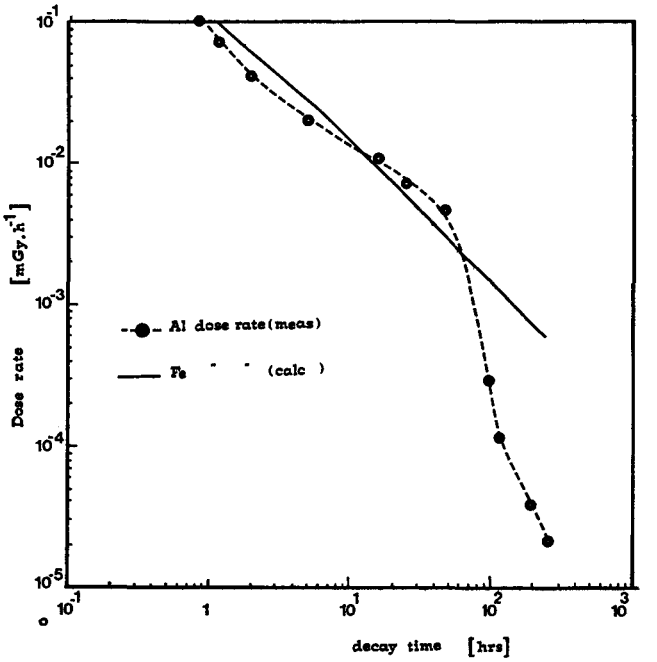

Fig. 4.

Measured decay of induced activity in a 12 mm thick Al target irradiated for 45 minutes at $4.62 \cdot 10^{11}$ ions/sec.

Dose rates measured at 1 m from target with a TE ionization chamber.

RADIATION SHIELDING DESIGN AND PERFORMANCE EXPERIENCE WITH A HIGH POWER ELECTRON LINEAR ACCELERATOR

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Unique radiation shielding problems for personnel protection have been created by the advanced Test Accelerator, ATA_4 a 50MeV electron linear accelerator with peak currents of 10 A, pulse durations of 30 to 50 nsec and pulse repitition rates of 10 per second.

Beam losses occur generally along the entire accelerator structure producing radiation with quite different spectra depending upon loss locations and accelerator structures.

The x-rays produced by the scattered electrons have a low average energy while those produced by the primary beam are high. These facts have created a unique shielding problem.

Nearly 200 penetrations in the shield roof, for pulse power, control and diagnostic systems are each special case problems. Radiation leakage through these penetrations does not necessarily follow the attenuation values suggested in the literature. Attenuation factors are often greater than current literature would suggest.

Equations and actual attenuation data are shown together with the experimental techniques used for the measurements.

Experience with wide dynamic range x-ray monitors designed specifically for intense x-ray measurements is presented. Dose rates have been determined for both radiation areas and areas normally occupied by operations and maintainence personnel.

CONTRIBUTION A LA MESURE DE L'ACTIVITE VOLUMIQUE DANS L'AIR ET DU NIVEAU DE TRAVAIL DES PRODUITS DE FILIATION DE ^{219}Rn

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Pendant plus de trente ans, différentes recherches ont été effectuées sur la chimie de ^{231}Pa à l'Institut Curie. Ces travaux, exécutés parfois dans des conditions précaires, ont entraîné la contamination des laboratoires. Comme l'un des descendants de ^{231}Pa ($T_{1/2} = 3,4.10^4$ans) est ^{219}Rn, les contaminations surfaciques de ^{231}Pa conduisent à une contamination atmosphérique due aux produits de filiation de ^{219}Rn.

$$^{219}Rn \xrightarrow[4s]{\alpha} {}^{215}Po \xrightarrow[1,8ms]{\alpha} {}^{211}Pb \xrightarrow[36,1mn]{\beta^-} {}^{211}Bi \xrightarrow[2,16mn]{\alpha} {}^{207}Tl \xrightarrow[4,8mn]{\beta^-} {}^{207}Pb \quad \text{stable}$$

Afin de pouvoir déterminer l'importance de ces contaminations, nous avons été amenés à mettre au point une méthode simple permettant d'évaluer l'activité volumique des produits de filiation de ^{219}Rn dans l'air (*). Nous avons par ailleurs examiné l'influence de la présence des produits de filiation de ^{222}Rn sur cette mesure. A partir des activités volumiques ainsi mesurées, il est possible d'évaluer le niveau de travail (WL) correspondant qui rend compte du risque encouru par les travailleurs lorsqu'ils se trouvent dans une telle atmosphère (1).

Notre méthode de mesure s'inspire de la méthode que THOMAS a mise au point pour les descendants de ^{222}Rn (2,3). L'activité volumique de ^{211}Pb et de ^{211}Bi est déterminée à l'aide d'un système à 2 équations basé sur le nombre de particules α mesurées au cours de 2 comptages α successifs d'un filtre sur lequel a été préalablement déposé à l'aide d'un aspirateur un échantillon des aérosols de l'air.

EQUATIONS PERMETTANT DE DETERMINER L'ACTIVITE VOLUMIQUE ET LE NIVEAU DE TRAVAIL DE ^{211}Pb ET DE ^{211}Bi

Une étude systématique nous a montré que les coefficients de variation des activités volumiques étaient minimums lorsque la filtration de l'air était effectuée sur un filtre rose pendant 1 mn avec un aspirateur ayant un débit de $0,45 m^3 mn^{-1}$ et que le premier comptage d'une durée d'une minute était effectué 1 mn après l'arrêt de la filtration tandis que le second comptage était effectué entre la 15ème et la 30ème minute après la fin de la filtration. Dans ces conditions, l'activité volumique de ^{211}Pb (C_A) et de ^{211}Bi (C_B) en $kBqm^{-3}$ est donnée par la relation

$$\begin{aligned} C_A &= -1,4607.10^{-7} I_1 + 3,6123.10^{-6} I_2 \\ C_B &= 6,9892.10^{-5} I_1 - 3,1228.10^{-6} I_2 \end{aligned} \qquad (I)$$

I_1 et I_2 étant le nombre de particules α émises respectivement au cours du 1er et du 2ème comptage.

Nous avons pu montrer également que le niveau de travail correspondant aux activités volumiques était donné par l'expression :

$$WL(^{219}Rn) = 0,158 C_A + 0,009 C_B \qquad (II)$$

(*) Ce travail a fait l'objet d'une partie du mémoire présenté le 29 juin 1983 pour l'obtention du titre d'ingénieur CNAM (Paris).

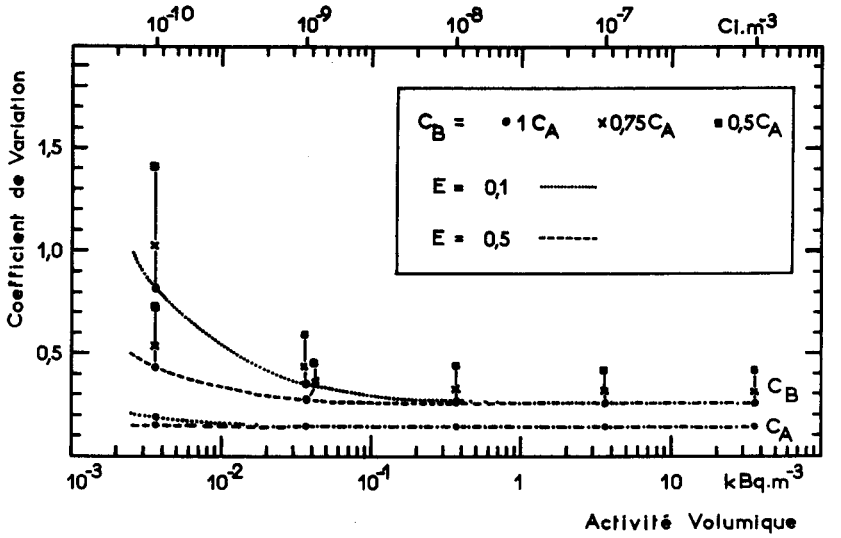

Figure 1 : Coefficient de variation des activités volumiques de $^{211}Pb(C_A)$ et de $^{211}Bi(C_B)$ en fonction de C_A, de C_B/C_A et du rendement du compteur dans $4\Pi sr(E)$.

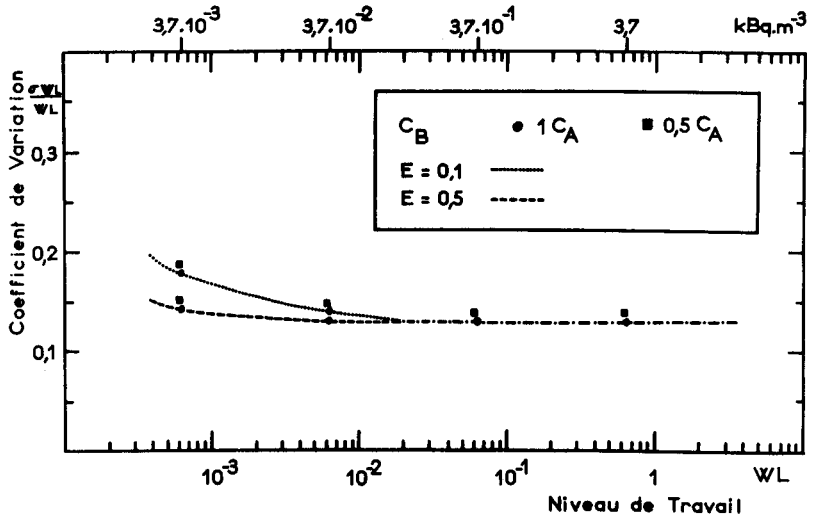

Figure 2 : Coefficient de variation du niveau de travail en fonction du niveau de travail (WL) et du rendement du compteur dans $4\Pi sr(E)$.

le niveau de travail étant égal à l'unité lorsque l'énergie volumique potentielle α est égale à 20,8μJm^{-3}. L'expression précédente est en accord avec celle donnée récemment par CRAWFORD (4).

CALCUL DU COEFFICIENT DE VARIATION DES ACTIVITES VOLUMIQUES ET DU NIVEAU DE TRAVAIL

Dans les conditions définies ci-dessus, nous avons calculé les coefficients de variation des activités volumiques et du niveau de travail pour différentes valeurs de ces deux paramètres (1) (figures 1 et 2). Malgré l'importance du coefficient de variation de l'activité volumique de ^{211}Bi pour des valeurs inférieures à 0,1kBqm^{-3}, le niveau de travail peut être déterminé avec une précision tout à fait acceptable du fait de l'importance relativement faible de l'activité volumique, de ^{211}Bi dans la détermination du niveau de travail (équation II).

DETERMINATION EXPERIMENTALE DE L'ACTIVITE VOLUMIQUE ET DU NIVEAU DE TRAVAIL DES PRODUITS DE FILIATION DE ^{219}Rn

Notre méthode a été tout d'abord testée au moyen d'une série de 10 prélèvements, à des jours différents, des aérosols de l'atmosphère d'une pièce confinée dans laquelle une source de ^{227}Ac (descendant de ^{231}Pa) avait été placée. Les fluctuations observées des activités volumiques de ^{211}Pb et de ^{211}Bi et du niveau de travail étaient un peu plus importantes que celles prévues par les figures 1 et 2. Nous avons attribué cette différence à des variations du pouvoir émanateur de la source de ^{227}Ac dues à l'influence des conditions atmosphériques (1).

Des mesures effectuées dans différentes pièces dont on soupçonnait l'existence de taches de contamination fixée de ^{231}Pa, ont montré que notre méthode permettait d'apprécier facilement une contamination atmosphérique par les produits de filiation de ^{219}Rn de l'ordre de quelques centièmes d'unité de travail avec une précision voisine de celle donnée par les figures 1 et 2 (1).

D'autre part une série d'expériences a été effectuée sur la mesure des activités volumiques de ^{211}Pb et de ^{211}Bi et du niveau de travail de l'atmosphère d'une pièce confinée dans lesquelles émanaient simultanément des sources de ^{219}Rn et de ^{222}Rn. Ces expériences ont toujours montré que la présence des produits de filiation de ^{222}Rn n'entraînait pas d'erreur importante quant à l'évaluation du niveau de travail même dans le cas extrême où il n'existait dans l'atmosphère de la pièce que les produits de filiation de ^{222}Rn (1).

CONCLUSION

Cette étude a permis de mettre au point une méthode simple et peu onéreuse pour déterminer l'activité volumique de ^{211}Pb et de ^{211}Bi : il suffit en effet pour la mettre en oeuvre de disposer d'un compteur α et d'un aspirateur muni d'un filtre. Cette méthode est crédible car elle permet de déterminer une valeur du niveau de travail proche de la valeur réelle dans le cas où les descendants de ^{222}Rn sont prépondérants dans l'atmosphère. Elle est suffisamment précise puisque même dans les conditions les plus défavorables (quelques centièmes d'unité de travail), elle permet d'évaluer le risque encouru par les travailleurs avec une précision meilleure que 20% (1).

BIBLIOGRAPHIE SUCCINTE

1. J.C. Franck. Contribution à la mesure de l'activité volumique dans l'air des produits de filiation de ^{219}Rn et de ^{220}Rn. Mémoire d'ingénieur CNAM(Paris)1983.

2. J.W. Thomas. Modification of the Tsivoglou method for radon daughters in air. Health Physics 19, 691, 1970.

3. A. Busigni, C.R. Phillips. Uncertainties in the measurement of airborne radon daughters. Health Physics 39, 943, 1980.

4. D.J. Crawford. Radiological Characteristics of ^{219}Rn. Health Physics 39, 449, 1980.

ACCESSABILITY WITHIN A NUCLEAR POWER PLANT IN POST ACCIDENT SITUATIONS

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Viki Lindblad
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INTRODUCTION

The interest in how an accident with moderate to severe core damage will affect accessability within a nuclear power plant origins from the accident at TMI-2, where high radiation levels were obtained in the auxiliary building. The accessability was thereby seriously affected.

In correspondence with the American requirements, published as item 2.1.6b in NUREG 0578; TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations, the Swedish Nuclear Power Inspectorate in 1980 demanded that the accessability in post accident situations, within two of the Swedish State Power Board's (SSPB) nuclear power plants (one BWR, one PWR), should be investigated before allowing their initial loading. This kind of analysis have now been performed for all plants owned by the SSPB, namely four Asea Atom BWRs (Ringhals 1, Forsmark 1, 2 and 3) and three Westinghouse PWRs (Ringhals 2, 3 and 4).

Based on the findings in the basic investigations, a second step is now in progress, consisting of an action plan to improve the accessability in vital areas.

BASIC ANALYSES

The aim of these investigations was:

- To determine where in the reactor building (BWR)/auxiliary building (PWR) high radiation levels are obtained
- To determine the areas to which access is required after an accident
- To calculate the radiation doses as a function of time in these areas
- To serve as a basis for improving accessability in a post accident situation.

To begin with, the types of accidents involved and the amount of nuclides to be released from the core were specified. For BWRs the accident chosen is a large break in the primary system (LOCA). Based on Regulatory Guide 1.3 and early experiences from the accident at TMI-2 the following assumptions were made as to how large a fraction

of the core inventory of different radionuclides is released:

100% of the noble gas inventory 2.5% of the iodine inventory	to the containment atmosphere
25% of the iodine inventory 10% of the inventory of Cs isotopes 1% of the inventory of a number of selected nuclides	to the containment water

The above assumed activity release is based on the TMI experiences that were available in the spring of 1980. Later analyses have indicated other release fractions as being more realistic, the most important being a 50% release of Cs. In our investigations, the same releases as in the table above have been used for all BWRs in order to be able to make comparisons. However, the influence of Cs on the doserates is listed separately, providing the possibility to adjust the figures according to higher release fractions.

For PWRs two main types of accidents were discussed, either a large LOCA or an accident where the primary system remains intact. In both cases it was assumed that the release of radioactive nuclides corresponds to the figures given in Regulatory Guide 1.4. Modified by the experiences from TMI-2 the final version is:

LOCA:	100% noble gases 2.5% iodine	to the containment atmosphere
	50% iodine 50% Cs 1% other nuclides	to the containment water
Intact primary system:	100% noble gases 50% iodine 50% Cs 1% other nuclides	to the reactor coolant

It was assumed that the entire release from the core takes place instantaneously and that the activity will be homogeneously distributed in the water and gas phases.

The next step in the basic investigations was to determine the systems, or parts of systems, which may become contaminated after an accident.

In selecting these systems, the intention was to cover as many operational cases as possible. In reality, the systems which will be utilized naturally depends on the circumstances surrounding the accident.

The following systems, or parts of systems, may circulate radioactive medium in the reactor/auxiliary building:

PWR: Residual Heat Removal, Containment Spray, Safety Injection, Chemical and Volume Control, Sampling, Containment Hydrogen Control, Vents and Drain, Waste Processing, Gaseous Waste Processing.

BWR: Residual Heat Removal, Containment Spray, Emergency Core Cooling, Sampling, Containment Hydrogen Control, Leakage Collection, Reactor Building Ventilation.

A review was then carried out, system by system, in order to establish in which rooms the systems are installed.

The next step was to determine to which rooms, and also at which point in time, access is required after an accident has occured. Apart from the rooms in which measures closely connected with the accident shall be taken, the access requirements are largely based on normal operational routines and thus confined to inspection rounds, some checks and minor system changes. In this phase no account was taken of any repairs or other non-routine measures.

The dose rates, for different times in the interval between one hour to one year after the accident, were then calculated for interesting areas, using computer programs based on the point-kernel method. The radiation levels in the reactor/auxiliary building were also illustrated by coloured lay-outs.

In general it was found that access is possible to most areas within a few days of the accident including the main control room for both BWRs and PWRs, where the radiation level will be less than 0.01 mSv/h (1 mrem/h) one hour after the accident due to direct radiation from the contaminated systems.

The main exceptions for both BWRs and PWRs are the rooms where the Residual Heat Removal (RH), the Containment Spray (SP) and the Safety Injection (SI) systems are installed. E.g. for PWRs the dose-rates in the RH- and SP-pumps rooms exceed 10 Sv/h (1000 rem/h) during the first week of the accident. Also the rooms where the charging pumps are located will be inaccessible during the following year, with a radiation level of 6 Sv/h (600 rem/h) an hour after the accident. The accessability to areas adjacent to the rooms discussed above is naturally also affected by the high radiation levels.

Another weak spot is the Sampling System since we, with the scenario described above, wouldn't be able to extract any representative samples of neither water nor gas from the containment, with the equipment presently installed.

Special studies in the LOCA case have also been made regarding containment penetrations, since the dose rate directly outside the penetrations can be considerably higher then the general radiation level.

SUBSEQUENT WORK

The findings of the basic investigations are now being used in the second step, which consists of an action plan either to improve the possibility of, or to eliminate the need for, access in a post accident situation.

Among the measures to be considered are shielding of certain strategic pipes, doors and penetrations, which would reduce the general radiation level in many areas.

Another measure discussed is to improve supervision of rooms with high doserates containing vital equipment, e.g. pumps, by means of television, lead glass or maybe relocation of interesting instrumentation to a more easily accessible area.

The use of robots in carrying different instruments to collect information from especially radioactive areas will also be taken into consideration.

The sampling systems are presently being redesigned at all units, in order to get a functioning system in a situation when severe core damage has occured. New sampling points, new laboratories and better analysing equipment are to be installed.

As a result of the findings in the basic investigations some changes in the manuals for post accident operation may be implemented, like bypassing of filters, different operation of certain pumps, etc.

In this second phase also maintenance and repair aspects are included to a certain degree. For instance some kind of "minimum required maintenance" of vital equipment is being considered. When discussing repairs, a vital step is to decontaminate the system or component in question. An important aspect in this work is therefore to investigate the possibilities of flushing the systems.

The measures exemplified above are to be thoroughly investigated. When discussing improvements concerning post accident situations one has to bear in mind that also normal operation in some parts will benefit.

As a continuation of the work concerning how an accident with severe core damage affects the accessability, the next step will be to study the effect these high radiation levels will have on existing materials in the contaminated systems, e.g. organics, lubricants and hydraulic fluids, in the long term perspective. This work has already been commenced on a small scale basis.

The aim of this work is to implement improvements concerning accessability during an accident, as well as to raise the general awareness of post accident situations among different categories of plant personnel. The subsequent work described in this paper is going to be continued within the next few years, in close connection with work on accident management presently in progress.

HEALTH PHYSICS IN FUSION REACTOR DESIGN - THE APPLICATION OF CANDU EXPERIENCE WITH TRITIUM

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Experience in the control of tritium exposures to workers and the public gained thorough the design and operation of Ontario Hydro's nucelar stations has been applied to design studies on emerging fusion reactor concepts. Application of dose management principles in the development of occupational dose and public exposure targets for fusion devices will be discussed. Ontario Hydro performance in occupational tritium exposure control will be examined, and the role of protective clothing, tritium monitoring and internal dosimetry highlighted. Environmental monitoring programs and accumulated data will be summarized. Particular attention will be paid to application of CANDU tritium exposure-risk management experience to fusion facilities. Priorities and progress in health physics related research and development sponsored by the Canadian Fusion Fuels Technology Project, which builds on the extensive CANDU information base will be reported.

RADIATION EXPOSURE OF INDIAN URANIUM MINERS AND ESTIMATE OF ASSOCIATED RISK

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1. INTRODUCTION

The uranium mine at Jaduguda, India is semi-mechanised and employs a large work force for underground ore winning operations. The strength of workers has increased from about 700 in the 1960s to 1250 at present. The radiation environment in the mine has been under surveillance from the very beginning to keep the radiation exposure of workers within the prescribed limits. The trend of radiation exposure during the past 18 years and an estimate of the mortality risk are presented in this paper.

2. METHODOLOGY AND RESULTS

2.1 Estimate of Radon Daughter Concentrations

The practice has mainly been to monitor the mine air for radon, using the scintillation cell technique. Radon daughters being the major contributors to the radiation dose, their concentrations (WL) have often been estimated with those of radon. At other times the equilibrium factor (WL:Rn) and radon levels have been used to estimate the radon daughter concentrations in Working Level (WL) units. The concentrations of radon and radon daughters ranged from 0.7 to 13.0 Bq/ℓ and 0.02 to 0.52 WL respectively. The mean F values varied from 0.10 to 0.43.

2.2 Duration of Exposure

For a reasonably realistic exposure estimate the number of hours spent underground annually by groups of workers was obtained from a sample survey. The results of this survey are given in Table 1.

Table 1, Estimate of time spent underground by groups of miners

Miner group	% of total work force	Time spent underground (h/y)		
		Moving about	On regular job	Total
Drillers	12	115	1606	1721
Muckers	26	132	1934	2066
Auxiliary workers	48	113	1786	1899
Maintenance staff	9	762	889	1651
Supervisors	3	762	889	1651
Engineers	2	380	380	760

2.3 Internal Exposure Estimate

The radon daughter levels in the mine during different operations over the years have varied widely. The cumulative exposure (WLM) of miners in different trades is estimated by multiplying the radon daughter concentrations (WL) with the corresponding durations of exposure. Table 2 shows the trend in cumulative exposure of the miners to radon daughters during 18 years of radiological surveillance.

Table 2, Estimated cumulative exposure of different categories of miners (WLM)

Year	Cumulative exposure (WLM)		
	Drillers	Muckers	Others
1965	4.9 ± 2.6	2.1 ± 1.1	1.7 ± 1.0
1966	2.3 ± 1.2	3.5 ± 1.8	2.2 ± 0.9
1967	2.0 ± 1.1	5.2 ± 2.7	1.6 ± 1.1
1968	3.8 ± 2.0	3.2 ± 1.6	2.6 ± 1.5
1969	4.1 ± 1.2	6.5 ± 3.4	2.4 ± 1.7
1970	2.1 ± 1.1	3.0 ± 1.2	0.8 ± 0.5
1971	1.7 ± 0.9	2.0 ± 1.1	1.1 ± 0.8
1972	0.7 ± 0.6	1.6 ± 1.4	1.4 ± 1.3
1973	0.6 ± 0.3	0.7 ± 0.3	0.7 ± 0.5
1974	1.6 ± 0.6	5.5 ± 5.0	2.0 ± 1.6
1975	2.2 ± 0.8	2.3 ± 1.9	3.5 ± 1.7
1976	5.5 ± 4.5	2.5 ± 1.1	0.7 ± 0.1
1977	1.6 ± 0.6	1.7 ± 0.7	1.4 ± 0.7
1978	0.8 ± 0.2	1.4 ± 0.7	1.5 ± 0.8
1979	2.6 ± 1.0	2.1 ± 0.6	1.7 ± 0.3
1980	2.2 ± 0.2	0.4 ± 0.2	0.2 ± 0.1
1981	1.8 ± 0.5	1.7 ± 0.5	1.4 ± 0.2
1982	1.8 ± 0.8	1.3 ± 0.6	1.1 ± 0.4

The effective dose-equivalent from WLM unit to mSv is obtained from the curve given in International Commission on Radiological Protection Publication No. 32 (ICRP 81). With the unattached fraction (radon daughters) of 0.07 for the mine, the dose-equivalent works out to 10 mSv/WLM.

2.4 External Exposure Estimate

The external gamma radiation levels in different areas of the mine range from 2 to 10 uGy/h, averaging about 5 uGy/h. The effective dose-equivalent computed from the average dose-rate and the duration of exposure ranges from 4 to 10 mSv/y.

2.5 Total Effective Dose-Equivalent

The total dose-equivalent is the sum of the dose-equivalents from internal and external sources (ICRP 77). The weighted mean of the effective dose-equivalent — both per caput and collective — for the miners are shown in Fig. 1.

2.6 Risk Estimate

Increase in the incidence of lung cancer among American and Czech uranium miners has been attributed to their prolonged exposure to radon daughters (Lundin 71, Sevc et al 76). Risk estimation being a complex problem, widely varying values of the risk factor are given in the literature for radiogenic lung cancer. After a critical evaluation of these values a lifetime risk factor of $1.5 - 4.5 \times 10^{-4}$/WLM has been suggested on epidemiological considerations (UNSCEAR 77; ICRP 81). Based on the data collected, the general level of mortality risk for Indian uranium miners from radiogenic lung cancer for 30 years exposure is given in Table 3.

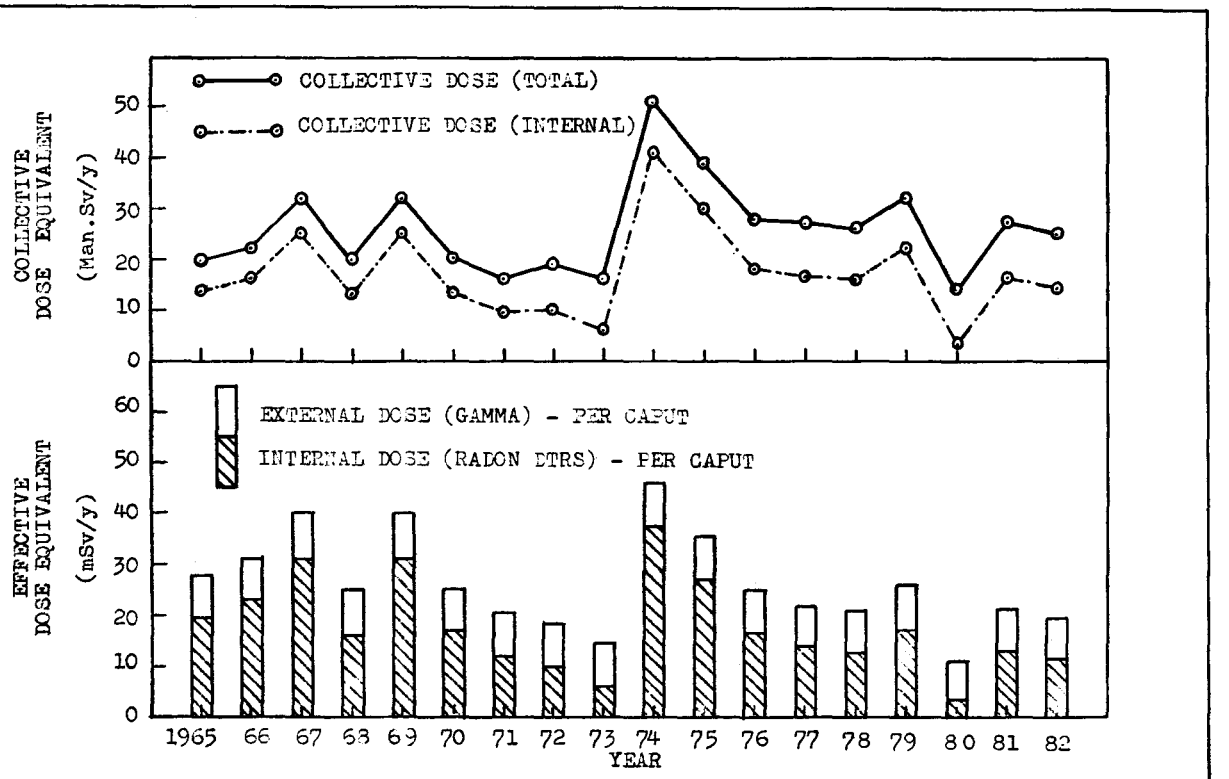

FIG. 1. TREND OF RADIATION EXPOSURE OF INDIAN URANIUM MINERS

Table 3, Estimated life-time risk of radiogenic lung cancer for 30 years exposure

Miner group	Mean cumulative exposure (WLM/y)	Individual risk	Collective risk per 1000 miners
Drillers	2.3	$1.1 - 3.2 \times 10^{-2}$	1 - 4
Muckers	2.7	$1.2 - 3.6 \times 10^{-2}$	3 - 9
Others	1.6	$0.7 - 2.1 \times 10^{-2}$	4 - 13
Weighted mean	1.8	$0.8 - 2.4 \times 10^{-2}$	8 - 24

Corrected for percentage composition.

3. DISCUSSION

From Table 1 the weighted mean of the time spent underground is 1868 h/y. The exposure of the miners has rarely reached the limit of 5 WLM/y; generally it has been much lower. The drilling and mucking crews are exposed to relatively higher concentrations (Table 2) but the weighted mean of the cumulative exposure to radon daughters is 1.8 WLM/y (Fig. 1). This will account for an average whole body dose of 18 mSv/y from internal exposure. The trends of total per caput and collective dose estimates show fluctuation around a mean of 26 mSv/y and 26 man.Sv/y respectively.

The risk estimates presented in Table 3 are based on the risk factors derived from cumulative exposures above 120 WLM. The risk factor may be different at lower exposures (below 60 WLM). However, no case of lung cancer has so far been detected among Indian uranium miners.

The estimated lung cancer risk of 8 - 24 per 1000 miners in 30 years is comparable to the risk of fatal accidents of 0.23 - 0.34 per 1000 miners per year in Indian metal mines (Patnaik & Ghose 82).

4. ACKNOWLEDGEMENT

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5. REFERENCES

1. ICRP 77: Recommendations of the International Commission on Radiological Protection, ICRP Publication 26 (1977)

2. ICRP 81: Limits for Inhalation of Radon Daughters by Workers, ICRP Publication 32 (1981)

3. Lundin, F.E., Wagoner, J.K. and Archer, V.E. 71: Radon daughter exposure and respiratory cancer — Quantitative and temporal aspects, NIOSH and NIEHS Joint Monograph No. 1(1971), USA

4. Patnaik, N.L. and Ghose, A.K. 82: Statistics of occupational risk in Indian mines and identification of training needs, J. Mines, Metals & Fuels, April 1982

5. Sevc J., Kunz E. and Placek, V. 76: Lung cancer in uranium miners and long term exposure to radon daughter products, Health Physics, 30: 433-437 (1976)

6. UNSCEAR 77: Levels and effects of ionizing radiations, Report of the United Nations Scientific Committee on Effects of Atomic Radiation, United Nations, New York (1977).

RADIATION RISKS TO THE PATIENT IN DIAGNOSTIC RADIOLOGY: AN APPLICATION OF RADIATION DETRIMENT

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Radiation detriment and effective dose equivalent have been introduced as concepts for assessing the health impact of radiation received by more than one tissue from whole or partial body exposures. Radiation detriment can be described on a uniform scale and is a useful concept for assessing patient risk from diagnostic radiology, but the effective dose equivalent cannot be described in a unique way for the same purpose.

Radiation detriment (G) can be defined by the relationship:

$$G = \sum s_T \, r_T \, \bar{D}_T .$$

The quantity $\bar{D}_T$ is the average absorbed dose (cGy)in a given tissue. The quantity r_T is the cumulative lifetime risk of incidence per cGy for a health effect occurring in a particular tissue. The magnitude of each risk depends on the specific tissue that is exposed, and in some circumstances whether the individual is a male or female. In some cases the risk is a function of the age at which an individual was exposed. Because the periods of followup for the populations that are under study are not complete, there are uncertainties in estimating cumulative lifetime risks of incidence. It is likely that the manner in which the risk for each health effect should be projected in time is not the same. In addition, there is considerable difficulty in projecting the dose-response relationship to absorbed doses as low as 1 to 10 cGy.

The factor s_T is the relative severity of the particular health effect induced in the tissue. If one equates radiation detriment to risk of incidence, then s_T would have a value of 1. If one equates radiation detriment to risk of mortality, then s_T would be the fraction of individuals not surviving the health effect for 5 years, or some other agreed upon convention. One could also express radiation detriment as risk of incidence modified by the severity of the health effect or its treatment.

Radiation detriment does not require the use of a linear nonthreshold dose response. Any method can be used to arrive at the values of r_T from observed scientific evidence. However, once r_T has been arrived at for the lower dose region, a proportional relationship is used to compute risks in the range of absorbed dose encountered in diagnostic radiology. This would generally be less than 10 cGy.

If an individual receives a uniform whole body absorbed dose (D_{wb}), then the average absorbed dose $\bar{D}_T$ is equal to D_{wb} for each tissue and the equation can be rewritten as:

$$D_{wb} = \frac{G}{\sum s_T \, r_T} = D_E .$$

This formulation is the effective absorbed dose (D_E) or the effective dose equivalent (H_E) if the radiations have different quality factors. If the sum of the $s_T \, r_T$ components is a constant, then the effective absorbed dose would be unique for a given radiation detriment. If the sum of the $s_T \, r_T$ components is not a constant, then the effective absorbed dose will not be unique.

An example where a single set of values for s_T and r_T have been selected is the set of current ICRP 26 risk factors for genetic and somatic health effects (1). These data apply to a mixed adult population and were intended for use in developing a radiation protection scheme for workers. The ICRP 26 risk factors are equivalent to mortality for cancer and leukemia, and to the incidence of serious hereditary illness for the first two generations of offspring. The sum of the s_T r_T components for these risk factors is 1.65×10^{-4}, and is a constant. Therefore, for any value of radiation detriment the effective absorbed dose is fixed.

An example where the important differences in the values of the cumulative lifetime risks of incidence (r_T) between males and females are recognized is the analysis of Laws and Rosenstein (2). Here, the effective absorbed dose concept was applied to diagnostic radiology for somatic effects. The risk factors are for incidence modified by a relative severity, patterned after the suggestions in ICRP 14 (3). This separation by sex acknowledged that breast cancer occurs mainly in women and that there are observed differences in risks between the sexes for thyroid cancer and leukemia. The sum of the s_T r_T components for the risk factors given in that analysis are 1.8×10^{-4} for the female and 9.9×10^{-5} for the male. Therefore, for the same value of radiation detriment, the effective absorbed dose for the male is 1.8 times that for the female.

This example is still a relatively simple one because only one major difference occurred, the presence of the female breast. The situation becomes more complex when one considers all the somatic and genetic health effects of importance. A more complete example is for the cumulative lifetime risks of incidence of various health effects for an average adult female. The relative severity is now 1. However, the female can be feritle or non-fertile, or she can be pregnant at the time of exposure. Table 1 gives illustrative risks for various cancers in the exposed adult (derived from the 1980 BEIR report)(4), for genetic illness in offspring of the exposed adult (from ICRP 26)(1), and for two somatic effects expressed in the child exposed *in utero* (from the 1980 BEIR and the 1977 UNSCEAR reports)(4,5).

Table 1. Cumulative Lifetime Risks of Incidence for Health Effects (Average Adult Female)

Health Effect	r_T (per cGy) $\times 10^{-4}$ Non-Fertile	Fertile	Pregnant
Breast Cancer	1.2	1.2	1.2
Leukemia	0.18	0.18	0.18
Lung Cancer	0.75	0.75	0.75
Thyroid Cancer	1.2	1.2	1.2
Other Cancers	1.5	1.5	1.5
Genetic Illness		2.0	2.0
Childhood Cancer			5.0
Congenital Effects			10.0
$\Sigma s_T r_T$ ($s_T = 1$)	4.8	6.8	21.8

Using these data, the sums of the s_T r_T components ($s_T = 1$) are 4.8×10^{-4} for the non-fertile female, 6.8×10^{-4} for the fertile female and 21.8×10^{-4} for the pregnant female. Therefore, for the same value of radiation detriment, the effective absorbed dose would be significantly different for each. For the non-fertile female it would be 4.5 times greater than that for the pregnant female. For the fertile female it would be 3.2 times that for the pregnant female. Therefore, it is not possible to establish the effective absorbed dose in a unique fashion.

However, radiation detriment is a useful concept which can be described on a uniform scale. With a consistent set of cumulative lifetime risks of incidence, radiation detriments for diagnostic x-ray exams can give a valuable perspective. Using the risks for the average adult female as an example, Table 2 gives the relative radiation detriment for some common radiographic x-ray exams. In Table 2 the relative radiation detriment is normalized to a value of 1 for that entry with the largest radiation detriment, which has a value of 1.43×10^{-3}. This is the cumulative lifetime risk of incidence for all the health effects applicable to a pregnant female (and unborn child) that underwent a typical barium enema exam, consisting only of the radiographic portion. Note the differences for the lumbar spine and the barium enema examinations as the status of the female changes.

Table 2. Relative Radiation Detriment Considering Various Health Effects - Average Adult Female (a)(b)

Exam	Somatic (Non-fertile Female)	Somatic & Genetic (Fertile Female)	Somatic, Genetic & In Utero (Pregnant, Fertile Female)
Chest	0.003	0.003	0.003
Thoracic Spine	0.05	0.05	0.05
Lumbar Spine	0.04	0.09	0.52
Barium Enema	0.05	0.15	1.00
Mammography (Xerox)	0.06	0.06	0.06
Mammography(Film/Screen)	0.02	0.02	0.02

(a) r_T from BEIR, ICRP and UNSCEAR; $s_T = 1$
(b) a relative radiation detriment of 1.00 corresponds to a radiation detriment of 1.43×10^{-3} (incidence)

For the purpose of presenting risks to the individual patient, an array of the cumulative lifetime risks of incidence that make up the radiation detriment would be the most meaningful. This would lead naturally to a discussion of the seriousness of any of the particular health effects. The following example for a barium enema exam for an adult female is illustrative. Assuming a typical radiographic exam of 4 films, the significant absorbed doses are 0.3 cGy for the active bone marrow, 0.05 cGy for the lungs, 0.4 cGy for other organs in the abdominal region, and 0.8 cGy for the ovaries and uterus (6). In Table 3 are displayed the cumulative lifetime risks of incidence for the corresponding health effects due to radiation, as well as the risks from natural causes. In this case, the risks of childhood cancer or congenital anomalies to the unborn child are the most notable. However, none of the radiation risks approach the natural risks for the same health effect. This kind of display places in perspective the radiation risk to the patient and would be very useful in a clinical setting.

Table 3. Array of Radiation Risks - Barium Enema (Radiographic), Pregnant Adult Female

Health Effect	Tissue Dose (cGy)	Cumulative Lifetime Risk of Incidence (Radiation)	Natural Lifetime Incidence
Leukemia	0.3	1 in 185,000	1 in 125
Lung Cancer	0.05	1 in 267,000	1 in 43
Other Cancers	0.4	1 in 16,700	1 in 4(a)
Genetic Illness	0.8	1 in 6,250	1 in 10
Childhood Cancer	0.8	1 in 2,500	1 in 540
Congenital Effects	0.8	1 in 1,250	1 in 25

(a) All cancers

The task is to develop an authoritative set of cumulative lifetime risks of incidence for the various health effects. Based upon data that are currently available, it appears that a provisional set with the characteristics identified in Table 4 could be developed. This collection would include separate values for genetic illness due to exposure of the the reproductive organs of fertile males and females. It would also include appropriate values for breast cancer in females for various ages at exposure. Values for other cancers would be listed separately, and where significant, would be a function of age and sex. Values for in utero exposures both for childhood cancer and congenital malformations would be necessary for the unborn child, perhaps further defined by trimester of pregnancy. It's time to recognize that different individuals are at different risk, particularly when applying risk concepts to patients in diagnostic radiology. The practice of using general statements is misleading.

Table 4. Desirable Array of Cumulative Lifetime Risks of Incidence for Various Health Effects

	MALE			FEMALE		
Health Effect	Child	Adult	Older Adult	Child	Adult	Older Adult
Genetic Illness		(a)			(a)	
Breast Cancer				(b)	(b)	(b)
Other Cancers	(c)	(c)	(c)	(c)	(c)	(c)
Childhood Cancer					(d)	
Congenital Effects					(d)	

(a) If fertile
(b) By appropriate age categories
(c) Each cancer and leukemia listed separately; by appropriate age categories
(d) If pregnant and by trimester

REFERENCES

(1) Recommendations of the International Commission on Radiological Protection, ICRP Publication 26. Pergamon Press, Oxford (1977)

(2) Laws, P.W. and Rosenstein, M. Health Physics 35, 629 (1978)

(3) Radiosensitivity and Spatial Distribution of Dose, ICRP Publication 14. Pergamon Press, Oxford (1969)

(4) The Effects on Populations of Exposure to Low Levels of Ionizing Radiation. National Academy of Sciences, Washington, D.C. (1980)

(5) Sources and Effects of Ionizing Radiation, United Nations Scientific Committee on the Effects of Atomic Radiation. United Nations, New York (1977)

(6) Kereiakes, J.G. and Rosenstein, M. Handbook of Radiation Doses in Nuclear Medicine and Diagnostic X Ray. CRC Press, Inc., Boca Raton (1980)

COMPUTERIZED PATIENT RADIATION DOSE ASSESSMENT IN DIAGNOSTIC RADIOLOGY

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I. Introduction:

Reduction of radiation dose received by patients undergoing radiological procedures has become of prime importance. This emphasis has been spurred by the fear of possible biological effects including genetic damage to future populations, cancer induction and life shortening. Dose reduction in Radiology can in fact be accomplished with current available technology[1]. To this end work has proceeded in such areas as introducing faster film-screen receptor systems utilizing rare earth phosphors and in using heavy element x-ray beam filtration. Additionally, various devices have been prepared for use in x-ray scatter control as well as optimization of x-ray beam energy and the use of lower attenuating carbon fiber materials for table tops and receptor cassette fronts.

Experience has shown a general lack of both knowledge of specific patient dosage and of understanding on how various equipment operation parameters such as KVP, beam filtration, and patient related factors such as thickness and x-ray field size effect dose delivered.

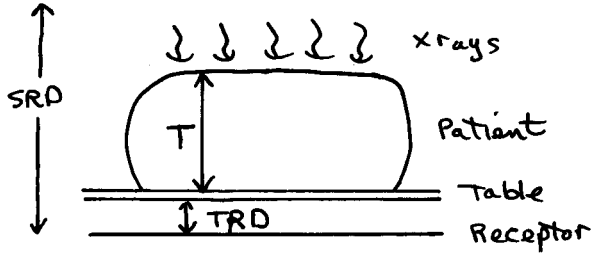
This problem has been further aggravated by the great variety of equipment features and available technologies which are available. The dose data available in the literature in general are not adequate in that they do not provide sufficient information for specific patient parameters and specific equipment utilized. That is, such data tends to be national averages for typical sized patients. There is, however, considerable dose variation due to patient thickness and radiographic techniques. One recent report shows a variation of up to 30 times within the same hospital between different rooms for the same patient.[2]

This work addresses itself to this problem. Here, we identify the factors affecting patient radiation dose, and possible methods of using digital computers to quantitate and report such dose data. From the dose models formulated insight into the full range of factors affecting patient dose and possible dose reduction schemes can be obtained. It is hoped that this work will also contribute to this formulation of more meaningful measures of dose itself as well as measurements to correlate dose with information content and its eventual optimization for a given radiological task.

II. Dose Model:

The dose model used here can be developed as follows: assume some x-ray exposure output value in air at some standardized source to receptor distance (SRD) as follows:

Figure 1



It is quite common to calibrate and express x-ray outputs (X_c) in Roentgens per mAs at a specific distance such as the SRD. From Figure 1 it is seen that the x-ray output at any point in space such as where the skin level would be located (X_s) for a patient of thickness T is given by:

$$X_s = X_c \left(\frac{SRD}{SRD - TRD - T} \right)^2 \qquad Eq.\ 1$$

Where TRD is the table top receptor distance. The expression within the square is just a geometrical inverse square factor indicating expected change in air exposure at the two different points considered in space and X_c is the calibrated output of the unit at distance SRD.

Equation 1 is incomplete in three ways. First it is expressed in units of exposure (Roentgens). In addition it does not include the back scatter (BSF) provided by the patient. Finally the actual film mAs. Taking these into account, the dose to the patient's skin may then be expressed as:

$$D_s = X_c \left(\frac{SRD}{SRD - TRD - T} \right)^2 \cdot F \cdot BSF \cdot mAs \qquad Eq.\ 2$$

Here the factor F is the Roentgen to rad factor which is typically equal to 0.92.

Note that dose to skin here is seemingly independent of beam filtration, receptor sensitivity and losses in attenuation between patient and the receptor due to anti-scatter grids. However, these factors do enter in that they determine the necessary air exposure to obtain the desired radiographic effect (which is proportional to the film mAs). This is an extremely important

point in possible dose reporting schemes because specific knowledge of attenuation occurring in components or efficiency of receptors is not necessary. The only factors which are needed as is seen in equation 2 are the calibrated x-ray output, patient thickness, the source receptor distance (SRD), the patient thickness and the exposure parameters of mAs. In addition, note that equation 2 does depend on beam energy since BSF and X_c both depend on KVP. The factor BSF is also determined from the size of the receptor. F is approximately constant over the typical range of diagnostic x-ray beam energies. If specification of dose is desired at some depth such as the fetal depth, ovarian depth, depth to bone marrow or any specific organ, the percent depth dose, which is critically dependent on the KVP, can be incorporated as follows:

$$D_d = X_c\left(\frac{SRD}{SRD - TRO - T}\right)^2 \cdot F \cdot BSF \cdot mAs \cdot \%DD \qquad Eq.\#3$$

At present the most limiting data to satisfy equation 3 is the %DD data. In contrast to the diagnostic x-ray case exquisite data have been developed for high energy radiotherapy applications. However, data of Trout[3] is used throughout this work. An alternate approach in setting up the patient dose model is that of using tissue air ratios (TAR) which then obviates the need for the geometrical inverse square factor. TAR data is, however, very incomplete at this time. Due to their simplicity, however, future dose models will undoubtedly be designed around TAR's as more such data becomes available.

If in the present work it was found that using Trout's[3] data the variation of dose with patient depth (d) was exponential and could be described with equations as follows:

$$\%DD = A e^{-Kd} \qquad Eq.\#4$$

where K is another exponential as a function of KVP and A is a constant. These exponentials are easily modelled and programmed for rapid calculation of patient dose at any desired depth given beam exposure parameters and geometry.

Simple exponential models were also fitted for the BSF, and the x-ray outputs as a function of KVP. Finally a phase factor was derived which generalizes depth dose data to any electrical phase φ (single or three phase). This phase factor takes the form:

$$\text{Phase factor} = (BK)^{\frac{\varphi-1}{2}} \qquad Eq.\#5$$

Where B is a constant. All of the above factors were incorporated into dose model to determine organ dose for all procedures within the Radiology Department.

Bibliography:

1. Wagner, R.F., Jennings, R.J.: The Bottom Line in Radiologic Dose Reduction. Proceedings of the Soc. of Photo-Optical Instrumentation Engineers, 206:60-66, 1979.

2. Taylor, K.W., Patt, N.L., Johns, H.E.: Variations in X-ray Exposures to Patients. Journal of the Canadian Assoc. of Radiologists, 30:6-11, March, 1979.

3. Trout, E.D., Kelley, J.P., Lucas, A.C.: The Effect of Kilovoltage and Filtration on Depth Dose. In Technological Needs for Reduction of Patient Dosage from Diagnostic Radiology. M.L. Janower, Ed.; C.C. Thomas Publishers, 1963.

REDUCTION OF PATIENT EXPOSURE PRESERVING IMAGE QUALITY IN MAMMOGRAPHY

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ABSTRACT

A programme for mammography optimization in each unit is briefly outlined. The principal parameters affecting dose and image quality (H.V.L., focal spot and exposure) are measured by means of simple devices (exposimeter, star pattern and phantoms) in 28 units. The dose and image quality evaluations show that mammography is not optimized in practice. For example even units employing the same types of X-ray apparatus and film-screen comination present very different exposure values, i.e. from 0.64 R to 10.70 R (1.65×10^{-4} - 27.6×10^{-4} C/kg).

INTRODUCTION

As is well known radiography is one of the most effective means for an early diagnosis of breast carcinoma; therefore the reduction of breast dose,preserving image quality, is an important point.

We carry out a programme for mammography optimization in each unit, named D.Q.M.(Dose and Quality in Mammography).

The steps of the programme are:

a) collection of the working parameters in each unit (1,2)
b) dose and quality evaluation by means of a computerized programme (3)
c) suggestions for improvements in each unit and for starting a quality assurance work with our help

INSTRUMENTS AND METHODS

The working parameters are measured by means of the following devices:

Exposimeter

The exposimeter is made up of two groups of TLD-100 detectors; the first group is put over a 0.5 mm Al filter and the second one below. Detectors and Al filter are held on the top of a plexiglass cylinder 5 cm high and 5 cm wide.

The device is exposed in the same way as a breast 5 cm thick. Exposure is measured by the first group of detectors; the ratio between the second and the first group gives H.V.L. value, on the basis of a calibration curve obtained employing a Mo anode tube with a Be window and a 0.03 Mo filter.

Star Pattern

The pattern is made by etching a Cu coated board for printed circuit (Cu thickness 0.07 mm); the very cheap cost of the pattern makes its employment easier.

A star pattern support defines the geometrical conditions for an optimal radiographic magnification of the pattern image.

The recording system, for the star pattern, consists of a sandwich of three filters and three films instead of a single film, so that whatever the exposure value may be, at least one film can have

a sufficient image contrast.

Plexiglass Phantom

A plexiglass phantom (10x10x5 cm) containing TLD-100 detectors, at various depth (0, 2.5, 5 cm), is exposed in the same way as breast 5 cm thick. The exposure for a radiograph is repeated at least five times, so that even TLD-100 detectors placed inside the phantom can reach a sufficient total exposure.

ITO Kodak for Mammography

As is well known ITO Kodak for mammography is a phantom having inside various kinds of details. It may be used both for optical and for physical evaluation of radiographs. It is exposed in the same way as a breast, and the film is developed by the unit operators.

RESULTS

Measurements were perfomed only in 31 units, 28 units employ Mo anode X-ray tube and 3 units W anode tube; only a unit employs a direct X-ray film, all the others use film-screen combinations.

Table I gives a syntesis the measurements and calculations referred only to Mo anode X-ray tubes (28 units), because the exposimeter calibration for W anode is in preparation.

The Mo anodes are devided into two groups : static and rotating anodes. The second group has high emission intensity, shorter exposure time, larger focus skin distance, so they may present better results.

H.V.L. values for static anodes range in a shorter interval than those for rotating anodes, see Table I and Figure 1.

The exposure distribution ranges in a large interval, see Fig. 2; the mean exposure of all the units is high, but the mean exposure for rotating anodes is about 40% of that for static ones. That is due both to larger focus skin distance and to sensitivity increase of film screen combination for shorter exposure time.

Nominal focal spot dimensions(values declared by the Firms) are smaller than those measured, for static anodes the mean value evaluted is twice the mean nominal focal spot.

The optical density of ITO radiographs measured in a region without details (4.5 cm thick) ranges in a large interval, see Tab.I and Figure 3. The different densities of radiographs do not allow a significant contrast intercomparison.

The computerized programme for breast dose calculation can give dose values at every centimeter of depth. The most significant dose values are presented in Table I : dose in the firt cm (0-1 cm) and dose in the mid plane (2-3 cm).

The resolution in the object plane at 5 cm from the recording system is expressed by MTF(total), product of geometry MTF and film screen MTF, see Fig. 4. In Table I resolution is expressed by the spatial frequency (lp/mm) corresponding to 0.5 value of MTF(total).

The resolution for rotating anodes is better than for static ones, because they present larger focus skin distance and, in some cases, a small focal spot ; but when a rotating anode has a too large focal spot, the MTF(total) is comparable or worse than for static ones.

TAB I. Mo anode units results.

		TOTAL	STATIC ANODE	ROTATING ANODE
N° OF UNITS		28	11	17
EXPOSURE (R)	mean	2.20	2.58	1.04*
	min ÷ max	0.16 ÷ 16.60	0.74 ÷ 10.70	0.16 ÷ 3.60
HVL (mm Al)	mean	0.32	0.29	0.33
	min ÷ max	0.20 ÷ 0.50	0.20 ÷ 0.37	0.20 ÷ 0.50
FOCAL SPOT (mm)	mean	1.34 x 1.01	1.48 x 1.07	1.12 x 0.97
OPTICAL DENSITY	mean	0.84	0.97	0.76
	min ÷ max	0.19 ÷ 1.37	0.27 ÷ 1.37	0.19 ÷ 1.20
1st cm DOSE (mrad)	mean	960	1100	460*
	min ÷ max	90 ÷ 7420	295 ÷ 4260	90 ÷ 1360
midplane DOSE (mrad)	mean	135	149	66*
	min ÷ max	18 ÷ 1110	36 ÷ 500	18 ÷ 158
RESOLUTION**	mean	3.34	3.01	3.47
	min ÷ max	2.4 ÷ 5.2	2.6 ÷ 3.3	2.4 ÷ 5.2

(*) The exposure value of the centre using direct film is not included

(**) Resolution is expressed by the spatial frequency (lp/mm) corresponding to 0.5 value of total M.T.F..

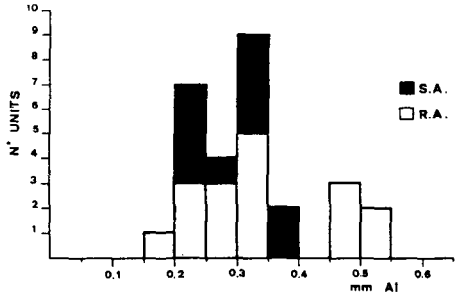

Fig. 1. H.V.L. distribution.

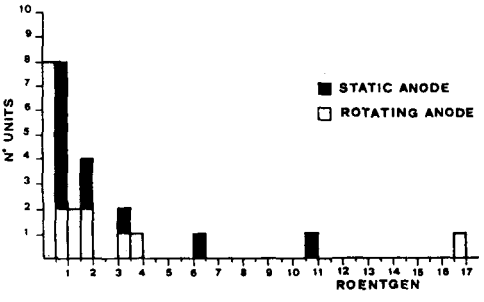

Fig. 2. Exposure distribution.

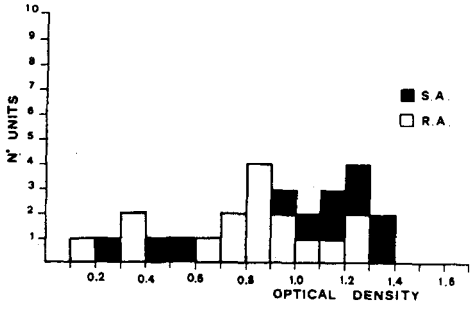

Fig. 3. ITO radiographs desity distribution.

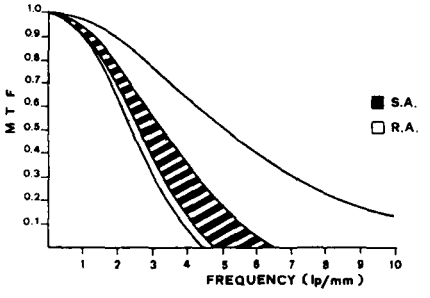

Fig. 4. Total M.T.F. range.

COMMENT

We compared exposure and dose values of Table I with those presented by G.R.Hammerstein et al. in Table V of Ref.(4).

The lowest exposure we found is 0.16 R, in Ref. (4) it is 0.17R, the highest exposure is 16.6 R, in Ref. (4) it is 1.10 R; the lowest mid plane dose is 18 mrad, in Ref. (4) it is 18 mrad; the highest mid plane dose is 1110 mrad, in Ref. 4) it is 83 mrad.

From the analysis of all the results we can see that mammography is not optimized in practice. In fact even units using the same types of X-ray apparatus and film screen combination present very different exposure values, i.e. from 0.64 R to 10.70 R (1.65×10^{-4} to 27.6×10^{-4} C/kg). This variation is due to various causes that have to be found out in each case.

For a significant comparison of the results it is important to know the real exposure parameters(kVp, mA, t), they may be different from those read on the plant instruments.

Another point to take into consideration is the control of film processing, an assured correlation between exposure and optical density is very important for technique standardization.

At last, units that now are working in optimal conditions may start (with our help) a quality assurance activity, employing the simple instruments named above: exposimeter, star pattern and phantoms.

Acknowledgements

We like to thank Mr.R.Nanni and Mr.I.Sermenghi of Laboratorio di Fisica Sanitaria ENEA (Bologna), who performed the reading of TLD-100 detectors.

REFERENCES

1) B.Bagni et al."Organ dose and image quality optimization in mammography .A quality assurance program", Proceeding of the" Seminar on patient exposure to radiation in medical X-ray diagnosis", pg. 245 - 259, Munich-Neuherberg, 27-30 April 1981.

2) B.Bagni et al."Correlazione tra dose e qualità della immagine in mammografia - Programma di ottimizzazione e controllo di qualità" Atti del Convegno "Protection au cours de la utilization des rayonnements ionisants et non ionisants en medicine",pg. 47-84, 14 - 16 Decembre 1981, Cannes.

3) O.Rimondi et al. "Organ dose and image quality optimization in mammography - An image quality assurance program", pg.55-63,Annali di radioprotezione, 1981.

4) G.R.Hammerstein et al. "Absorbed radiation dose in mammography", 130, 485-491,Radiology, 1979.

SCREENING FOR X-RAY FILM PROCESSING PROBLEMS AT DENTAL AND MEDICAL FACILITIES

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INTRODUCTION

Improper film development often leads to higher than necessary skin entrance exposure to patients and or radiographs of less than optimal diagnostic quality. Some facilities with skin entrance exposures within acceptable levels could further reduce patient exposure and improve diagnostic quality of radiographs by optimizing film processing conditions. It is therefore desirable to have a method to measure film processing problems independent of measurements of skin entrance exposure.

A simple method was developed utilizing pre-exposed x-ray films and a questionnaire which may be sent to medical and dental facilities or used at facilities by inspectors during routine inspections. It was determined that films exposed to x-rays or visible light could be stored and later sent or carried to facilities for use in testing film processing systems. Despite anticipated problems with latent image fading, exposure or damage in the mail, and insensitivity of pre-exposed film to non-optimal processing conditions compared with freshly exposed films, this system appears to determine accurately whether a processing system is operating optimally.

The questionnaire is useful in providing a catalog of film processing conditions and workloads that result in optimal film processing as measured with the pre-exposed film. The surveyor can then estimate whether a system is under optimal conditions when the test film is processed at a time that does not represent worst case processing conditions. Optimal processing conditions measured at one facility may then be used to suggest changes at facilities where processing conditions are not optimal.

PROCEDURE

A questionnaire form was developed to be completed at each facility studied. Information concerning the age of the processing chemicals and how often chemicals are changed, temperature, processing time, type of film used, and type of processing system is requested.

Step-wedge images were made on films with a single reproducible dental x-ray unit for dental facilities and with a single sensitometer for medical facilities. These pre-exposed films were then processed as a normal patient radiograph would be processed at each medical and dental facility.

When the processed test films and the questionnaires were returned from the facilities, the films were compared to the control films by use of a densitometer on the step-wedge image. Control films consisted of pre-exposed films from the

same batch which had been processed under optimal film processing conditions.

Comparison of the test film from the facilities being surveyed to the corresponding control film allowed us to classify each facility into one of the following three groups: underdeveloped films, optimally developed films, or uncertain development.

For facilities submitting underdeveloped films, changes in processing techniques are recommended based on the questionnaire information and the test film results. New x-ray technique factors may also be recommended. A later study of the facility may document whether recommended changes were actually made. For facilities with optimally developed films, the facility is notified that its film processing technique appears to be achieving complete development of the film.

When it is uncertain whether the facility is achieving optimal film development, the facility is contacted to clarify information that has led to the uncertainty or to obtain additional information. A second test film and follow-up questionnaire may be sent to these facilities. The facility may be asked to process the second test film on the day before changing chemicals.

To compare each test film to the corresponding control film we measured the ratio of the optical density on the same step of the test film and the control film respectively, the density being corrected for base plus fog in each case. This density ratio appears to be a good indicator of completeness of development, may be regarded as a fraction of completion of development relative to the control film, is useful in comparing film processing conditions, and allows a simple method of tabulating test results from facilities.

RESULTS

It was found that density ratio was nearly constant over a wide range of step-wedge densities. The density ratio for any specific non-optimal processing conditions will vary somewhat depending on the film used.

The results that we have obtained to date indicate that the density ratio parameter is not sensitive to selection of technique factors used to expose the film. We have established separate control films for each batch of exposed films; however, it may be possible to use control films for more than one batch of test films.

Films and questionnaires were sent to 116 dental facilities and completed at 60 medical facilities by inspectors. Out of 96 dental facilities which returned the survey by mail 72 were determined to be underdeveloping their films, 7 had optimal film processing, and 17 had uncertain processing conditions which required follow-up surveys or phone calls to obtain additional information. Based on the completed processing questionnaires it was found that the facilities underdeveloping were usually not following the recommendations of the chemical manufacturer and/or those of the automatic processor manufacturer. Test films returned from three dental facilities were much darker than the control films. All three of these films were processed within one day after complete change of developer solution at each facility.

This survey method will detect serious film fogging problems which appear as increased density on the test film compared to the control film. Our facility visits have confirmed such film fogging problems noted on test films. Pre-exposed

test films are useful for carrying out darkroom fog tests at facilities during inspections, since they have a step-wedge image resulting in a range of normal radiographic densities when processed; however, when the procedure is used by a surveyor at a facility it is advisable to carry out a separate fog test using film from the facility. Film from the facility will detect fogging due to outdated film or film exposed to radiation at the facility that would not be detected with the pre-exposed film.

To compare D speed and E speed dental film we asked dental facilities to process test films of both types as they would process a normal dental radiograph. We then calculated the density ratio for each film type using separate D speed and E speed control films. The calculated density ratio for Kodak E speed film was greater than the calculated density ratio for Kodak D speed film unless the processing conditions were extremely poor (i.e., density ratio of D-speed film less than 0.5). For processing conditions that were nearly optimal for D-speed film (i.e. density ratio greater than 0.9 but less than 1.0) E speed test films sometimes indicated optimal processing conditions. Therefore, choice of E speed dental film to test processing conditions may not accurately determine optimal processing conditions for D speed film as noted above. These results indicate that many facilities with less than optimal processing could switch to E speed film without problems resulting from film processing and with little, if any, loss of contrast at the same selected kVp. E speed film appeared to be more sensitive to film fogging than D speed film.

For medical facilities we have tested several commonly used film types and so far have found no indication that testing with one type of film will lead to false conclusions about a processing system used for another film type. However, we have found that use of high contrast medical film, such a Kodak XRP film, gives less reproducible testing results than use of a medium speed medical film, such as Kodak XL film pre-exposed films.

CONCLUSIONS

The results of this screening procedure indicate it to be a useful and inexpensive method of determining film processing problems. The method is versatile in that it can be used while visiting facilities, or it may be used by sending the questionnaire and test films through the mail. When the questionnaire and films are sent through the mail and the results are presented to the facility by telephone or form letter, it is estimated that the personnel time expended on each facility averages approximately 30 minutes.

It is apparent from our results that the density ratio may vary with the film type used for testing. We have found that a convenient way to assess the magnitude of this problem is to use one film type as a standard and run pre-exposed films of other film types through the same processing systems. A density ratio is then calculated for each film type processed under identical conditions and a plot of density ratio of one film type vs density ratio of another film type may be made. Using this graph one may determine whether a particular type of film is appropriate as a test film depending on which film type(s) are used by the facility. This additional factor may be incorporated into the testing procedure. Although the density ratio results are somewhat dependent on the type of film used for testing the system, this method appears to detect serious film processing problems accurately when the test film type is not identical to the film type used at the facility.

The reproducibility problems that we have encountered with high contrast

medical film point out the need to check for reproducibility of results by processing several films through the same processing system before using a batch of pre-exposed films.

Film processing problems detected with this procedure may be grouped into two types. First is selection of inadequate film processing conditions that will not achieve complete film development even immediately after the chemicals have been changed; this is apparent when a test film processed very soon after the chemicals have been changed is not fully developed. Second is failure to replenish or change chemicals frequently enough to avoid depletion of chemicals; this is obvious when a test film processed just before the facility changes chemicals is not fully developed and a test film processed immediately after the chemicals have been changed is fully developed. Film processing problems may also be due to a combination of both of the above situations. We have found less film processing problems of the second type at medical facilities using automatic processors with automatic replenishers than at facilities using tanks or at dental facilities.

Using a density of greater than 2.0 for the control film area used to calculate the density ratio may have an advantage in that the higher density portion of the test films appears to be more sensitive to the second type of film processing problem noted, failure to change or replenish chemicals often enough. In this case the density ratio for high density areas of the control film tends to be lower than the density ratio for lower density areas on the control film.

We have used control films which are overexposed for dental facilities to aid in detecting "sight development" which results in the facility being classified as underdeveloping.

The overall results of this study indicate that dental film processing may be less than optimal in most facilities, including facilities that do not have skin entrance exposures above the accepted normal range. This method can be used in conjunction with other procedures to aid in decreasing patient exposure and to improve diagnostic information.

Some of the anticipated problems with this screening procedure have not been found once the survey was implemented. For example, there has been no evidence of latent image deterioration between the time of first use of pre-exposed films and the time of processing at subsequent facilities several months later. Test films were not processed until at least five days after the initial exposure of the film. Results have been consistent in that test films processed at facilities that are following proper processing techniques do match the control films. Conversely, when test films indicate a processing problem exists we have been able to find correct conditions to obtain a processed test film that matches the control film. The films have not been damaged in the mail.

While attempting to improve processing conditions at dental and medical facilities, the need to know specific conditions which give optimal results in a given system becomes apparent. Tabulations of data and the manufacturer's recommendations are essential to the effective use of this processing survey technique. Without these aids, the questionnaire answers have little meaning and it may require several test films and changes in processing conditions to achieve optimal film processing. Storage of these data and appropriate form letters on a computer system would allow creation of computer software to diagnose film processing problems based on questionnaire data and a density ratio.

MESURE SIMULTANEE DE LA QUALITE DE L'IMAGE ET DE LA DOSE EN RADIODIAGNOSTIC

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1. INTRODUCTION

La valeur globale d'un système radiologique dépend d'une part de la qualité de l'image obtenue, d'autre part de la dose délivrée au patient au cours de la procédure. La connaissance des caractéristiques de chaque élément de la chaîne radiologique ne permet pas en général de déterminer de manière précise le comportement du système, vu la forte interaction de l'ensemble des paramètres. En outre, le test de chaque élément serait très long et immobiliserait de manière inacceptable l'installation. Ainsi, une technique rapide de contrôle global du système radiologique s'avère très souhaitable.
Les méthodes classiques (1), consistant à apprécier visuellement la qualité des clichés effectués sur patient ou des radiographies de fantôme, ne sont pas satisfaisantes à cause de leur caractère subjectif. En outre, elles ne permettent pas en général de mettre en évidence de faibles différences de qualité et ainsi de choisir entre deux dispositifs (écran, film, etc.) qui ont des performances voisines.
L'amélioration de la qualité de l'image ne peut en outre être effectuée aux dépens de la dose délivrée au patient et la mesure simultanée de ces deux grandeurs s'avère nécessaire.
Dès 1980, un programme de mesure a été envisagé, dont l'objectif est la détermination, à partir de la prise d'un seul cliché, de la qualité de l'image et de la dose liées à une procédure radiologique. La première étape de ce programme a été consacrée à la mammographie; les méthodes et les résultats obtenus sont présentés dans ce travail. Les problèmes posés par l'extension de cette méthode à la radiographie du thorax sont discutés.

2. PARAMETRES REPRESENTANT LA QUALITE DE L'IMAGE ET LA DOSE DELIVREE AU PATIENT

Les trois facteurs intrinsèques de la qualité, à savoir la résolution, le contraste et le bruit de l'image, ont été appréhendés séparément. Afin d'obtenir une idée globale de la qualité de l'image, une grandeur unique appelée IQI (index de qualité d'image) a été déterminée; il s'agit du diamètre de la plus petite sphère d'aluminium que le système est capable de détecter, selon la théorie de Harris (2), avec un taux de faux négatifs de 2%.

La grandeur IQI est donnée par la relation suivante développée dans une précédente publication (3) :

$$IQI = 2\left(\sigma_h^2 + \frac{2G}{\sqrt{\pi}\,\gamma A}\right)^{\frac{1}{2}}$$

avec σ_h : la demi-largeur à mi-hauteur de la fonction de résolution spatiale.

G : coefficient de Selwyn.

γA : contraste global du système pour un objet d'aluminium d'épaisseur unitaire.

L'aluminium a été choisi comme matériel de référence à cause de ses propriétés d'absorption proches de celles des microcalcifications.
La mesure du risque radiologique peut être déterminée à partir de la répartition de la dose et de celle du tissu radiosensible à l'intérieur du sein. La variation de la dose est relativement importante et dépend de l'épaisseur du sein et de la qualité du rayonnement utilisé. Le but de la mesure étant la détermination de la qualité globale du système, une épaisseur moyenne du fantôme a été définie. La répartition du parenchyme est mal connue et la majeure partie de l'énergie incidente est atténuée dans le tissu; le rapport entre la dose absorbée dans le parenchyme et la dose surface étant assez constant dans le domaine de tension utilisé en mammographie, la mesure de la dose surface (D_s) représente un bon indicateur du risque radiologique; de plus, cette grandeur permet une comparaison aisée de nos résultats avec les données de la littérature.

3. METHODES EXPERIMENTALES

La mesure ayant pour but de donner une idée précise et dans une situation générale de la qualité d'une mammographie, le matériau utilisé doit être comparable au tissu dans une large plage de tension (20 à 50 kV). A cet effet, un fantôme contenant de l'eau a été élaboré, à l'intérieur duquel l'objet test de Kodak (ITO) a été placé, ceci en vue d'une première analyse visuelle de la qualité de l'image.

L'épaisseur totale du fantôme est de 4,9 cm. Le contraste du système est mesuré à l'aide d'une feuille d'aluminium de 0,2 mm. Une plaquette de cuivre rectifiée produit un saut de contraste, à partir duquel la grandeur σ_b est calculée. Le bruit est mesuré sur une plage d'épaisseur uniforme de l'objet. Les densités optiques sont lues sur un microdensitomètre (Joyce-Loebl, MX III CS) couplé à un miniordinateur. La dose est mesurée à l'aide de dosimètres thermoluminescents placés à différentes profondeurs dans le fantôme et permettant un contrôle de l'allure de la répartition de la dose en fonction de la profondeur.
Les valeurs obtenues sur une installation de référence (Siemens, Mammomat) et pour quelques tensions sont représentées à la figure 1. Ces résultats donnent une idée de la valeur de notre méthode, dont la reproductibilité, testée dans les conditions normales d'utilisation, est de l'ordre de 3%.

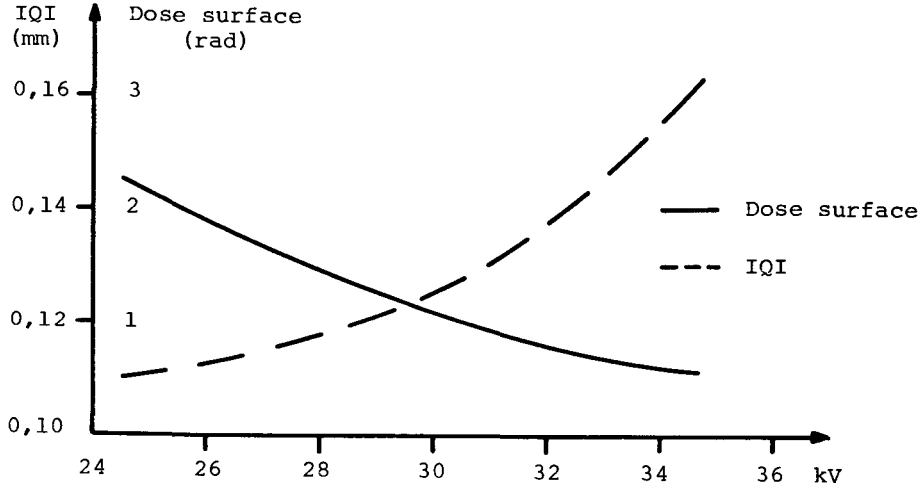

Figure 1. Valeurs obtenues sur une installation de référence

4. RESULTATS OBTENUS

Le fantôme, chargé de dosimètres, est envoyé aux services qui en font la demande. Après une prise de cliché unique, il nous est renvoyé pour analyse. Au cours de 2 années d'exploitation, environ 40 installation ont été testées. La répartition des résultats obtenus est donnée à la figure 2. Pour chaque mesure, un commentaire indiquant les possibilités d'amélioration du système est envoyé au radiologue utilisateur. Dans de nombreux cas, des changements de technique et de matériel ont été effectués permettant une amélioration de la qualité ou une réduction de la dose.

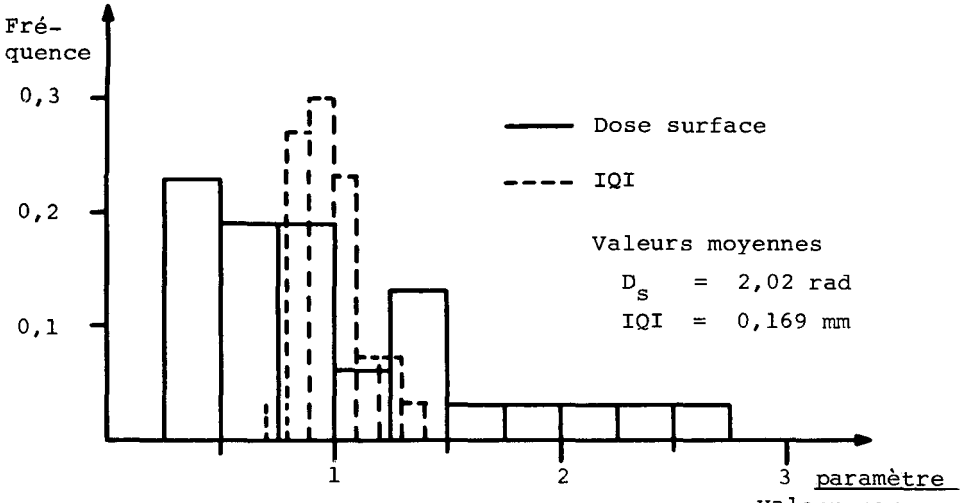

Figure 2. Répartition des paramètres mesurés

5. CONCLUSIONS

Le test effectué est d'une mise en oeuvre facile dans les services de radiologie et permet la saisie globale de la qualité du système radiologique. Il assure en outre une appréciation objective de la qualité de l'image et permet de diagnostiquer dans une certaine mesure les origines des mauvais résultats. L'extension de cette méthode à la radiographie du thorax est à l'étude actuellement. Deux problèmes majeurs se posent : au niveau théorique un nouvel index de qualité doit être défini; sur la plan pratique le volume et le poids d'un fantôme équivalent, sur une large plage d'énergie, à un thorax, représente une difficulté sérieuse lors de la mise en exploitation.

BIBLIOGRAPHIE

1. E.A. Sickles in "Reduced Dose Mammography", ed. W.W. Logan and E.P. Muntz, Masson Publ. (1979).

2. J.L. Harris, J. Opt. Soc. Am. 54, 606 (1964).

3. Y. Pochon, C. Depeursinge, C. Hessler, S. Raimondi, J.F. Valley in Processing of SPIE 347 (1982) Application of Optical Instrummentation in Medicine X, 238.

ESTIMATE OF LENS DOSES TO PATIENTS UNDERGOING RADIOTHERAPY TREATMENTS

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The correct estimation of the dose to the neighbouring organs is essential in order to determine the tumor dose in the application of the radiation therapy. It is sufficent to consider that some critical organs are able to tolerate such low doses that the evalutation of the risk-benefict ratio can be very difficult.

Generally less attention is paied to the neighbouring organs at a distance of more than 2 cm apart from the geometrical edge of the irradiation beam.

In this paper we analize dosimetric measurements performed at various distances from the irradiation field, in order to estimate the lens dose in the radiation therapy by Co 60 and the linear accelerator RX 9MV for neck and head tumors and for Hodgkin's desease both in female Rando phantom and patients.

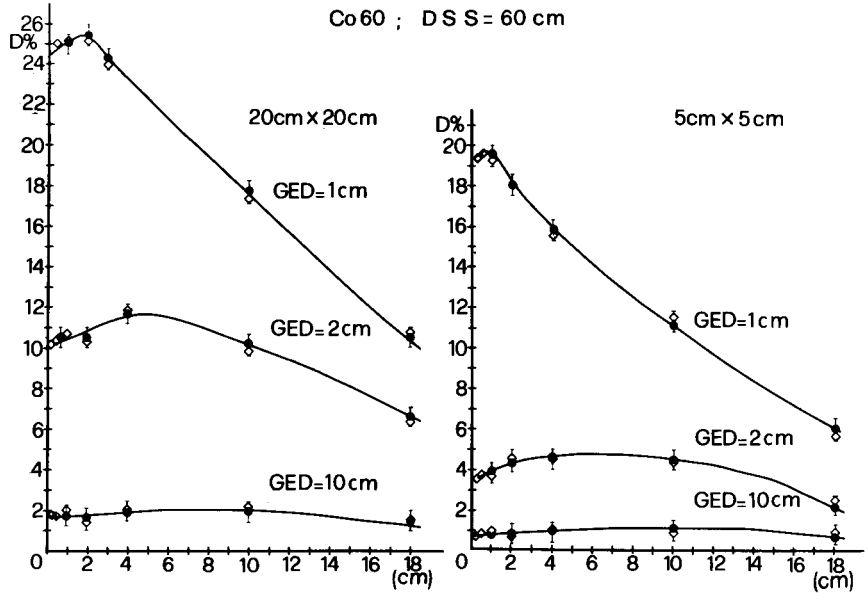

FIG. 1 - Co 60 Percent Dose measurements vs. the depth for different geometrical edge distance (GED).
● Ionization Chamber ; ◇ TLD.

The measurements were realized by vacuum glass encapsulated Victoreen Litium Fluoride TLD cilindrical dosimeters 1,2 cm height and 0,14 cm diameter. The readings of the TLD data were done by Victoreen mod. 2800. The TLD dosimeters were dose calibrated by exposition to the primary beam.

In order to control the method, we measured the esposure by a PTW Normal ionization chamber in a polystyrene phantom as a function of both the depth and the distance from the geometrical edge(GED) of the irradiation field.

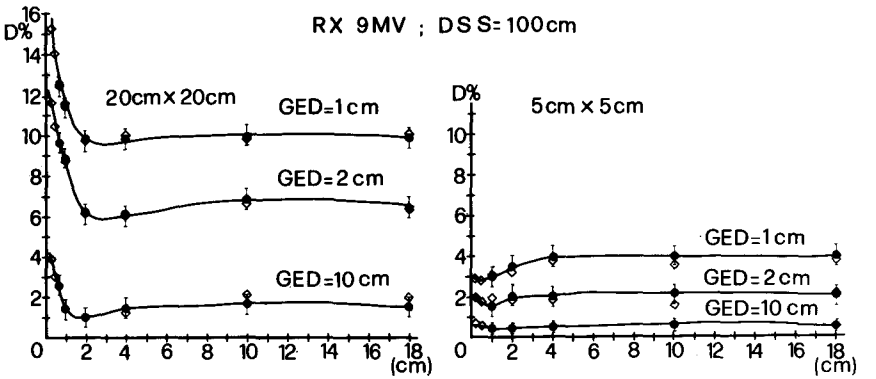

FIG. 2 - RX 9MV Percent Dose measurements vs. the depth for different edge distance (GED).
● Ionization Chamber ; ◇ TLD

FIG. 1 and FIG. 2 reports the data of the doses obtained from both TLD and ionization chamber measurements, as a percent of the build-up dose, either from Co 60 or RX 9MV radiation. The percent dose is plotted against the depth in polystyrene phantom along different GED of two fields 5 cm x 5 cm and 20 cm x 20 cm in size at the phantom surface. The good agreement between the TLD and ionimetric measurements allowed us to assume that the TLD dosimeters can be utilized for a depth to 18 cm and for a GED to 15 cm.

Furthemore we measured in polystyrene phantom the percent dose by TLD dosimeters after positioning lead shields on the collimators of the primary beam, in order to know the contribution of the transmitted radiation to the dose.

These results reported in FIG. 3 (dotted line) for the 2 cm GED and 20 cm x 20 cm irradiation field as a function of the depth, are compared with those previously shown in FIG. 1 and FIG. 2. The theird line was obtained by difference. It appears clear that in the RX 9MV applications the contribution to the dose down to a depth of about

1 cm is essentially caused by the radiation trasmitted through the collimators. The same consideration is not true for the Co 60 radiation. In this case the radiation does not come from a point source and, therefore, the penumbra effect can not be totally eliminated by the lead shields.

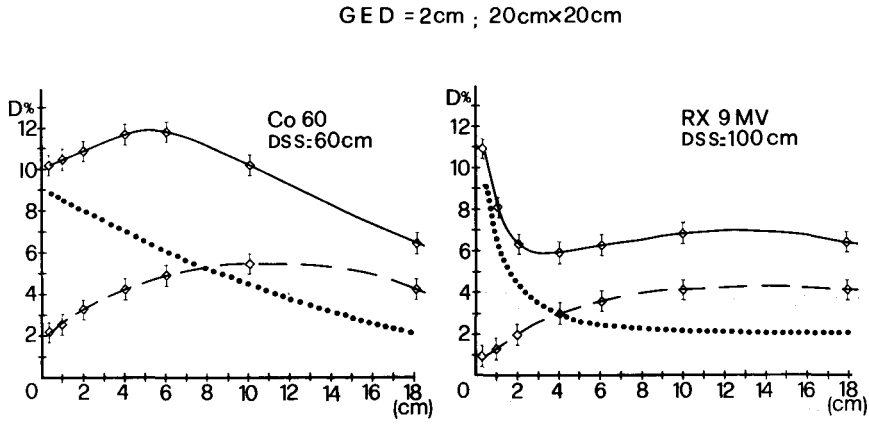

FIG. 3 - Co 60 and RX 9MV Percent Dose measurements by TLD:
____ without lead shields; - - - with lead shields;
.... difference behaviour.

Table reports the dosimetric data obtained from measurements

Type of treatment	Tumor dose (Gy)	Lens dose (Gy) Co 60	RX 9MV
Head fields	45÷70	1,80÷15,00	0,45÷4,00
Neck fields	60	2,8	2,6
Hodgkin's	40	2,1	2,4

in female Rando phantom. The data are summarized from a set of measurements carried out for differents parallel opposing fields of the neck and head tumors and of the Hodgkin's disease, without the lead shields.

It is cleare that the doses to the lens can reach very high values.

In conclusion we want to point out first that the doses to the lens can by reduced by the use of the lead shields in the anterior irradiations. Furthermore calculations (B.Keller et al,1974) of scattered radiation distributions seem to be very hard down to a depth of about 2 cm, beacause the complexity of the different contributions. On the other hand, it is very important to remember (M.W. Charles et al, 1975) that the minimum depth of the epitelial cells of the lens, which are most at the risk from radiation, is about 2,3 mm.

References

M.W. Charles, Nicholas Brown. Dimension of the human eye relevant to radiation protection.Phys. Med. Biol. 20, 202-218, 1975.

B. Keller, C. Mathewson, P. Rubin. Scattered radiation dosage as a function of X-ray energy. Radiology 111, 447-449, 1974.

VERMINDERUNG DER STRAHLENEXPOSITION DES PATIENTEN IN DER RÖNTGENDIAGNOSTIK DURCH OPTIMIERTE FILTERUNG.

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Es ist eine seit langem bekannte und akzeptierte Tatsache, daß in der Röntgendiagnostik durch Filterung der Primärstrahlung die Exposition des Patienten ohne merkliche Einbußen für die Bildqualität deutlich verringert werden kann. Die Filterung schwächt vorwiegend die niederenergetischen Anteile des Röntgenspektrums, die wegen ihrer geringen Durchdringungsfähigkeit unnötigerweise die Oberflächendosis erhöhen ohne zum Aufbau des Röntgenbildes beizutragen.

Mindestwerte der Filterung für Röntgendiagnostik- Strahler wurden daher schon vor längerer Zeit festgelegt (ICRP 16, DIN 6811). Diese Filterwerte werden traditionell in Millimeter Aluminiumgleichwert angegeben. Das ist wohl der Grund, weshalb meist als Zusatzfiltermaterial auch Aluminium verwendet wird, obwohl seit längerer Zeit bekannt ist, daß Materialien mit höherer Ordnungszahl günstigere Filtereigenschaften besitzen (BÖSCHE, BÄUML). Bei Röntgenröhren konventioneller Bauart ist ein wesentlicher Teil der Gesamtfilterung durch die Eigenfilterung, d.h. durch die Materialien im Bereich des Röhrenfensters (Glas, Öl, Verdrängungsgefäß, Lichtvisierspiegel) vorgegeben. Eine Optimierung der Filterung konnte sich daher nur auf das Zusatzfilter beziehen. Bei modernen Röhrenkonstruktionen jedoch ist es möglich, das Röhrenfenster aus Beryllium zu fertigen, das eine vernachlässigbar geringe Eigenfilterung besitzt. Solche Röntgenröhren erlauben durch Wahl eines optimal geeigneten Filtermaterials eine weitaus deutlichere Verringerung der Oberflächendosis ohne Beeinträchtigung der Bildqualität.

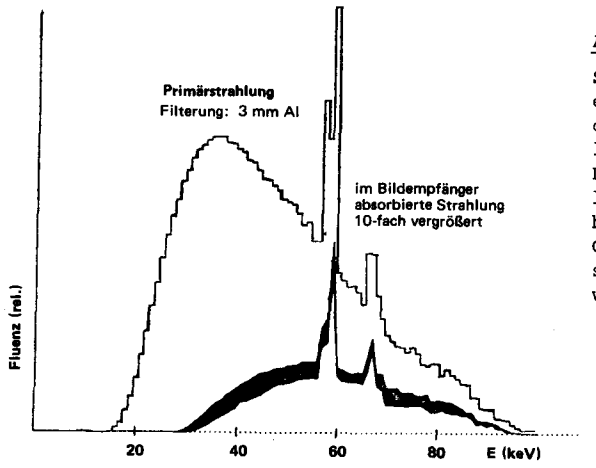

Abb. 1:

Spektrale Verteilung einer in der Röntgendiagnostik üblichen Strahlenqualität.
Darunter: Spektrale Verteilung der Strahlung, die hinter 15 cm Wasser bzw. Gewebe in einem Paar Verstärkerfolien absorbiert wird.

Dicke und Material der optimalen Filterung werden durch die Dicke des untersuchten Objektes bestimmt. Die Optimierung der Filterung wird an zwei Fällen demonstriert: Abdomenuntersuchung mit 15 cm Gewebedicke und Mammographie mit etwa 5 cm Gewebedicke.

In Abbildung 1 ist die spektrale Verteilung der Röntgenstrahlung bei einer Röhrenspannung von etwa 100 kV und einer Filterung von 3 mm Aluminium dargestellt.
Der untere Teil der Darstellung zeigt in 10-fach überhöhtem Maßstab den Anteil der Röntgenquanten, der hinter einem 15 cm dicken Wasserphantom am Bildempfänger ankommt und in einem Paar normal verstärkender $CaWO_4$-Folien absorbiert wird. Der dunkel angelegte Bereich stellt den relativen Anteil der Strahlung dar, der bei der jeweiligen Energie der Röntgenquanten absorbiert, zu einem bestimmten Prozentsatz in Licht umgewandelt wird und den Film schwärzt. Die Messung wurde im engen Strahlenbündel durchgeführt, so daß der Einfluß der Streustrahlung keine Rolle spielt.

Das ideale Filtermaterial sollte die niederenergetische Strahlung sehr stark, die höherenergetische Strahlung aber möglichst wenig schwächen. Am besten wird diese Forderung von Eisen und den im Periodischen System benachbarten Elementen erfüllt.

Zum Vergleich der Filtereigenschaften von Al und Fe sind in Abb. 2 die Dosisspektren von mit Al und Fe gefilterter Strahlung dargestellt. Die Dicken wurden so gewählt, daß die Dosiswerte bei 60 kV übereinstimmen. Unterhalb dieser Energie ist Al durchlässiger für Röntgenstrahlung als Fe, oberhalb aber zeigt Fe eine etwas größere Transparenz als Al.

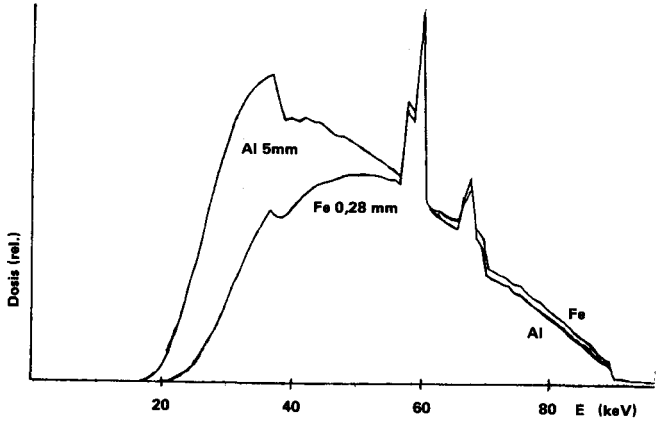

Abb. 2:

Unterschiedliche Filtereigenschaften von Aluminium und Eisen. Dicke des Eisenfilters wurde so gewählt, daß gleiche Schwächung bei 60 kV erreicht wird.

Wenn man die Eisen-gefilterte Strahlung in Abb. 2 mit der aus Abb. 1 ermittelten Ansprechwahrscheinlichkeit des Bildempfängers vergleicht, so ergibt sich, daß für Untersuchungen des Abdomen eine Filterung mit 0,25 mm bis 0,3 mm Eisen eine vorteilhafte Alternative zur üblichen Aluminiumfilterung darstellt: Die Strahlenexposition des Patienten wird deutlich verringert, die Bildqualität wird nicht merklich vermindert. Die Belichtungszeit wird, wegen der besseren Durchlässigkeit des Eisens im höheren Energiebereich, kaum verlängert.

In der Mammographie, bei der wegen der Darstellung von Gewebestrukturen mit sehr geringen Dichteunterschieden mit niederenergetischer Röntgenstrahlung gearbeitet werden muß, werden meist Röntgenröhren mit Molybdän-Anode und 0,03 mm Mo-Filter verwendet.
Diese Technik liefert gute Ergebnisse, wenn auch mit relativ hoher Strahlenexposition. Auch hier ließe sich die Belastung der Patientin verringern, wenn man Röhren mit Wolfram-Anoden verwendet, deren Strahlung je nach Dicke des zu durchstrahlenden Gewebes mit Filtern aus Elementen von Molybdän bis Zinn modifiziert wird. Diese Materialien besitzen Absorptionskanten für Photonen im Energiebereich von 20,0 keV bis 29,2 keV. Unterhalb der Absorptionskante wird die Strahlung gut durchgelassen, oberhalb dagegen stark geschwächt, so daß sich zwar keine nahezu monochromatische Energieverteilung wie bei der Molybdänröhre mit Mo-Filter, wohl aber eine recht schmale Energieverteilung ergibt. In einer kürzlich erschienenen Untersuchung (BEAMAN) wurde gezeigt, daß bei Verwendung einer Wolfram-Anoden-Röhre mit Palladium-Filterung ebenso gute diagnostische Ergebnisse erzielt werden wie mit der Molybdän- Röhre, jedoch mit wesentlich geringerer Dosis.
Durch die Wahl der Filterdicke läßt sich die spektrale Verteilung der Primärstrahlung an die Verteilung der bildgebenden Strahlung hinter dem Untersuchungsobjekt anpassen. Durch die Wahl eines Filtermaterials mit zunehmend höherer Absorptionskante für zunehmende Gewebedicke ließe sich wahrscheinlich die Strahlenexposition ohne Verzicht auf diagnostische Information zusätzlich verringern.

Die Kontrastwiedergabe wird zwar bei zunehmender mittlerer Photonenenergie schlechter, doch scheint die Verwendung eines Silber-Filters in Verbindung mit einer Wolframanodenröhre niedrige Strahlenexposition mit akzeptabler Bildqualität zu ermöglichen. Wegen der höheren Belastbarkeit einer Röhre mit W-Anode sind außerdem kürzere Belichtungszeiten und daher weniger Bewegungsunschärfen zu erwarten.

Durch optimale Wahl des Materials und der Dicke der in der Röntgendiagnostik verwendeten Filterung kann die Strahlenexposition des Patienten minimiert werden.
Es wäre denkbar, daß bei modernen Gerätekonstruktionen ein automatischer Filterwechsler vorgesehen wird, mit dem bei Anwahl einer Automatik-Taste zugleich die Röhrenspannung vorgewählt und das optimal geeignete Filter in den Strah-

lengang eingefahren wird.

Literatur:

S. BEAMAN, S.C. LILLICRAP, J.L. PRICE:
Tungsten anode tubes with K-edge filters for mammography
Britisch Journal of Radiology 56 721-727 (1983)

A. BÄUML
Verbesserter Strahlenschutz des Patienten in der Röntgendiagnostik durch optimal geeignete Filtermaterialien
Bundesgesundheitsblatt 24 Nr. 4/5 55-59 (1981)

H. BÖSCHE, W. FRICK
Vorteile der Strahlenfilterung mit Eisen im Spannungsbereich der Röntgendiagnostik
Röfo 96 136-140 (1962)

DIN 6811 Medizinische Röntgeneinrichtungen bis 300 kV. Strahlenschutzregeln für die Herstellung (1972)

ICRP Publication 26
Radiation protection of the patient in x-ray diagnosis
Pergamon Press London (1969)

QUANTITATIVE X-RAY FLUORESCENCE MEASUREMENT OF THE STABLE IODINE IN THE THYROID GLAND

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In-vivo detection of the thyroidal iodine content (TIC) by X-ray fluorescence (XRF) was first suggested (1) in 1968. This method has not yet been applied clinically, mainly because of its limited accuracy, about 50% in existing experimental systems (2,3), while it is expected (4) that various thyroid conditions can be differentiated only when the estimation error does not exceed 20-30%.

The low accuracy of the existing experimental systems stems mainly from the geometrical differences which naturally exist between the calibration phantom and the various patients. This results in an uncertainty in the actual distance of the thyroid from the XRF unit and also in the thickness of the tissue covering the thyroid, thus introducing errors in the estimation of absorption and dispersion of the radiation involved.

In order to obtain an improved estimation of the effective depth of the thyroid we examined two methods.

a) The energy difference between the iodine K_α and K_β lines (28.5 and 32.3 keV, respectively) results in a difference of 7% in their absorption per cm tissue (5). Consequently it is possible to obtain the effective depth of the thyroid from the intensities ratio K_α/K_β.

The effective depth T is given by Eq. (1)

$$T = [\ln I(K_\alpha) - \ln I(K_\beta) - \ln(A)]/[\mu(K_\beta) - \mu(K_\alpha)] \qquad (1)$$

where $I(K_\alpha)$ and $I(K_\beta)$ are the intensities of the K_α and K_β lines, $\mu(K_\alpha)$ and $\mu(K_\beta)$ are the respective attenuation coefficients and A is the unattenuated K_α/K_β ratio (6).

In phantom measurements that we conducted, simulating normal human conditions, the signal-to-background ratio was 1:5 at K_α and 1:27 at K_β. This relatively high background is due to the strong scattering of the Am radiation by the neck. Thus, if an error of less than 20% is called for, it can be shown by conventional error propagation calculations that the number of counts in the K_α region must exceed 10^6. This would result in a dose of about 2 rads to the patient (as measured by a LiF crystal dosimeter attached to the front of the phantom).

b) The XRF intensity is naturally distance-dependent. Hence, the effective depth of the thyroid can be obtained by comparing the XRF intensity of measurements at two distances.

We developed a computer simulation of our system in which the thyroid lobes had either a cylindrical or an ellipsoidal shape, and their dimensions and separation were the variable parameters. Volume

varied from 15 to 40 cm^3. (The normal volume is 20 cm^3 (4)). In this simulation we assume a $1/R^2$ radiation dispersion from each point of the Am source to each point of the thyroid, and from each thyroid point to each point of the Si(Li) detector.

The distance dependence is expressed in Eq, (2)

$$\frac{I_1}{I_2} = \left(\frac{D_2 + T + C}{D_1 + T + C}\right)^p \qquad (2)$$

where I_1 and I_2 are the XRF intensities at the distances D_1 and D_2, respectively, the constant C has the value 1.2, p is taken as 3.54 and T is the effective depth of the thyroid.

Equation (2) is valid under the following conditions:

i) 6cm < D_1 < D_2 < 10 cm where D_1 and D_2 are the neck-to-XRF unit distances.

ii) $D_2 - D_1$ < 2 cm.

iii) the diameter of the annular source (Fig. 1) is 4-6 cm.

Under these conditions we obtained a ∿20% error (after depth correction) in TIC measurements on a phantom (Fig. 1), with ∿ 10^4 counts in the K_α region. The dose to the patient is ∿ 20 mrad (measured by a TLD dosimeter).

The system is presently being applied to clinical measurements and the results will be reported elsewhere.

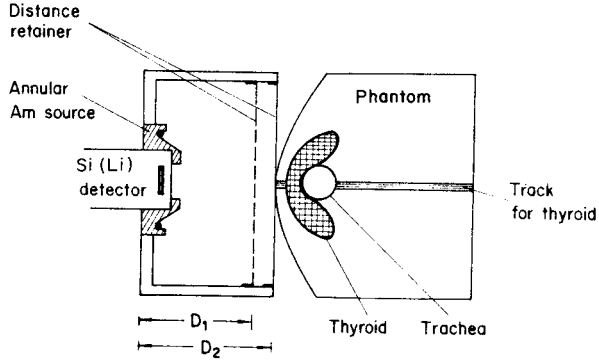

Fig. 1. XRF system for TIC measurements: The XRF unit consists of a 5 cm diam, 40 mCi ^{241}Am source and a 200 mm^2 area, 5 mm depth Si(Li) detector. The phantom consists of a polyethylene exterior and is water filled. The "thyroid" slides on a track for depth variation.

References:

1. P.B. Hoffer, W.B. Jones, R.B. Crawford, R. Beck and A. Gottschalk, Radiology 90, 342 (1968).
2. D.A. Gollnick and M.A. Greenfield, Radiology 126, 197 (1978).
3. K. Imamura, Y. Sasaki, N. Sekita, K. Someya and M. Fujii, Invest. Rad. 14, 316 (1979).
4. S. Silver, Radioactive Isotopes in Medicine and Biology (Lea and Febiger, Philadelphia, 1962), Vol. II, p. 19.
5. Handbook of Radiological Health, U.S. Department of Health Education and Welfare, 1970, p. 138.
6. M. Margaliot, Y. Eisen, T. Schlesinger, S. Friedland and E. Lubin, Trans. Nucl. Soc. Israel, Vol. 10, 1982, p. 201.

PREVISIONS DES CONSEQUENCES SANITAIRES D'UN REJET RADIOACTIF ATMOSPHERIQUE DANS UNE REGION A OROGRAPHE COMPLEXE

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Nous présentons, dans une première partie, la méthodologie mise au point pour établir les prévisions, en termes de conséquences sanitaires d'un rejet radioactif dans l'atmosphère d'une région à orographie complexe, et dans une seconde partie le support technique nécessaire à la mise en oeuvre de cette méthodologie.

PREVISION DU TRANSFERT ATMOSPHERIQUE

Le modèle de transfert atmosphérique qui a été développé pour figurer dans cette méthodologie est tri-dimensionnel, ce qui permet de prendre en compte les caractéristiques du relief [1].

Le premier modèle (modèle météorologique) permet de déterminer la courantologie et l'hygrométrie atmosphériques régionales à partir de la résolution des équations de l'hydrodynamique, dont les conditions initiales sont déterminées à partir de compagnes météorologiques régionales, d'études statistiques des types de situations météorologiques et des paramètres météorologiques au moment du rejet.

Le second modèle (modèle de transfert atmosphérique) permet de calculer, à la suite du précédent, les concentrations de radioéléments et les dépôts au sol. Ces concentrations sont les résultats de la résolution de l'équation de la diffusion-convection par la méthode de Monte Carlo [2]. Cette équation permet de décrire le transfert d'aérosols polydispersés.

PREVISIONS DES CONSEQUENCES SANITAIRES

La seconde partie de la méthodologie a pour objectif de faire une estimation des doses aux individus de la population, aux principaux organes et à l'organisme entier ; ceci afin de définir les mesures de protection à appliquer dans les zones concernées.

Ces doses sont calculées pour chaque radionucléides rejeté, puis sommées, pour l'individu du groupe critique. Les différentes doses calculées sont les suivantes :

a) Doses par irradiation externe γ due à la présence des radionucléides dans l'atmosphère.

b) Doses par irradiation externe γ due au dépôt des aérosols.

c) Doses par irradiation interne due à l'inhalation, N jours après l'inhalation.

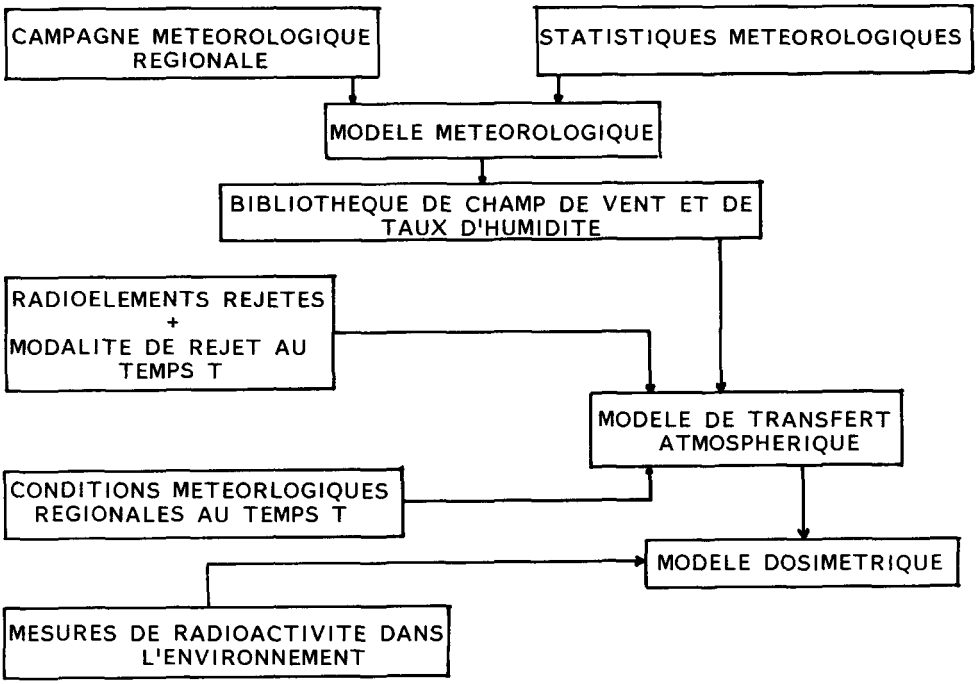
Ces doses permettent de prévoir les conséquences radiopathologiques éventuelles dans le délai d'un an après l'accident, pour les principaux organes (thyroïde, poumon, moelle osseuse).

Ces doses sont ensuite pondérées par le type d'habitat, rural ou urbain, et les habitudes sociales des populations. C'est-à-dire :

- la filtration des aérosols suivant le système de ventilation des habitations,
- l'atténuation des rayonnements γ suivant le type de construction,
- le taux d'occupation des habitations.

SUPPORT TECHNIQUE DE LA METHODOLOGIE

Le support technique de notre méthodologie est décrite par l'organigramme ci-cessous :



Dans une première partie les campagnes météorologiques effectuées par la Météorologie Nationale [3] et l'étude statistique des relevés des stations régionales permettent de préparer les données nécessaires pour calculer les champs de vent tri-dimensionnels relatifs aux diverses situations météorologiques statistiquement prépondérantes sur la région. Ces situations couvrent des périodes de 24 heures ; elles sont représentées par 8 champs de vent et 8 champs d'humidité, considérés comme stable durant 3 heures [4].

Dans une seconde partie, en cas d'alerte, un calculateur autonome connecté aux Services de la Météorologie Nationale peut :

- recevoir les conditions météorologiques régionales
- sélectionner les situations météorologiques archivées correspondantes
- acquérir la composition du terme source et les modalités de rejet
- calculer les transferts atmosphériques
- calculer les estimations des doses aux individus de la population concernée.

Cette méthodologie a été mise en oeuvre pour mieux suivre le nuage de gaz et d'aérosols radioactifs qui pourraient être émis lors d'un hypothétique accident survenant sur un réacteur situé dans une région dont l'orographie est complexe. A titre d'exemple, on montre quelques éléments de l'étude effectuée dans la moyenne vallée du Rhône, ou sont implantés, 5 tranches de réacteurs sur les deux sites de Bugey et de Creys-Malville.

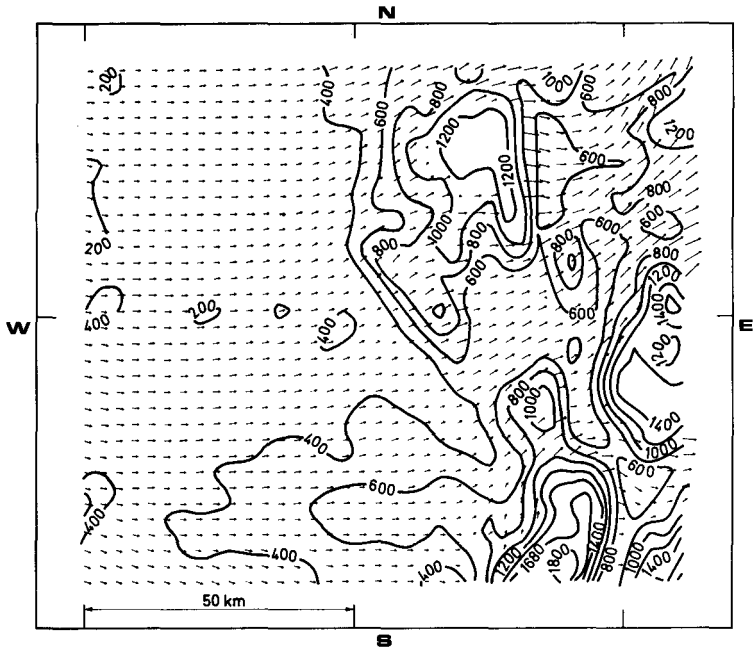


Figure 1

La figure 1 montre une coupe horizontale du champ de vent au-dessus des 2 sites à 900 mètres d'altitude.

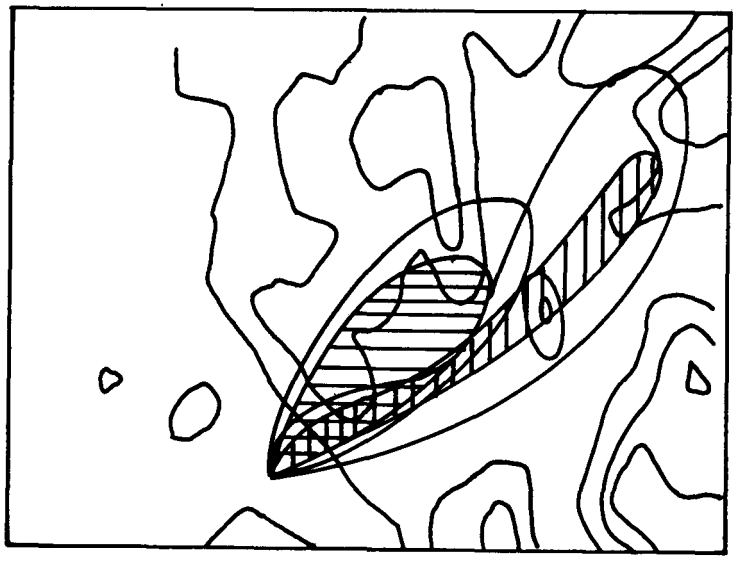


Figure 2

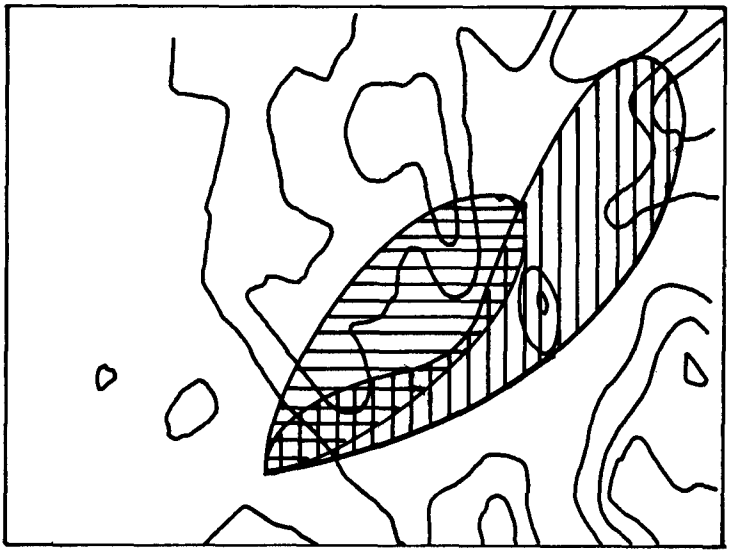


Figure 3

Sur les figures 2 et 3, les courbes délimitant les surfaces à hachurages verticaux sont les courbes de concentrations atmosphériques maximum respectivement après 9 heures et 14 heures de rejet. Ces courbes ont été établies à partir de la méthodologie précédemment décrite. Sur ces mêmes figures, les courbes délimitant les surfaces à hachurages horizontales sont les isoconcentrations maximum au sol, respectivement après 9 heures et 14 heures de rejet, établies par un modèle de type gaussien avec pour vecteur de convection, un vecteur correspondant à l'intensité et à la direction du vent mesurées au point de rejet.

Le faible recouvrement des surfaces présentées montre bien tout l'intérêt du développement d'un modèle régional, parallèlement au développement d'un modèle local.

REFERENCES

[1] BLONDIN C., THERRY G.
Analysis of Particles trajectoires during a land sea breeze cycle using two-dimensional numerical meso-scale models
Proceeding du 12ème ITM. PALO ALTO (USA), Août 1981

[2] ROBEAU D., BLONDIN C. and al.
Atmospheric Transfert Model for Radiological Emergency Preparedness for Complex Terrain
Proceeding fo the International Meeting on Thermal Nuclear Reactor Safety, CHICAGO (USA), August 1982 (Vol. 2 page 803)

[3] BLONDIN C.
Projet CYBELE. Exploitation de la Campagne "Vallée du Rhône 77"
Note technique de l'EERM n° 65

[4] BLONDIN C.
Un modèle de Meso-Echelle : conception - utilisation - développement
Note technique de l'EERM n° 416

EVACUATION AND REENTRY POLICY FOR CASES OF GROUND DEPOSITION FOLLOWING NUCLEAR ACCIDENTS

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INTRODUCTION

Evacuation due to ground deposition after a large nuclear accident has to be decided on the basis of dose rate at a given time or of integrated dose over a defined period. This paper describes the rationale underlying the intervention levels used in Argentina, which have the objective of preventing serious non-stochastic effects and of limiting individual risk. The paper also discusses the basis of the levels of remaining ground contamination at which reentry would be permited, used in Argentina.

It should be stressed that, as it is the case in all inverventions levels, the overriding principle is that the countermeasure should reduce risk: the risk avoided by the countermeasure should be larger than the risk of the countermeasure itself. The intervention levels, therefore, should be taken as indication for action, subject to judgement based on that overriding principle.

INTERVENTION LEVEL FOR EVACUATION

The main objective of evacuation is to prevent accumulation of doses leading to serious non-stochastic effects or to "unacceptably high" risk of stochastic effects. As non-stochastic effects depend not only on dose but also on the time distribution of dose, the intervention level must reflect the fact that exposure from ground deposition is a function of time.

Several expressions have been proposed to relate the "instantaneous equivalent dose" with an actual dose distribution giving the same magnitude of a non-stochastic effect. A very useful expression for the case of dose rate varying with time is the Walinder modification of the Kirk formula(1)

$$D_e(T) = \int_0^T 0.568\ \dot{D}(t)\ (T-t)^{-0.29}\ dt$$

where: $D_e(T)$ is the "instantaneous equivalent dose" received up to time T; $\dot{D}(t)$ is the dose rate at time t, and the time is expressed as the number of days. In the case of constant dose rate, the above equation results in the classical Kirk formula

$$D_e = 0.8\ \dot{D}\ T^{0.71}.$$

The time evolution of the dose rate caused by ground contamination will depend on the radionuclide composition released in the accident, but usually for large reactor accidents it could be represented by a power function of the type

$$\dot{D}(t) = D(1)t^{-n},$$

where n is a constant and the time, again, is expressed as a number of units.

For radionuclide compositions considered to be representative of the postulated accidents on which the Argentine emergency plans are based, the value of n can be approximated by 0.55 and, if the time is expressed as number of days, D(1) is close to 90% of D(0). In this case, the Walinder modification of the Kirk formula predicts the following evolution of the accumulated "instantaneous equivalent dose":

TIME	1/4 DAY	1 DAY	1 WEEK	1 MONTH	1 YEAR
D_e	1	3	10.5	24	45

where the dose is normalized to the value of one at six hours. From the above time evolution, it is clear that accumulated actual doses of 0.1 Gy in 1/4 day will correspond to "instantaneous equivalent doses" of about 0.3 Gy in a day and more than 1 Gy in a week, entering from that time in a region of doses leading to serious non-stochastic damage. On the other hand, accumulated actual doses of 0.1 Gy in a day will correspond to D_e of somewhat more than 0.3 Gy in a week and only reach the 1 Gy level in many months.

On this basis, the intervention levels adopted are the following: for projected doses exceeding 0.1 Gy in six hours evacuation is indicated for all involucrated individuals, while for smaller doses but exceeding 0.1 Gy in 24 hours evacuation is indicated for those which are "easy" to evacuate (namely not involving particular evacuation risks). The situation is reevaluated for the remaining individuals at a later time.

In both cases, consideration of the stochastic risk also underlines the intervention levels. A whole-body dose of about 0.1-0.3 Gy corresponds to a stochastic risk (2) of a few times 10^{-3}, which is deemed to justify the countermeasure unless such countermeasure involves larger risk by itself. However, if non-stochastic damage becomes possible after a few days (such as in the case of doses exceeding 0.1 Gy in six hours), then this fact becomes overriding.

RE-ENTRY POLICY

Long after evacuation, when the ground deposit has decayed sufficiently, it has moved down in the groun or has been washed away, a decision is required regarding areas where reentry would be allowed for permanent occupancy. While there is not still a formal policy adopted in Argentina a value of 0.05 Sv in a year of effective dose equivalent is used in planning and assessments.

Such dose would imply a stochastic risk over a lifetime of the order of a few percent. Cost-benefit considerations, with the value of 10^4 $/man Sv used in Argentina, indicates that permanent refusal of re-entrey authorization at doses lower than that indicated above would not be justified. It should be pointed out that effective dose equivalents of that order are due in some cases to natural exposure to indoor radon(3).

BIBLIOGRAPHY

(1) Walinder, G. Radiologisk Katastrofmedicin, F.O.A., Stockholm, 1981.

(2) ICRP Publication 26. 3. ed. Oxford, Pergamon Press, Annals of the ICRP v. 1: N° 3, 1977.

(3) United Nations Scientific Committee on the Effects of Atomic Radiation. 1982 Report to the General Assembly, with annexes. New York, United Nations, 1982. 773 p.

ASSESSING THE RADIOLOGICAL CONSEQUENCES OF SURFACE CONTAMINATION IN URBAN AREAS - AREAS OF UNCERTAINTY AND THEIR RESOLUTION

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INTRODUCTION

In the preparation of contingency plans for accidents involving releases of radioactive materials to the atmosphere it is necessary to have an appreciation of the likely effects of any countermeasures which may be employed. The effects to be considered are the radiation dose saved by, and the social economic costs of, the implementation of countermeasures. This type of basic information is also necessary for predictive assessments of accident consequences. The basic data necessary for the assessment of the consequences of land contamination by long-lived radionuclides are incomplete. The deposition of radionuclides from atmosphere and their retention on surfaces in urban areas are not well understood; neither is the effectiveness of large scale decontamination techniques well-known for other than deposits of large insoluble particles. In this paper a summary is presented of the results of a review study into the available data on these subjects[1]. Areas of data shortage are identified and the outline of a research programme to remedy some of these data deficiencies is presented.

POTENTIAL SOURCES OF LONG-LIVED ENVIRONMENTAL CONTAMINATION

In any industrial, medical or research application involving the use of significant quantities of radioactive materials there exists a small but finite potential for an accident to occur which may lead to the dispersal of radionuclides in the environment. Potential accidents at nuclear reactors, and particularly water-cooled reactors, have received considerably more attention than many other types of accident. This is, in part, due to the increasingly large numbers of such reactors in the world, some sited close to large population centres and because of the existence, in a reactor system, of mechanisms which under fault conditions can lead to the release and environmental dispersal of significant quantities of radioactive material albeit with very low probabilities of occurrence. For many other types of installation it is more difficult to conceive similar large scale releases largely due to the absence of a potential energy source which could disrupt the protective enclosures provided.

Where mixtures of radionuclides are involved in an accidental release, for example, one involving nuclear fuel, it is generally predicted that the largest fractions of released nuclides will be of those which are volatile in nature, for example, isotopes of iodine and caesium. The volatility of caesium and the long-lived nature of its radioisotopes is likely to result in their constituting the major long-term environmental problem in the majority of accidental release scenarios which can be envisaged from nuclear installations. This has been demonstrated in an analysis of exposure pathways to man from ground deposits of radionuclides released in representative hypothetical reactor accident sequences[2] The radiation dose to man from external irradiation and from ingestion of foodstuffs grown on the contaminated soil is seen in figure 1 to be dominated, at times beyond the first year, by the contributions of the isotopes of caesium. Attention is therefore focused in this review on the environmental problems associated with radio-caesium although much of the discussion is likely to be applicable to other environmental contaminants.

The most probable forms of release for caesium from thermal reactors are as a sub-micron aerosol formed from coagulated fume or as one of several fission products absorbed on to particles of iron or other materials present in the containment[1]. At distances of more than a few kilometres from the reactor the deposit is likely to be predominantly of small (< 10 μm) particles.

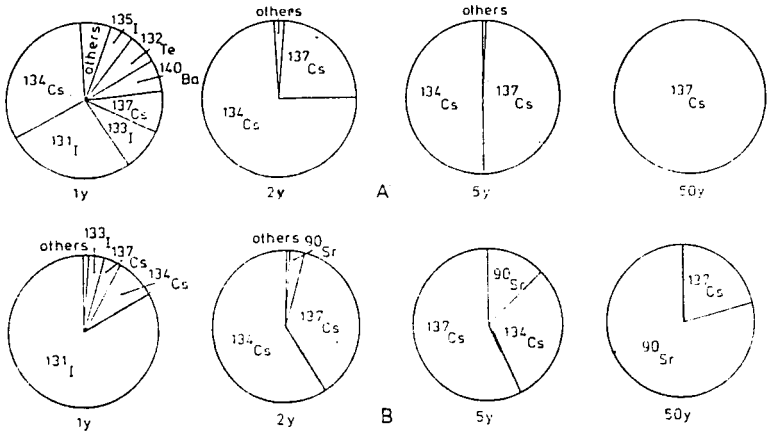

Figure 1 Fractions of annual effective dose equivalent due to (A) external irradiation and (B) ingestion

DEPOSITION OF PARTICLES ON VARIOUS SURFACE TYPES

The radiological consequences of an accidental release may be strongly influenced by the pattern of distribution of deposited material in relation to centres of population. Rainfall during the passage of the plume may cause higher areas of ground deposit, resulting in relatively higher radiation doses to people living in these areas. Uneven patterns of deposition may also occur in dry conditions mainly due to differences in the nature of the terrain over which the plume passes. Experimentally, the variations in deposition behaviour with particle size, wind velocity and surface roughness are measured using the deposition velocity, Vg.

Most measurements of Vg have been made over vegetated surfaces and, in particular, over grass. Few measurements have been made in urban areas and as a result, considerable uncertainty exists in the prediction of dry deposition in urban areas. From a theoretical viewpoint, vegetated surfaces have a greater surface area per unit ground area than smooth urban surfaces and therefore a greater deposition per unit area due to interception and impaction processes might be expected. This has been observed on bare soil surfaces and other smooth surfaces placed in rural areas for small particles of various materials. In an urban area the extra turbulence created by the presence of buildings may counteract this effect although it has been suggested, on the basis of caesium fall-out data that deposition velocities on walls, roofs and roads in urban areas is less by an order of magnitude than for rural vegetated surfaces[3].

The radiation dose to people inside buildings depends upon the level of surface deposits inside as well as outside the building. However, the available data on the subject strongly suggest that in the event of an atmospheric release of small particles there will be significantly lower deposition inside houses as compared with surfaces out of doors[1].

RUN-OFF IN URBAN AREAS

The scavenging action of rain passing through an airborne plume can be very effective in depositing a large fraction of its contents over a small area. If there is a significant amount of rainfall in a short time some of the deposited

material may be washed away from the place where it was deposited and in urban areas be transported via gulleys into drains and eventually into sewerage systems. For accidental releases of caesium to atmosphere, wash-out from the plume into urban areas is one of the scenarios which produces large environmental consequences[4]. Run-off is therefore potentially important as a mechanism for removing caesium from man's immediate environment, but is not considered in most accident consequence assessments due to the absence of suitable models.

SHIELDING FACTORS IN URBAN AREAS

Recent studies indicate that most external dose models used in generic calculations tend to underpredict the average amount of shielding provided by buildings in urban areas[5]. The shielding factors vary depending upon the nature of urban building structures and need to be evaluated on a national basis.

THE NATURAL WEATHERING OF SURFACE DEPOSITS

Weathering, due to the action of wind, rain and temperature change, can be effective in removing contaminants from environmental surfaces. The removal processes vary depending upon the nature of the surface and are very different for urban as compared with rural surfaces. A quantitative estimate of the rates of the weathering processes for the most important of the long-lived radionuclides would be valuable for the purposes of prediction in relation to accidental release consequence and in pre-planning.

There appear to have been no studies directed at observing the natural weathering of radionuclides on urban surfaces and what little information is available has to be gleaned from a few measurements made of fall-out caesium on building surfaces. Measurements on roofing materials and bricks indicate that only a few per cent of the total fall-out caesium deposit remains on these surfaces[1]. However, the rate of weathering shortly after deposition cannot be determined from these studies.

DECONTAMINATION OF ENVIRONMENTAL SURFACES

A substantial amount of information on some aspects of this subject became available as a result of studies conducted during the 1950's and 60's to provide information for use in planning recovery operations for the event of a nuclear attack[6]. Decontamination was attempted using comparatively simple and inexpensive techniques which could be applied on a large scale; the most commonly applied techniques were mechanical sweeping (roads and pavements), fire hosing, compressed air blowing and vacuum cleaning. Other more sophisticated and expensive techniques were also applied, for example, sand blasting and steam cleaning. Decontamination factors of 10 or more were readily achievable for most surfaces tested with the simpler techniques for particle sizes greater than about 75 μm diameter although, as might be expected, the efficiency was lower on rough surfaces. It was observed that the same techniques applied to surfaces contaminated with simulated fall-out in the 20-75 μm range were much less effective; decontamination factors of two or three were all that could be achieved. Only one study appears to have been reported in which the results may be directly applicable to the estimation of the likely effectiveness of decontamination techniques for small particle caesium deposits in urban areas[7]. Solutions of rubidium-86, as an analogue for caesium, were deposited on asphalt road surfaces by watering-can spray and left to dry. Their removal was attempted by fire-hosing at various times afterwards. In no case was a decontamination factor greater than 2 achieved even after several hosings.

SIGNIFICANCE OF THE IDENTIFIED UNCERTAINTIES

The uncertainties in the topics identified in the preceding sections may have a significant influence on the predicted consequences of an accidental release. For example, the suggested values for dry deposition velocity in urban areas, and for the average shielding factor in urban areas could lead to overall reductions

in dose by up to two orders of magnitude, compared with calculations based on standard assessment models. Similarly in wet conditions, although the ground deposition would not necessarily be underpredicted by current models, when account is taken of that which may be lost due to run-off, doses in urban areas may be significantly reduced.

For a given hypothetical release the scale of the plans necessary to implement this countermeasure would be influenced by any uncertainty in the factors affecting dose assessment and, in addition, by the assumed effectiveness of decontamination procedures.

A RESEARCH PROGRAMME TO RESOLVE SOME OF THE IDENTIFIED DEFICIENCIES

This experimental programme is currently under way at the Atomic Energy Research Establishment, Harwell and at the National Radiological Protection Board, Chilton. The work is receiving support from the United Kingdom Nuclear Installations Inspectorate, London.

The topics to be studied in the research programme may be summarised as follows:

(i) the development of techniques for contaminating surfaces that can be readily applied in investigations of both natural weathering and forced decontamination.

(ii) measurements of the retention of surface contamination on a variety of urban materials as a function of time after contamination (natural weathering).

(iii) evaluation of the effectiveness of techniques for decontamination of urban materials in which caesium, in representative physico-chemical forms, has been deposited (forced decontamination).

(iv) measurement of the long-term retention of fall-out Cs-137 on urban surfaces.

(v) assessment of run-off in an urban environment by investigating the behaviour of fall-out Cs-137 and other non-radioactive materials.

(vi) assessment of shielding factors in UK urban areas.

REFERENCES

(1) Linsley, G.S. Surface contamination following accidental releases of radionuclides: uncertainties in consequence prediction and research to resolve them. Chilton, NRPB-M98, to be published.

(2) Simmonds, J.R., Haywood, S.M. and Linsley, G.S. Accidental releases of radionuclides: a preliminary study of the consequences of land contamination, Chilton, NRPB-R133 (1982) (London, HMSO).

(3) Roed, J. Investigation of surface deposition pertaining to the calculation of the deposition of aerosols released in core-melt-down accidents in power reactors. Riso National Laboratory, Riso-M-2274 (1981) (in Danish).

(4) Kelly, G.N. and Clarke, R.H. An assessment of the radiological consequences of releases from degraded core accidents for the Sizewell PWR, Chilton, NRPB-R137 (1982).

(5) Hedemann Jenson, P. Shielding factors for gamma radiation from activity deposited on structures and ground surfaces, Riso National Laboratory, Riso-M-2270 (1982).

(6) Maloney, J.C. and Meredith, J.L. Decontamination of land targets, vehicles and equipment. US Army Nuclear Defence Laboratory, Maryland. NDL-TR-66 (1966).

(7) Warming, L. Weathering and decontamination of radioactivity deposited on asphalt surfaces, Riso National Laboratory, Riso-M-2273 (1982).

ENVIRONMENTAL EMERGENCY RESPONSE PROGRAM FOR A UNITED STATES NUCLEAR MATERIALS FABRICATION FACILITY*

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INTRODUCTION

The Rocky Flats Plant is a United States Government-owned and contractor-operated facility. It is part of a nationwide nuclear materials research, development, and production complex administered by the U.S. Department of Energy (DOE). The operating contractor for the Rocky Flats Plant is the Energy Systems Group of Rockwell International. Operations at the Plant include fabrication and assembly of weapons components from plutonium, uranium and beryllium. Chemical recovery and purification of process-produced transuranic radionuclides, fissile material studies, and critical assembly experiments also are conducted. Due to these operations, emergency response to accidental releases of airborne or waterborne effluents may be critical to avert any significant health impact. This paper describes the environmental emergency response program for the Rocky Flats Plant.

PLANT GEOPOLITICAL SETTING

The Rocky Flats Plant is located 26 kilometers from Denver, Colorado, whose population exceeds one million people. The site consists of 2,650 hectares of federally owned land. Typical air flow patterns are toward the metropolitan area. Plant surface water flows into two downstream drinking water reservoirs.

The natural environment of the Plant site is influenced by the Front Range of the Rocky Mountains and is characterized by dry, cool winters and warm, somewhat moist summers. There is considerable clear-sky sunshine. The average precipitation and relative humidity are low. Area hydrology is influenced by gravelly and highly permeable alluvium topsoil. Water retention in this soil is poor, and vegetation is sparse. These features produce harsh, semiarid conditions. Winds at Rocky Flats, although variable, are predominantly westerly, with strong winter winds. A description of the facility and its impact on the surrounding environment is given in the Environmental Impact Statement.[1]

The environs are routinely monitored for ionizing radiation and for pertinent radioactive, chemical, and biological pollutants. Air, water, soil, and vegetation are sampled on the Plant site and throughout the surrounding region. The results of these tests are reported in an Annual Environmental Monitoring report.[2] Several Federal, State, and Local governmental agencies independently conduct additional environmental surveys on and off the Plant site.

Monthly environmental data exchange meetings are held with the Colorado Department of Health (CDH). Government health officials at all levels, interested citizens, and the news media may attend this meeting. These meetings provide the opportunity to document results of monitoring programs,[3, 4] and for public discussion regarding reported levels of various materials associated with

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the Rocky Flats Plant. Special discussion topics include analytical procedures, calculational methods, new and ongoing construction projects at the Plant, and health effects studies. This ongoing interface with the State of Colorado and others provides a positive means to promote cooperative actions and to enhance public communications.

EMERGENCY RESPONSE PLANS AND CAPABILITIES

Rocky Flats has comprehensive emergency plans that provide guidance and procedures designed to protect life and property within the facility, the health and welfare of surrounding metropolitan communities, and the defense interests of the nation during any credible emergency situation. Mutual assistance and coordination with Federal, State, and Local agencies is assured on a cooperative basis. Periodically, Plant scientific staff, in conjunction with personnel from the CDH, conduct a complete review of the Plant's defined Maximum Credible Accident (MCA). The purpose of this review is to ensure the current adequacy of the MCA for emergency response planning. The MCA is defined as the accident of highest consequence that can reasonably be expected to occur. A lower limit probability value of 1×10^{-7} per year was selected as a cutoff value on accidental releases of radioactivity for which the Plant would prepare. Accidents with a lower probability were not considered credible. The MCA event identified for the Rocky Flats Plant postulates a large, fully fueled aircraft crashing into one of the plutonium processing buildings. The consequence of this MCA is a possible plutonium release of 100 grams.

Other events were considered. Natural phenomena providing the greatest potential hazards were earthquakes, tornados, and extreme winds. Concurrent or event-precipitated accidents, such as fire and explosion following an earthquake, were evaluated. Operational accidents were analyzed based on estimated probability of occurrence and associated quantity and physical properties of material released. Leak path and dispersion analyses were used to predict the potential material release from buildings. All other events were found to be of lesser consequence.

The Rocky Flats Emergency Plan is designed for response to all postulated accidents.[5] The plan includes integration of local, state and federal emergency response agency plans. The plan provides guidance during emergencies and for recovery to normal operations and directs that the Plant be as self-sufficient as possible in handling emergencies. During an emergency, a cadre of management assemble at the Plant Emergency Operations Center (EOC) to implement the plan. The EOC is equipped with radio, telephone, and closed circuit television communications systems.

Emergency support services at the Plant include a medical facility, analytical laboratories, fire department, security department and service garage. Teams of health physicists, industrial hygienists, safety engineers and environmental engineers are available to direct actions to minimize and control any emergency.

For environmental control and assessment, the Plant operates a routine environmental surveillance program that includes building air effluent sampling, an ambient air sampling network, and water and soil sampling programs. Surface water is contained and controlled by a series of retention ponds. Additionally, gas operated generators, portable air samplers, portable meteorological stations, radiation monitoring instruments, a four-wheel drive pickup truck and an all-terrain vehicle are available for emergency use.

Meteorological data are obtained from several onsite meteorological stations and from a computer-accessed regional 16-station network. Using the site-observed meteorological data, localized impact is estimated with a straight-line Gaussian model.

Due to complex terrain at the Plant, spatially and temporally changing windfields are produced resulting in nonlinear, time-dependent plume trajectories. Therefore initial regional modeling is

conducted utilizing computer assessed real-time meteorological data from the regional meteorological monitoring stations. These data are hand charted on a topographic overlay of the surrounding region. Standard streamline/isotach analysis of the data then results in a nonlinear trajectory that projects downwind arrival time for the plume. Further site specific regional modeling is conducted using a government-supported nationwide service entitled Atmospheric Release Advisory Capability (ARAC).[6] The ARAC system consists of a central computer facility at the Lawrence Livermore National Laboratory in California. Local, regional and global meteorological data and pollutant release rate information are evaluated by a three-dimensional, particle-in-cell transport and diffusion computer code. The code incorporates the effects of stratified shear flows, calm conditions, variable regional topography and wet and dry deposition rates and calculates atmospheric concentration, ground deposition and radiation dose. This information is transmitted to a local computer system and is available on a CRT screen and as a printed copy. ARAC projections are useful during prolonged emergencies, for post-emergency guidance and for accident evaluation.

In addition to the Plant response capabilities, the DOE has nationwide and regional radiological assistance plans in which the total resources of the DOE and its contractors can be called upon.

If an incident at the Plant is perceived to endanger the health and safety of the general public, the Colorado Radiological Emergency Response Plan for Rocky Flats[7] would be activated. This plan has been developed through agreements with the State of Colorado, county and city governments, the DOE, and Rockwell International. Coordination of emergency activities in the public sector is controlled by the Governor of the State of Colorado and coordinated through the State Division of Disaster Emergency Services.

Exercises of the Plant and State Emergency plans and procedures are conducted periodically to ensure that emergency response teams remain fully trained and ready to handle emergency situations. Exercises have included the participation of offsite emergency support groups. Post test critiques are conducted to determine efficiency and to modify procedures as required.

ENVIRONMENTAL EMERGENCY RESPONSE ACTIVITIES

The Environmental Analysis (EA) Emergency Response Team supports all emergencies with potential for accidental airborne or waterborne emissions of radioactive or chemical substances. This Team consists of a response coordinator, recorder, communications person, meteorologist, health physicist and environmental engineers.

Notification of a potential emergency is obtained through a dedicated telephone alarm system. Within 10 minutes of notification, the EA Team is assembled, manpower assignments are made in accordance with a pre-established check sheet and communications are established with the Plant EOC. Within 15 minutes, information on current weather data, potential geographic impact area, Gaussian and nonlinear plume trajectories, and initial health impact estimates are determined and forwarded to the EOC. If the emergency is determined to be real and involves the potential for abnormal airborne discharges, the ARAC Central Facility is activated and site specific regional models of the plume are available within 45 minutes to two hours. Additionally, analytical laboratories are alerted and a check is made to determine that the surface water control dams are all closed. During the duration of the emergency, communications with the EOC continue with 15-minute EA Team action updates. Information is periodically obtained from the EOC regarding the status and magnitude of the emergency.

Field activities during the first hour include a physical check of the dams and routine air samplers and, as deemed necessary, generator-powered supplementary portable air samplers and a meteorological tower are established at strategic locations.

After termination of the emergency, an environmental sampling plan is established, samples are collected and analyzed, and the final impact is assessed. Also a post emergency critique is conducted.

In conclusion, the Rocky Flats Environmental Emergency Response plan is a viable, workable, and tested plan that is designed to define and/or avert adverse environmental impact.

REFERENCES

1. Final Environmental Impact Statement, Rocky Flats Plant Site, U.S. Department of Energy, April 1980, DOE/EIA-0064.
2. Annual Environmental Monitoring Report, Rockwell International, April 15, 1983, RFP-ENV-82.
3. Hornbacher, D. D. and C. J. Barker, Private communication, June 1983.
4. Environmental Surveillance Report on the USDOE Rocky Flats Plant, Colorado Department of Health, June 1983.
5. Hayen, J. C., Private communication, April 1982.
6. M. H. Dickerson and R. C. Orphan, "Atmospheric Release Advisory Capability," Nuclear Safety 17, 281-289 (1976).
7. Colorado Radiological Emergency Response Plan for Rocky Flats, February 1979.

EMERGENCY RESPONSE TO TRANSPORTATION ACCIDENTS INVOLVING RADIOACTIVE MATERIALS IN THE SOUTHERN UNITED STATES

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INTRODUCTION

The widespread transportation of radioactive material (RAM) requires programs for responding promptly and effectively to accidents and incidents. In the United States, the U.S. Department of Transportation (DOT) regulates RAM shipments but the states are responsible for emergency response with the support of the Federal government. This study examined the emergency response capability in the Southern States Energy Board region of 16 states, namely Alabama, Arkansas, Florida, Georgia, Kentucky, Louisiana, Maryland, Mississippi, Missouri, North Carolina, Oklahoma, South Carolina, Tennessee, Texas, Virginia and West Virginia.

The magnitude of RAM shipments is difficult to delineate in brief because of the variety of materials, their quantities, types of containers, and transport modes. A number of packages in excess of 2.5 million was estimated for 1975 (Gr76). Among the more common shipments are Mo-99 for medical use, Ir-192 for radiography, and radioactive wastes for near-surface burial. In addition to these shipments at the level of 1-100 curies (Ci), some much higher curie levels are transported in Co-60 teletherapy sources (1,000 -10,000 Ci) and spent fuel elements (100,000 - 1,000,000 Ci).

External radiation exposure and environmental radioactivity contamination from these shipments are limited by regulations concerning surface radiation and radioactivity levels, radionuclide amounts, packaging, labelling, and handling in Title 49 Code of Federal Regulations Parts 171-178. Nevertheless, some transportation accidents and incidents of noncompliance will occur.

The DOT Hazardous Material Incident Report System, to which carriers must submit reports of incidents and accidents, shows an average of 20 accidents with RAM per year between 1971 and 1980, divided almost evenly between handling and transportation (Mc80). Incidents, such as excessive surface contamination, external radiation beyond that indicated by the label, or mishandling of a package, occurred 46 times per year. These frequencies are very low compared to the average 8,600 accidents per year for all hazardous materials. Only 11 major vehicular and railroad accidents, and no airline accident, involved RAM during the same 10-year period and none caused radioactivity in the environment (TF81).

Impetus for emergency planning by states has come from the U.S. Nuclear Regulatory Commission requirements for a Radiological Emergency Response Plan in support of nuclear power plants (Gr80). A Transportation Radiation Emergency Response Plan (TRERP) can be based on a number of available guides (TF81, Be80, IA81, B180, Mc77). Plan implementation is initially by police and firefighters who are usually first at the scene of an accident, and by radiation protection officials who respond to reports of radiation problems. In serious

cases, support may be needed from Federal emergency response teams (SE81), from nearby states, and other radiation protection specialists within the state. Response quality depends on planning, training, and availability of sampling and detection instrumentation.

Information from this study, obtained by inquiries with officials and RAM carriers in the 16 participating states and at two conferences held by the Southern States Energy Board, is presented in a detailed agency report (Ca 82). Emergency response capability was evaluated from the availability of a well written TRERP and of organizational preparedness.

RESULTS AND DISCUSSION

Each of the 16 states has a TRERP, completed or under development. The TRERP is usually part of a Radiological Emergency Response Plan or an Emergency Operations Plan. More recently prepared plans are models of detailed information but some of the older ones are so poorly organized that their application in an emergency would be difficult. The latter ones are being revised with guidance by the Federal Radiological Preparedness Coordinating Committee (TF81).

State radiological protection officials are responsible for the TRERP because of the specialized competence needed in dealing with radiological problems. These officials are assigned to various agencies in the 16 states, notably those concerned with public health or the environment. Officials from disaster response agencies are formally involved in some of the plans. Their participation is important when the accident is serious or has other associated hazards.

The only broadly operative interstate agreement is the Southern Mutual Radiation Assistance Plan (SE81), although 3 states do not participate. Assistance from Federal response teams is invoked through the Interagency Radiological Assistance Plan.

Qualified individuals within the state, who could provide support, are listed in only a few TRERPs. These lists do not always indicate the specific competence of the individual.

Maintaining current addresses and telephone numbers for reporting accidents or requesting assistance was indicated as a major problem in many states. Changes in personnel or responsibilities among state offices are sources of delay when rapid response is necessary.

State officials recognize the importance of special training through attendance at short courses and distribution of manuals, particularly because effective response depends on decisions by those first on the scene -- usually the carrier's employees and police or firefighters. Short courses on RAM accident response are presented by Federal agencies (Va80, TF81), and state employees also obtain radiological training through other emergency response and defense courses (Mi80). Useful training materials (RE79) and hazards guides (DT80) are available and some carriers provide training courses and manuals. Such training is a continuing function because of personnel turnover, and difficulty in keeping pace with turnover is generally identified as a problem.

The Federal Emergency Management Agency and predecessor agencies have supplied the states with many radiological survey instruments (SE81). Simple survey instruments are widely distributed within states, but the more complex and specialized the instrument, the fewer are available and the less widely distributed they are (Mi80). Instances of inadequate maintenance were cited. In general, for highly specialized detection instruments in good working condition, one needs to depend on specific radiological laboratories within a state or the Federal emergency response team.

Transportation and communication appear to be readily available to state emergency response personnel. Some emergency vehicles equipped with radiological detection instruments are also available. Equipment maintenance, again, is a major problem.

Program effectiveness has been evaluated by responses to reported accidents or incidents. In a few cases, the program has been field tested in training exercises, but some of these have been criticized because results were not rigorously evaluated.

Another limitation in planning and implementing emergency action response is the lack of needed Protective Action Guides which would cover most of the potential situations encountered in transportation accidents.

CONCLUSIONS

Each of the 16 states in the Southern States Energy Board region has an agency responsible for handling transportation radiation emergencies and operates under a response plan that is either completed or under development. Each state also has available to it supporting Federal emergency response teams, and most of the states have an interstate agreement for mutual support.

Certain older emergency response plans need improvement in accord with Federal guidance for greater clarity and information content. Direct agreement with neighboring states for mutual assistance would provide a larger pool of professionals and trained technicians than is normally available to any one state. Agreements with hospitals to admit patients with possible radioactivity contamination must also be prearranged. Maintaining current listings of names, addresses,and telephone numbers for officials and consultants should be accomplished by computer application.

Effective implementation of the plan requires a continuous training effort, especially for firefighters and law enforcement officials who make initial emergency responses on the basis of on-the-scene observations, shipping papers, and survey meter readings. Newly employed persons will need short course training, and manuals that provide guidance in terms of relatively simple categories must be available to all responders. Maintenance of radiation detection equipment was indicated as an important priority. Also needed for use in serious accidents are appropriate Protective Action Guides established by the U.S. Environmental Protection Agency.

In planning and implementation, a three-tiered response is needed. First is a false alarm, followed by minor radiation exposure or radioactivity contamination, while occurrences of major exposure or contamination are most rare. All three events, however,

require well planned responses that are appropriate for the circumstances and have the confidence of the public.

REFERENCES

Be80 Bernardo, B.C., 1980, "The Need for Emergency Preparedness in Transport of Radioactive Materials", in Patram '80, BAM, Berlin,pp. 1371-1393.

Bl80 Bland,W.F.,1980, "National Arrangements of Incidents Involving Radioactivity (NAIR)", NRPB Bulletin, July, pp.17-18.

Ca82 Carter, Melvin W. and Kahn, Bernd, 1982, "Emergency Response to Transportation Accidents Involving Radioactive Material", report to the Southern States Energy Board and the U.S.Department of Energy, DOE/ET/47916-1 Vol. III.

DT80 U.S. Department of Transportation, 1980, "Hazardous Materials 1980 Emergency Response Guidebook", U.S. DOT Report, DOT P 5800.2.

Gr76 Grella, A.W., 1976, "A Review of Five Years Accident Experience in the U.S.A. Involving Nuclear Transportation", in Transport Packaging for Radioactive Materials, IAEA, Vienna, pp.225-239.

Gr80 Grimes, B.K. and Ryan, R.G., 1980, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants", U.S. NRC and FEMA Report NUREG-0654/FEMA-REP-1.

IA81 International Atomic Energy Agency, 1981, "Emergency Response Planning for Transport Accidents Involving Radioactive Materials", A Pre-Publication Working IAEA Document, IAEA-TECDOC.

Mc77 McLean, D.R., Seymour, C.G., and Weeks, J.L., 1977, "Emergency Planning and the Transportation of Irradiated Nuclear Fuel in Canada", in Handling of Radiation Accidents, IAEA, Vienna.

Mc80 McClure, J.D. and Emerson, E.L., 1980, "A Review of U.S. Accident/Incident Experience Involving the Transportation of Radioactive Material (RAM), 1971-1980", in Patram '80, BAM, Berlin, pp.811-816.

Mi80 Mitter, E.L., Hume, R.D., Vilardo, F.J., Feigenbaum, E., Biggs, H., and Piercy, R., 1980, "Survey of Current State Radiological Emergency Response Capabilities for Transportation Related Incidents", U.S. NRC Report NUREG/CR-1620.

RE79 Radiation Emergency Assistance Center, 1979, "Response to Radioactive Materials Transport Accidents", DOT/RSPA/NTB-79/8.

SE81 Southern Emergency Response Council, 1981, "The Southern Mutual Radiation Assistance Plan - Revision II", Southern States Energy Board, Atlanta, GA.

TF81 Task Force on Transportation Accidents, 1981, "Guidance for Developing and Reviewing State and Local Radiological Emergency Response Plans for Transportation Accidents", Federal Radiological Preparedness Coordinating Committee, Working Draft, September 19, 1981.

Va80 Vandevender, S.G., and Reese, R.T., 1980, "Emergency Response Capabilities Developed in the United States to Deal with Nuclear Materials Transportation Accidents", in Patram '80, BAM, Berlin, pp. 597-603.

THE CURRENT STATE OF EMERGENCY PREPAREDNESS AT U. S. POWER REACTORS

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Following the accident at Three Mile Island in March 1979, the U.S. Nuclear Regulatory Commission (NRC) began a program to improve the state of emergency preparedness at power reactors in the United States. This program has included four major phases:

- review and evaluation of emergency plans submitted by reactor operators;
- appraisal of emergency procedures, equipment, facilities, and personnel capabilities;
- observation of drills and exercises designed to test the reactor operator's ability to implement the emergency plans; and
- follow-up inspections to assure maintenance of emergency preparedness.

The objectives of the program were to ascertain whether the reactor operators were in compliance with NRC regulations and to determine whether they were taking appropriate actions to protect nuclear materials and facilities, the environment, and the health and safety of the public. The NRC has conducted this program with technical assistance from Pacific Northwest Laboratory (PNL).[a]

The NRC and the Federal Emergency Management Agency (FEMA) developed guidance for emergency planning in January 1980 and published NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (USNRC and FEMA 1980). The NRC amended it's regulations (Code of Federal Regulations) that pertain to emergency preparedness on August 19, 1980. Reactor operators were required to submit revised emergency plans by January 2, 1981 and to implement the plans by April 1, 1981. They were also required to demonstrate, by July 1, 1981, their ability to alert and provide prompt instructions to the public within the plume exposure pathway emergency planning zone (EPZ). Due to procurement and installation problems, this latter deadline was extended to February 1, 1982.

The revised emergency plans of each reactor operator were reviewed against the 16 planning standards and the 96 evaluation criteria of NUREG-0654. The review reports included a synopsis of the planning elements that were satisfactory and a specific listing of criteria that were not met. These reports have served as a base for additional revisions and upgrading of emergency plans. This effort has resulted in significant improvements in emergency plans and further enhancements are expected in the future.

The second phase of the program was a comprehensive appraisal of the overall adequacy and effectiveness of the emergency preparedness programs of each reactor operator. The appraisal was structured to facilitate an integrated evaluation of the total program. A modified Management Oversight and Risk Tree (MORT) analytical logic methodology was used. The MORT tree included seven

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(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

major elements of emergency preparedness: administration, organization, training, facilities and equipment, procedures, offsite coordination, and drills and exercises. Each branch of the tree included sub-elements and questions designed to ensure consideration of all essential aspects of an emergency preparedness program (USNRC 1983a).

Appraisals at 54 nuclear power plants in the U.S. were completed between April 1, 1981 and April 1, 1982. Some of the more significant problems that were found are as follows:

- Emergency plan implementing procedures were either incomplete or needed clarification.
- Training programs were either incompletely developed or not adequately documented.
- Interim emergency response facilities, such as the technical support center, operations support center, and emergency operations facility, were inadequate because of improper location, insufficient size, or lack of equipment.
- Emergency action levels were incomplete or were not integrated into procedures so that operating personnel could quickly recognize and classify an emergency.
- There was an incomplete basis for recommending offsite protective actions to appropriate officials.
- Accident and dose assessment techniques were not adequate for estimating potential offsite radiation exposures.
- Post-accident sampling and analysis systems were not capable of assessing reactor damage.
- Public information brochures were not adequately developed or distributed (Pagano 1982).

In addition to these major problems, a number of less significant improvement items were identified which, if implemented, would enhance the emergency preparedness program.

The revised NRC regulations (Code of Federal Regulations) requires each reactor operator to conduct an annual exercise of the emergency plan. The intent of the exercise is to test the major elements of the onsite and offsite emergency plans and organizations and to demonstrate their integrated capabilities. The number of exercises that have been conducted in 1980, 1981, 1982, and 1983 are 6, 34, 52, and 59, respectively. Initially, the observation of exercises was concentrated on the basic capabilities of the participants. More recent exercise observations have revealed some significant problems, which are summarized as follows:

- Exercise scenarios have included unrealistic or incredible equipment failures that have confused reactor operators and interfered with their ability to diagnose and mitigate the accident. Scenarios have also included inconsistent or unrealistic radioactive source term data, dose rates, and release pathways.

- In the Control Room, emergency management has been weak in some cases, access control has been poor, and the operator's log has not been maintained.

- In the Technical Support Center, there has been a tendency to classify emergencies based primarily on radiological conditions rather than deteriorating plant conditions, which has frequently resulted in late or incorrect declarations of emergency classifications.

- In the Operations Support Center (OSC), contamination control has been poor and communications between the OSC and in-plant teams has not been adequate. Briefings of the OSC staff have been infrequent or incomplete.

- At the Emergency Operations Facility, coordination between the plant and government agencies has been less than adequate. Offsite officials have not had an adequate understanding of the plant, plans, or procedures.

- Radiation dose assessment coordinators have not used their offsite radiological monitoring teams to effectively gather meaningful data.

- At the News Center, plant representatives have not been capable of adequate interpretations of plant status and radiological data.

In spite of these problems, the exercises have demonstrated that reactor operators could successfully identify and respond to a radiological emergency so as to protect the health and safety of it's employees and the public.

The final phase is an on-going inspection program to assure the maintenance of emergency preparedness. Initially, the inspections focused on the problems that were identified by the appraisals. Revised emergency plans and procedures were re-evaluated. Facilities that were built and equipment that was installed were evaluated against the criteria of NUREG-0696, "Functional Criteria for Emergency Response Facilities" (USNRC 1981). Training programs were evaluated by a series of interviews with emergency workers in which they demonstrated their response capabilities. Formal inspection procedures have been developed to provide requirements and guidance to the inspection teams (USNRC 1983b). These procedures assist the NRC in applying a uniform and consistent inspection program at all the U. S. power reactors.

The state of emergency preparedness at U. S. power reactors is considerably improved since 1979. Emergency plans and procedures have been enhanced to meet the criteria of NUREG-0654. The appraisal program identified major problems at individual plants that have now been corrected for the most part. The appraisal program will also assure that new plants are adequately prepared before initiating operation. The corrective actions promoted by the inspections have resulted in significant improvements in emergency preparedness. Although program enhancements are continuing, a reasonable degree of emergency preparedness and response capability has been established to protect the health and safety of reactor employees and the public.

REFERENCES

Code of Federal Regulations. Title 10, Part 50, Appendix E.

Pagano, F. G. 1982. "NRC Experience in Upgrading Emergency Preparedness at Power Reactors." In Transactions of the American Nuclear Society, 41:536.

U.S. Nuclear Regulatory Commission and Federal Emergency Management Agency (NRC, FEMA). 1980. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants. NUREG-0654, FEMA-REP-1, Revision 1, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1981. Functional Criteria for Emergency Response Facilities. NUREG-0696, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1983a. Emergency Preparedness Implementation Appraisals. I.E. Inspection Procedure 82101, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1983b. Emergency Preparedness Inspection Procedures. IE Inspection Procedures 82201-82211, Washington, D.C.

PLANNING THE APPLICATION OF PROTECTIVE MEASURES FOR ACCIDENTS PRODUCING SEVERE RADIOLOGICAL CONSEQUENCES IN NUCLEAR POWER PLANTS

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INTRODUCTION

In order to evaluate the consequences and pre-plan the application of countermeasures on the population irradiated by a release of radioactive products from a nuclear power plant, the dose equivalent is calculated during the passage of the "radiactive cloud" and during irradiation resulting from the radiactive deposit accumulated on the ground.

When an accident takes place, the resulting exposure can only be substantially limited by implementing protective measures. The possible protective measures are: "sheltering" and supplying those exposed with stable iodine tablets, during the early phase. In a second stage, when the irradiation is due to ground deposit, the main countermeasure is the evacuation of people. This practice takes more time but may be implemented without extreme urgency.

RADIOLOGICAL CRITERIA FOR IMPOSING COUNTERMEASURES

In pre-planning the application of protective measures for accidents producing severe radiological consequences in nuclear installations, the underlying short-term objetive is to ensure that prompt action is taken to avoid the incidence of high levels of individual exposure. This may take the form of advice to the public to shelter in their houses, while stable iodine tablets may be supplied as a thyroid blocking agent. In a second stage, evacuation must be considered, when ground deposit external irradiation is the dominant way of exposure.

The Argentine radiological authority has established the following dose limits above which the countermeasures must be implemented: a) Sheltering and distribution of stable iodine tablets, when the effective dose equivalent is 1 rem or greater, or when the organ dose equivalent reaches 10 rem, both concerning the passage of the cloud; b) When the dose due to external irradiation from the ground is over 10 rem within the first 6 hours after the accident, evacuation is mandatory. Nevertheless, if the dose is 10 rem or more within the first 24 hours after the accident, the evacuation should be implemented only for those cases in wich the detriment expected from the countermeasure is less significant than the radiological detriment.

CHARACTERISTICS OF THE RELEASES

The characteristics of the 3 hypothetical releases, designated as 1, 2 and 3, are summarized in Table 1. The fraccional releases of different groups of nuclides from the reactor core are specified along with other assumed characteristics, such as duration, height and heat content of the release and delay between reactor shut-down and release.

In general terms, the characteristics of release 1 are typical of a large degraded core accident release postulated for HWRs (1). Release 2 is similar to Nr. 1, but the release of most elements, except for noble gases, is a factor of ten lower. Release 3 is a typical noble gas release.

The reactor inventory at shut-down is summerized in Table 2 and was calculated by means of the ORIGEN 1 code, adapted for a 1100 MW heavy water reactor (2). The quantities of each radionuclide released in each case can be obtained from the release fractions shown in Table 1.

DOSIMETRIC MODEL

The SEDA (3) dose-assessment model has been used for the calculation of individual doses. The dispersion model used in it, is the so-called Gaussian model. In it, it is assumed that the material released to the atmosphere will be carried with the wind and spread like a smoke plume. At distances greater than 20 km, concentrations and doses calculated with the Gaussian model are usually overestimated.

The plume rise due to the heat content of the release was taken into account and calculated according to a formula provided by Briggs (4). Decreases in the concentration of the plume due to deposition were evaluated by using the source depletion model.

The individual doses are calculated for three forms of irradiation: i) External gamma irradiation from the plume; ii) Inhalation (lungs and thyroid); and iii) External gamma irradiation from ground deposited material.

Finally, it has been assumed that people stay outdoors all the time and, therefore, shielding factors were disregarded so as to fix the distances at which the countermeasures had to be implemented.

DOSES TO INHABITANTS

The individual doses calculated as functions of the distance from the reactor, for the three releases under consideration, are shown in Figures 1 to 6.

The highest doses due to immersion in the radioactive cloud were 140 and 750 rem for Pasquill's classes D and F respectively, in release 3, both within one kilometer from the reactor. Releases 1 and 2 produce lower doses but at greater distances because of the heat content of the released gases.

The doses in thyroid due to inhalation of radioactive material were calculated. The highest doses are caused by release type 2: over 1500 and 2200 rem, under D and F conditions, within two kilometers from the source. For release type 1, the maximum doses reach approximately 800 rem, at 15 km from the reactor.

The highest doses in lungs were produced by a release type 1, with 130 and 170 rem at 4 km and 12 km from the reactor, for D and F Pasquill atmospheric conditions, respectively.

Doses due to irradiation from the radioactive material deposited on the ground were calculated for six and twenty-four hours after the end of the release. In the first case, the maximum dose was 23 rem, at 1.5 km from the reactor, for a release type 2, under the F Pasquill condition, while, at 4 km from the reactor, the doses were over 10 rem.

The dose integrated until 24 hours after the deposition, was larger in a type 2 with values over 60 rem at less than 1 km from the source, under a D Pasquill condition. Doses over 10 rem were calculated at 45 km, in a release type 1, under the F Pasquill condition.

CONCLUSIONS

The individual doses were calculated on the "hot line" of the radioactive plume, under unfavourable meteorological conditions for 3 hypothetical releases. Some simple countermeasures, such as "sheltering" and blockade of the thyroid gland, can significantly reduce the doses incurred by immersion in the cloud and by inhalation.

The doses due to irradiation from the radioactive material deposited on the ground indicate that the late evacuation plans must cover areas within at least a ten kilometer radius around the reactor site. This could be extended to 45 km for those cases in which the countermeasures themselves do not mean further significant risk.

Table 1

Release categories for accidental releases considered in the assessment

TYPE	tr (h)	td (h)	hs (m)	QH (MW)	Nobie gases	Iodine	Cs-Rb	Te-Sb	Sr-Ba	Ru	La
1	2.5	0.5	25	5.88	$9\ 10^{-1}$	$7\ 10^{-2}$	$4\ 10^{-2}$	$3\ 10^{-2}$	$5\ 10^{-3}$	$4\ 10^{-2}$	$3\ 10^{-4}$
2	2.0	3.0	0	0.29	$6\ 10^{-1}$	$9\ 10^{-3}$	$4\ 10^{-3}$	$3\ 10^{-3}$	$5\ 10^{-4}$	$3\ 10^{-4}$	$4\ 10^{-5}$
3	2.0	3.0	10	-	1	$1.5\ 10^{-4}$	$5.1\ 10^{-4}$	$5\ 10^{-4}$	$5.7\ 10^{-5}$	$4\ 10^{-5}$	$6.5\ 10^{-6}$

Table 2

Inventory of nuclides considered in this study

Nuclide	Inventory (10^6 Ci)	Nuclide	Inventory (10^6 Ci)	Nuclide	Inventory (10^6 Ci)	Nuclide	Inventory (10^6 Ci)
Kr- 85	0.055	Cs-136	0.56	Ba-140	59.91	Ce-141	54.05
Kr- 85m	8.83	Cs-137	0.45	Mo- 99	60.90	Ce-143	50.77
Kr- 87	17.25	Te-127	2.14	Tc- 99m	52.88	Ce-144	12.37
Kr- 88	25.05	Te-127m	0.23	Ru-103	17.30	Pr-143	49.60
Xe-133	61.72	Te-129	12.23	Ru-105	30.15	Nd-147	22.67
Xe-135	4.60	Te-129m	1.82	Ru-106	3.30	Np-239	931.06
I-131	32.82	Te-131m	4.93	Rh-105	20.0	Pu-238	$4.0\ 10^{-4}$
I-132	48.64	Te-132	46.24	Y- 90	0.36	Pu-239	$4.7\ 10^{-3}$
I-133	65.09	Sb-127	2.46	Y- 91	34.91	Pu-240	$3.9\ 10^{-3}$
I-134	74.20	Sb-129	13.09	Zr- 95	41.46	Pu-241	0.25
I-135	58.09	Sr- 89	29.53	Zr- 97	56.24	Am-241	$3.0\ 10^{-5}$
Rb- 86	$5\ 10^{-3}$	Sr- 90	0.27	Nb- 95	29.33	Cm-242	$3.7\ 10^{-3}$
Cs-134	0.165	Sr- 91	41.58	La-140	62.15	Cm-244	$1.6\ 10^{-5}$

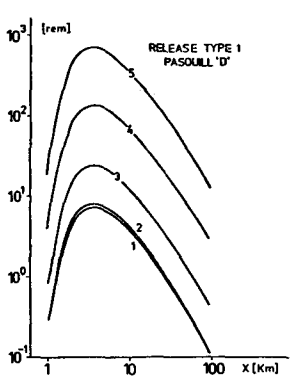

FIGURE 1

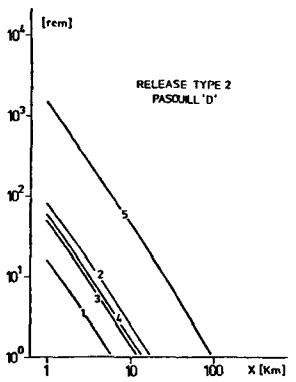

FIGURE 2

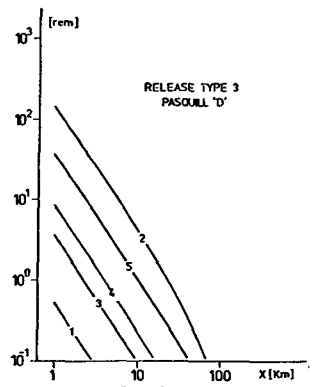

FIGURE 3

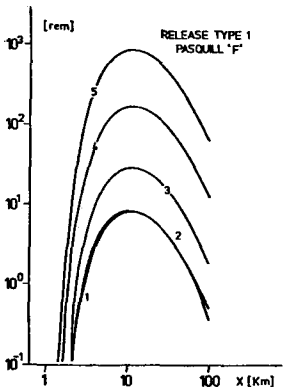

FIGURE 4

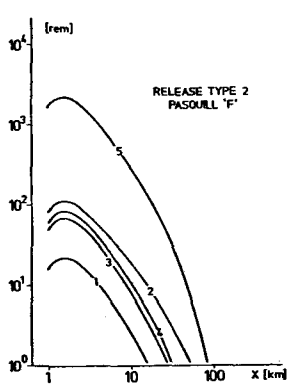

FIGURE 5

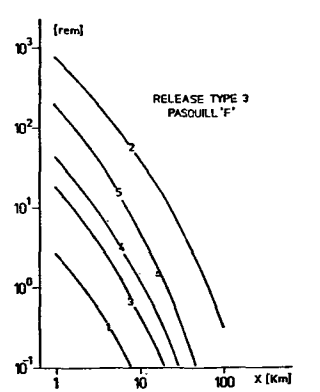

FIGURE 6

1. H_E due to a deposit, integrated in 6 hours.
2. H_E due to a deposit, integrated in 24 hours.
3. H_E due to immersion in the cloud.
4. Dose equivalent in thyroid, due to inhalation.
5. Dose equivalent in lungs, due to inhalation.

BIBLIOGRAPHY

1. Gesellschaft für Reaktorsicherheit, Germany, F.R. "Deutsche Risikostudie Kernkraftwerke" Germany, Verlag TUV Rheinland GmbH, 1980.
2. Kivilovsky, L. and Volkis, J. 1978 CNEA-RE CA 78-14.
3. Kunst, J.J., Boutet, L. and Bruno, H. 1982 CNEA (in print).
4. Briggs, G.A. 1969 TID-25075.

RADON IN HOUSES: A RADIATION PROTECTION PROBLEM IN SWEDEN

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ABSTRACT

High radon levels in Swedish homes constitute a significant radiation protection problem. Of the order of 100 000 individuals are exposed to inhaled radon daughters to levels above the occupational limit. The highest levels may surpass that limit forty-fold. The arithmetic mean of the equilibrium equivalent radon daughter concentration is about 50 Bq/m^3. The main source of the radon is probably the ground under the houses, but the building material also contributes. In some limited parts of Sweden, use of water from deep-bored wells in granite has led to elevated radon levels.

In the spring of 1983 a government commission suggested countermeasures for existing dwellings as well as investigation of the ground and design criteria for planning new buildings. These requirements are discussed from the point of view of radiation protection.

INTRODUCTION

Natural radiation in Swedish dwellings was studied by Hultquist in the early fifties and the work was continued on a smaller scale into the sixties. An article from the national radiation protection institute in 1968 warned that low ventilation rates might cause high radon levels, and the warning was repeated with greater emphasis in connection with the energy crisis 1974. Continued research demonstrated high levels of radon in various types of houses, and in 1979 the reports were given extensive coverage in mass media. Details of this development have been presented by Swedjemark (1979). The government appointed a commission to study the problems involved, and after an interim report in the spring of 1979 the final report was published in March, 1983. The government's proposal based on the final report is expected to be submitted for approval by the parliament in the spring of 1984.

LEVELS OF RADON AND DAUGHTERS

The most accurate information on levels of radon in Swedish houses stems from an investigation made during 1981 and 1982 (Swedjemark and Mjönes 1983). About 500 houses were subjected to systematic measurements with the aim of finding representative averages of the radon concentrations in four climatic regions, as well as correlations between the radon concentration and such parameters as building materials and ventilation systems. The measurements were carried out using passive radon monitors with thermoluminescent dosemeters. The results are given in Figure 1 together with results from measurements performed under the supervision of the local health authorities to determine whether the levels are above the action level. These measurements are therefore not based on random sampling. The overall distribution appears to be approximately log-

normal with an arithmetic mean of about 53 Bq/m^3 (equilibrium equivalent concentration, EEC). It is believed that in the early fifties the mean was around 15 Bq/m^3. The highest recorded value so far is about 20 000 Bq/m^3. Despite an intensive measurement program during three years, probably not more than about one-tenth of the houses with levels above 400 Bq/m^3 have been found.

The highest radon levels are associated with radon-rich ground. Radium-rich building materials alone are not known to have given higher values than 800 Bq/m^3. Separate programs to study tap water have indicated that tap water can not be a significant source of radon in dwellings except for water from a small number of deep bored private wells.

ASSESSMENT OF HAZARDS

The national institute of radiation protection has made a preliminary evaluation of the risk of attracting lung cancer from radon in dwellings. In a report in 1982 to the government commission, the institute recommended that political action be based on an assumed risk factor of 2.5×10^{-6} cases of lung cancer per unit exposure of radon daughters, (Bq/m^3) x year. This estimate is almost twice as high as the UNSCEAR estimate from 1982. If the estimate were correct, the present lung cancer incidence would increase by almost one-half after full expression of the cancers caused by the recent rise of Swedish radon levels.

No firm evidence exists yet, however, on the induction of lung cancer after radon exposure in dwellings despite several studies. Swedish studies have mainly been aiming at demonstrating the feasability of an epidemiological survey. These pilot studies have all been inconclusive, although a cancer connection similar to that for miners can not be ruled out (Edling 1983).

The extremely high levels encounted in some dwellings also raise the question of possible non-stochastic effects on the lungs. Evidence from experimental animals indicates that injury to the lung function can not be ruled out after prolonged exposures in the dwellings with the highest levels. It has, however, been judged as very unlikely that any such injury will ever be demonstrated.

POSSIBLE COUNTERMEASURES

Presently, the government commission and the Swedish national authorities have suggested or realized the following measures:

For new dwellings, a classification of the ground into three categories is required: low, normal and high probability of excessive exhalation of radon. Three corresponding types of building design are described: traditional, radon-shielding and radon-safe. Extensive information on this has been provided to the local building licencing authorities. Further, the Swedish building code requires limitation of the activity concentration of building materials.

For existing dwellings, priorities for measurements have been assigned and local health authorities have been trained on measurement techniques and interpretation of results. A proposed action

level of 400 Bq/m^3 (EEC) is associated with favorable loans, and countermeasures at the house-owner's expense below this level is encouraged. Information has been published on such actions as improved general ventilation, forced ventilation with heat exchangers and measures preventing the leaking or diffusion of radon from the ground into the house.

CRITERIA FOR SELECTING COUNTERMEASURES

The national institute of radiation protection has claimed that no effort should be spared to ensure that for life-time exposure

- o there is no risk of non-stochastic lung injury
- o the risk of lung cancer is not large compared with other cancer risks
- o the risk is below the risk from life-time exposure at the occupational dose limit.

The institute has suggested that these criteria are met if the radon daughter levels at continuous exposure are kept below 500 Bq/m^3.

The institute has further required that significant efforts be devoted to keeping the levels far below this value. Available data indicate that on the average the cost of countermeasures per unit saved collective effective dose equivalent for existing dwellings is rising as the action level is decreasing. The cost corresponds to 200 000 USD per prevented case of lung cancer if action is mandatory at 600 Bq/m^3, and to 600 000 USD at 100 Bq/m^3. These are recognised as fairly high costs in comparison with the costs for protection in Sweden in general. Since large numbers of dwellings are involved, mandatory action at levels below 100 Bq/m^3 might entail annual costs above one-half per cent of the gross national product, thus being of great economic significance.

For new dwellings the costs of protective measures are much lower. The measures selected will probably ensure an average level below 40 Bq/m^3 although occasionally a dwelling might have much higher levels.

FUTURE RESEARCH

Following several pilot studies, the government committee on the prevention of cancer has planned a major epidemiological study and judged it as feasible. Final decision to undertake this study is awaiting the report of the committee, which is expected in the fall of 1983.

Priority will be given to studies of the transport of radon in the ground, to enable differentiating between areas with varying leakage of radon from the ground into houses, and to studies on measures presenting this leakage. The mechanisms of diurnal and seasonal variations in the radon and radon daughter levels need to be elucidated.

Radon measurement techniques also deserve further study. Research on countermeasures should be directed towards study of their cost effectiveness and durability.

CONCLUSIONS

The effective dose equivalent to the average Swede due to radon in dwellings amounts to about 3 mSv per year (using the conversion factor 0.06 mSv/a per Bq/m^3 including an occupancy factor of 0.8) while all other sources together give 2 mSv/year. The variability is very high, with the maximum radon contribution being about 1000 mSv/year. It has been recognized at the political level that this is a serious environmental hazard which should be strongly reduced. The programs implemented so far indicate that it is possible by reasonable means to prevent high levels in houses to be built. For existing houses it is very difficult to track the highest levels of radon. The political decision on countermeasures expected during the spring of 1984 is likely to exhibit a very careful balance between risk reduction and the associated cost.

REFERENCES

Edling C., Lung Cancer and Radon Daughter Exposure in Mines and Dwellings, Linköping University Medical Dissertations No. 157, Dep of Occup Med, Linköping University, Linköping, Sweden, 1983.

Swedjemark G.A., Indoor Measurements of Natural Radioactivity in Sweden, SSI:1979-026, National Institute of Radiation Protection, Box 60204, S-104 01 Stockholm, Sweden, 1979.

Swedjemark G.A. and Mjönes L., Radon and Radon Daughter Concentrations in Swedish Homes. Paper presented at the International Seminar on Indoor Exposure to Natural Radiation and Related Risk Assessment, Anacapri, Italy, 3-5 October 1983.

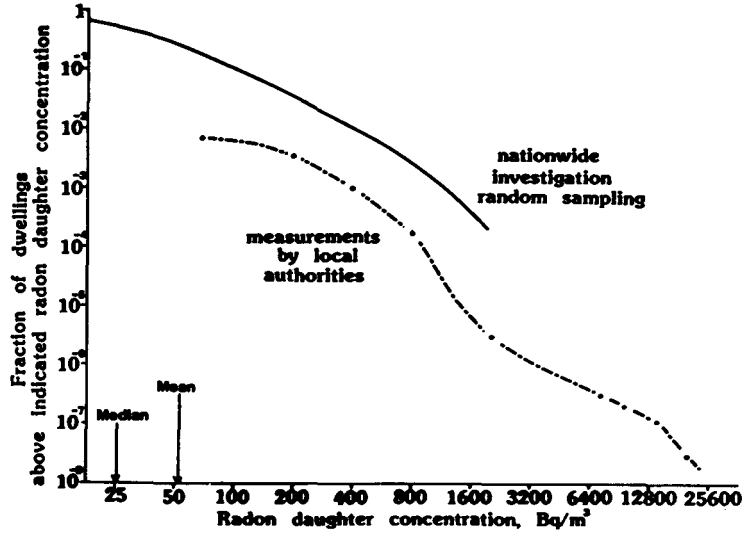

Fig 1. Distribution of dwellings according to radon daughter concentration.

L'IMPACT RADIOLOGIQUE DES REJETS DE TRITIUM D'UNE CENTRALE PWR.

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INTRODUCTION

L'objectif de cette communication est de présenter les principaux résultats d'une étude /1/ relative à l'évaluation de l'impact radiologique des rejets liquides et gazeux de tritium d'une centrale PWR en fonctionnement normal (*). Au vu du bilan de l'ensemble des rejets d'effluents des centrales PWR françaises, on voit que la majeure partie (9/10) du tritium est rejeté sous forme liquide. Compte tenu des modalités d'utilisation des eaux des divers cours d'eau sur lesquels les premières tranches PWR du programme électronucléaire ont été implantées, il est apparu intéressant d'évaluer l'impact différentiel des rejets tritiés pour différents sites : Seine, Rhône et Moselle. Pour certains sites, présentant des débits d'étiage relativement faibles, l'impact radiologique et économique d'une conversion des rejets liquides en rejets gazeux a par ailleurs été évalué.

L'EVALUATION DES IMPACTS INDIVIDUEL ET COLLECTIF

Les indicateurs qui ont été retenus pour l'évaluation de l'exposition des populations sont l'équivalent de dose effectif individuel maximal pour les personnes les plus exposées à proximité des réacteurs et l'équivalent de dose effectif collectif pour la population régionale. Les voies d'exposition retenues ont été l'inhalation, le passage transcutané et l'ingestion pour l'eau de boisson, les poissons et les produits agricoles (irrigués ou non).

Les calculs ont été effectués pour une tranche PWR de 1 300 MWe, rejetant annuellement 5.10^{12} Bq (136 Ci) sous forme gazeuse et $3.9\ 10^{13}$ Bq (1 050 Ci) sous forme liquide, valeurs conformes à l'hypothèse majorante indiquée par l'exploitant. Ces chiffres représentent donc des valeurs maximales et dans la réalité l'expérience des premières tranches en fonctionnement montrent que les rejets tritiés sont sensiblement plus faibles.

Le calcul de l'équivalent de dose effectif individuel maximal a été effectué sur la base d'un scénario pessimiste qui concerne la population "critique" résidant à proximité du site en permanence sous le vent dominant, buvant de l'eau ayant une concentration équivalente à celle existante à l'aval du point de rejet et consommant des produits agricoles locaux.Pour l'évaluation des concentrations, les modèles de transferts dans l'environnement utilisés ont été les modèles classiques proposés dans l'ouvrage C.E.A.-NRPB /2/. Pour les rejets liquides il s'agit du modèle à compartiment de Schäeffer s'appuyant sur l'hypothèse que l'activité rejetée est instantanément diluée dans tout le débit du fleuve. Le modèle adopté pour l'évaluation des concentrations dans l'air est du type panache gaussien prenant en compte les réflexions sur le sol et la couche d'inversion de température. Sont également introduits dans le modèle, les phénomènes d'appauvrissement sec et humides /3/. Les équivalents de dose effectif collectif ont été établis pour l'ensemble de la Communauté Européenne dans le cas des rejets atmosphériques (rayon de 900 km autour du site) et pour l'ensemble du bassin fluvial pour les rejets liquides.Les données relatives à la répartition géographique des populations et des produits agricoles ont été tirées respectivement de /4/ et /5/. Pour les quantités d'eau de boisson, de poissons et de produits irrigués ingérées relatives à chaque site des enquêtes spécifiques ont été réalisées /1, 6/. Les tableaux 1 et 2 détaillent les résultats correspondant aux trois sites.

(*) Ce travail a été effectué dans le cadre d'un contrat financé par le Service Etudes et Projets Thermiques et Nucléaires d'Electricité de France.

Sites		Equivalent de dose individuel maximal /μ.Sv.an^{-1}/ Voies de transfert					
		eau de boisson	poissons	végétaux	lait + viande	inhalation + passage transcutané	TOTAL
Seine	1 (*)	0,12	$1,2\ 10^{-3}$	$2,3\ 10^{-3}$	$6,4\ 10^{-4}$	-	0,130
	2	-	-	$7,9\ 10^{-3}$	$1,3\ 10^{-2}$	$3,5\ 10^{-2}$	0,057
Rhône	1	$8,9\ 10^{-3}$	$9,3\ 10^{-5}$	$3,9\ 10^{-4}$	$9\ 10^{-2}$	-	0,010
	2	-	-	$1,1\ 10^{-2}$	$1,8\ 10^{-2}$	$5,1\ 10^{-2}$	0,080
Moselle	1	0,069	$7,0\ 10^{-4}$	-	-	-	0,070
	2	-	-	$3,2\ 10^{-3}$	$5,0\ 10^{-3}$	$1,3\ 10^{-2}$	0,020

(*) 1 . Rejets liquides : $3.9\ 10^{13}$ Bq/an
2 . Rejets atmosphériques : $5\ 10^{12}$ Bq/an

Tableau 1 : Contribution de chaque voie de transfert à l'équivalent de dose individuel maximal.

Sites		Equivalent de dose effectif collectif /h.mSv.an^{-1}/ Voies de transfert					
		eau de boisson	poissons	végétaux	lait + viande	inhalation + passage transcutané	TOTAL
Seine	1 (*)	570	0,30	0,50	-	-	570,0
	2	-	-	0,70	0,10	1,60	2,5
Rhône	1	140	0,02	-	-	-	140,0
	2	-	-	0,50	0,20	2,30	3,0
Moselle	1	53	0,10	66,0	-	-	120,0
	2	-	-	1,6	0,40	3,0	5,0

(*) 1. Rejets liquides : $3,9\ 10^{13}$ Bq/an
2. Rejets atmosphériques : $5\ 10^{12}$ Bq/an.

Tableau 2 : Contribution de chaque voie de transfert à l'équivalent de dose collectif.

ANALYSE DES RESULTATS ET COMMENTAIRES

L'examen des résultats relatifs aux doses collectives montre quelques différences. Pour les rejets liquides il s'agit de l'importance relative des consommations de boisson. (Cette dernière est de loin la voie d'exposition la plus prépondérante puisqu'elle représente pratiquement 100 % de la dose collective pour les sites Seine et Moselle et plus de 50 % pour le site Rhône, le complément provenant des produits irrigués). Pour les rejets gazeux les différences s'expliquent en fonction des variations des diverses roses des vents. En ce qui concerne les doses individuelles maximales l'ingestion d'eau de boisson est également la voix d'exposition prépondérante (7/10 à 9/10 de la dose selon le site). Les différences pour les rejets liquides s'expliquent essentiellement par les différences de débit des fleuves et pour les rejets gazeux par la distance à la source des habitants les plus proches.

Confronté aux problèmes posés par la dilution des rejets liquides sur certains sites présentant des débits faibles aux périodes d'étiages, il est apparu intéressant d'analyser l'intérêt que pouvait présenter la suppression des rejets liquides par leur transformation en rejets gazeux, soit 4.4 10^{13} Bq (1 186 Ci). Afin d'évaluer l'efficacité éventuelle tant du point de vue radiologique, qu'économique d'une telle solution il a été fait l'hypothèse que l'ensemble des effluents liquides passait sur un évaporateur naturel. Le coût total actualisé d'une telle opération serait de l'ordre de 40 à 50 millions de francs pour 1 tranche (environ 7,5 millions d'investissement et 3 millions de coût d'exploitation annuel par tranche).

Le tableau 3 ci-dessous montre pour les deux sites présentant des débits plus faibles à l'étiage l'impact différentiel pour l'exposition des populations. On note le peu d'avantage qui ressort d'une telle solution.

Sites	Modalité de rejet	Equivalent de dose effectif Individuel maximal µSv/an	Collectif h.mSv/an
Seine	1	0,19	573
	2	0,50	22
Rhône	1	0,09	143
	2	0,70	26
Moselle	1	0,09	145
	2	0,20	43

1 . Rejets liquides : 3,9.10^{13} Bq/an, Rejets gazeux : 5.10^{12} Bq/an.

2 . Rejets liquides : nuls, Rejets gazeux : 4,4.10^{13} Bq/an.

Tableau 3 : Comparaison des équivalents de doses effectifs maximaux individuels et collectifs selon les modalités de rejets.

Si par ailleurs on prend en compte les coûts de l'opération dans une perspective d'optimisation telle qu'elle est préconisée par la C.I.P.R. /6,7/ les coûts de l'homme-Sievert associés à cette option sont de l'ordre de 80 millions de FF (1980) par homme-Sievert évité dans le cas du site Seine, 380 et 450 millions dans les cas respectifs du site Rhône et du site Moselle. Ces chiffres démontrent le peu d'intérêt d'une telle solution du point de vue de la diminution de l'impact collectif, l'impact individuel restant toujours inférieur à 1 µSv/an. Il faut de plus noter que l'évaporation des effluents liquides peut entrainer des contraintes d'utilisation eu égard à l'exposition des travailleurs /1/.

REFERENCES

/1/ F. RANCILLAC - L'impact radiologique des rejets de tritium des tranches du palier P'4. Influence de l'utilisation de l'A.C.E.R.E.N. Rapport C.E.P.N. n° 59. Nov. 1982.

/2/ N.R.P.B-C.E.A. - Methodology for evaluating the radiological consequences of radioactive effluents released in normal operations. C.E.C. report - 1979.

/3/ P. PAGES, F. RANCILLAC. Modèle conversationnel pour l'évaluation des transferts atmosphériques de rejets continus. Rapport C.E.A.-R-5228. Mai 1983.

/4/ A. GARNIER, A. SAUVE - Etude de la répartition géographique de la population européenne en vue de l'évaluation des conséquences radiologiques des rejets d'effluents. Rapport C.E.A-R-5096 - 1981.

/5/ A. GARNIER, A. SAUVE, C. MADELMONT. - Représentation géographique de diverses données dans une grille européenne in 5th I.R.P.A. Congress proceeding. Jerusalem 9-14 mars 1980. Vol. 3 p 307-310.

/6/ J. LOCHARD, P. PAGES - Analyse coût-efficacité du traitement des drains de plancher d'un réacteur PWR-1300 MWe. Rapport C.E.P.N. n° 50 - 1982.

/7/ I.C.R.P. - International Commission of Radiological Protection : I.C.R.P. Publication 26 - Pergamon Press 1977.

/8/ I.C.R.P. - Cost benefit analysis in the optimization of radiation protection. I.C.R.P. Publication 37 - Pergamon Press - 1983.

VARIABLE VALUES OF UNIT COLLECTIVE DOSE EQUIVALENT FOR MEMBERS OF THE PUBLIC: DIFFICULTIES WHEN APPLIED TO UK FISH AND SHELLFISH CONSUMPTION

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1. INTRODUCTION

The system of dose limitation recommended by the International Commission on Radiological Protection (ICRP) includes the requirement for optimisation, that exposures should be as low as reasonably achievable, social and economic factors being taken into account[1]. The ICRP suggests that cost-benefit analysis is one of the main techniques which may be used in this optimisation process. Basic to cost-benefit analysis is the monetary value to be assigned to the detriment caused by the exposure. The ICRP suggests[2] that this value may reflect the following two factors: first, the objective health detriment, which accounts for health effects and the value of which increases linearly with the collective effective dose equivalent; second, subjective detriments which may reflect risk-averse attitudes or regulatory requirements, and the value of which may vary with the level of individual dose.

This paper discusses the difficulties of applying a non-linear valuation of detriment when the dose is spread over a large population. The exposure pathway used as an example is the main source of public collective dose from the operations of UK nuclear industry, which is consumption of fish containing radiocaesium due to discharges from the Sellafield reprocessing plant in Cumbria. Estimates of the collective doses from fish and shellfish consumption are reported annually[3]; in 1981 they amounted to 130 man-Sv to the UK population and 150 man-Sv to the population of other European countries. Most of the collective dose is due to consumption of fish. Discharges of radiocaesium from Sellafield have been reduced significantly in recent years due to the use of zeolite in the fuel element storage ponds; the use of zeolite is optimised using cost-benefit analysis on the basis of ICRP principles[4]. The detriment is currently costed for this purpose by using a fixed value of unit collective dose. This has the advantage that the value of the detriment can be linearly related to the quantity of the radiocaesium discharged; it is unnecessary to take additional account of how the dose is distributed individually. In order to illustrate the problems caused in using a detriment costing scheme which includes a consideration of individual dose distribution, a simple categorisation has been attempted of apportioning the collective dose into three categories defined according to the level of individual dose rate: $< 1\%$, 1-10% and $> 10\%$ of the ICRP-recommended dose limit for members of the public.

2. THE UPPER DOSE RATE RANGE, > 10% ICRP DOSE LIMIT

It is difficult to assess precisely how much collective dose falls into this range but existing critical group data provide a basis for judging its importance. The dose rates in this range are similar to those received by members of both of the fish-consuming critical groups which are regularly assessed in relation to Sellafield discharges. Those critical groups are characterised by the source of the fish - the Cumbrian coast or the commercial landings at the Irish Sea fishing ports of Fleetwood and Whitehaven. The mean dose rates to these critical groups in 1981 were[3] 7.4% and 11.1% respectively of the ICRP dose limit, based on a group of 15 observations with a mean consumption rate of 100 g d^{-1} in the first case and a group of 20 observations with a mean consumption rate of 360 g d^{-1} in the second.

The collective dose to these groups was thus 0.017 man-Sv. As the dose rates given are means around the 10% level, many of these people will not have received > 10% of the ICRP dose limit. On the other hand, those actually exposed represent a larger group (but not excessively larger) than the observations on which the mean consumption rates are based. However, it is clear that the collective dose received at this individual dose rate level represents a very small fraction of the total collective dose. It is unlikely that in any variable valuation scheme the cost of the detriment at this individual dose level would be so much higher that this fraction would represent a significant proportion of the total.

3. THE RANGE 1-10% ICRP DOSE LIMIT

3.1 Irish Sea Landings

The fish landings from the Cumbrian coastal area near to Sellafield are small, being estimated to be a maximum of 20 t $year^{-1}$. Using the data given in section 2 for the critical group who consume this fish, the total collective dose, using an edible fraction of 0.5, is only 0.1 man-Sv. The effect of this very small fraction of the total collective dose on the overall detrimental costs using a variable valuation scheme may therefore be neglected.

The major problem in this assessment occurs in estimating the fraction of Irish Sea fish, from the commercial ports, consumed at rates such as to give doses in the range 1-10% of the ICRP limit. This needs a detailed knowledge of the mix of fish at different concentrations consumed by each person, as well as a detailed knowledge of the distribution of consumption rates within the population. The mix of fish will vary with seasonal and distributional factors and will in general be unknown to the consumer, ruling out estimation using a simple survey. For the present illustration, the simplifying assumption has been made that at a given port of landing the fish are uniformly contaminated, diluted as appropriate by fish from other sea areas, so that people can be categorised for dose rate on the basis of consumption rate alone.

Using the dose rate data given in section 2 for the critical group of commercial fish consumers, individuals would have received doses > 1% of the ICRP dose limit in 1981, if they had consumed more than 32.5 g d^{-1} Irish Sea fish from the commercial ports. The total fish landed at Fleetwood in 1981 was 11 867 t, 65% of which was from the Irish Sea. An average consumption rate for these landings of more than $32.5/0.65 = 50$ g d^{-1} will lead to dose rates > 1% ICRP limit. This conclusion is based on the assumption that fish from outside the Irish Sea contain comparatively negligible amounts of radiocaesium and is justified on the basis that the adjustment to this rate would only be a few percent even if all other fish landed at Fleetwood were assumed to come from Scottish waters. Effectively, all the fish landed at English and Welsh ports on the Irish Sea other than Fleetwood, is from the Irish Sea so that no dilution factor needs to be applied. A similar procedure has been followed for Irish Sea fish landed at ports in the Isle of Man, Northern Ireland and Scotland.

Further problems arise in assessing the distribution of people's fish consumption rates. This would require a large random survey over a wide range of the population, addressing a different objective from that relevant to habits surveys, which are biased towards those most exposed, in order to define a critical group. The only data presently available on a sufficiently large scale, which were used here, are from the National Food Survey[5]. However, this addresses foodstuff acquisition rates by household, not consumption by individual. So the following assumptions were made: first, that storage, e.g. of bulk purchases in freezers, has no effect (in practice this could lead to overestimates of consumption rates);

secondly, that the edible fraction of the fish purchased is 0.75 (this is higher than the fraction given above and allows for the large amount of fish bought in filleted and prepared forms); thirdly, that the average UK household consists of 2.8 individuals[5]. The distributions for the important categories of fish showed that consumption rates in excess of 32.5 g d^{-1} account for only about 39% of fish consumed; for the increased consumption rate of 50 g d^{-1} appropriate to the diluted Fleetwood landings, this fraction reduces to 19%.

Table 1 shows that using these assumptions, 32% of Irish Sea fish landed in 1981 could have given doses at > 1% of the ICRP limit. It is estimated that Irish Sea landings gave a collective dose of 48 man-Sv; thus about 15 man-Sv were received in this individual dose rate category.

Table 1 The fraction of Irish Sea fish landings by the UK giving > 1% of the ICRP limit, 1981

Port of landing	Landings from the Irish Sea (t)	Percentage giving > 1% of the ICRP limit	Landings giving > 1% of the ICRP limit, (t)
Fleetwood	7 754	19	1 473
Other English and Welsh ports	1 441	39	562
Isle of Man ports	2 809	39	1 096
Northern Irish ports	16 731	39	6 525
Scottish ports	1 611	0	0
Total	30 346		9 656

3.2 Landings from Scottish Waters and Further Afield

The mean ^{134}Cs and ^{137}Cs concentration in fish from the Minch, an important area of supply off the west coast of Scotland, were 1.2 Bq kg^{-1} and 30 Bq kg^{-1} respectively in 1981[3]. Thus, even when neglecting any dilution from further afield, a consumption rate of over 300 gd^{-1} is needed to achieve a dose rate of 1% ICRP limit. Consumption rates at these levels apply to very few individuals, and the collective dose received at these rates will be negligible compared with the total. The collective dose received from fish landed from Scottish waters and further afield was therefore effectively all received at individual dose rates < 1% ICRP limit. It should be noted that effectively all consumers in continental European countries would also receive doses at this low rate; fish from any of their landings from the Irish Sea would be diluted by those from other sea areas.

4. CONCLUSIONS

There are difficulties in applying a detriment valuation scheme which depends on the level of individual dose to members of the public. The first major problem is in assessing the mix of different sources of supply which are of different concentrations, so as to to determine each individual's dose rate; this becomes impracticable for a large number of people, even if the mix can be traced. The second major difficulty is in defining the detailed consumption rate distribution, which requires a large random survey; some data exist, and have been used here, but the different objectives introduce unquantifiable error. Further difficulties arise if the valuation of detriment is extended to a true optimisation, which requires iteration using an appropriate model of these variables. Any uncertainty in realism introduced may reduce confidence in the results of such an optimisation.

In any case, for this important example of collective dose received by the public, only about 15 man-Sv out of the UK total of 130 man-Sv for 1981 were received at a level > 1% of the ICRP limit; most of the 15 man-Sv would have been received at a level only slightly in excess of 1% of this limit. This distribution will move towards even lower individual doses when the Sellafield Site Ion-Exchange Effluent Plant (SIXEP) begins to remove radiocaesium from the fuel element storage ponds. At these levels of individual dose it is doubtful if any non-linear risk-aversion component of the detrimental cost will be significant. Thus a valuation of detriment which is linearly related to collective dose and radioactivity discharged is still justifiable, for this and examples of lesser importance.

5. REFERENCES

1. ICRP (1977). Recommendations of the International Commission on Radiological Protection. Annals ICRP 1 (3). Pergamon Press, Oxford, 53 pp. (ICRP Publ. (26)).

2. ICRP (1983). Cost-benefit analysis in the optimization of radiation protection. Annals ICRP 10 (2/3). Pergamon Press, Oxford, 74 pp. (ICRP Publ. (37)).

3. Hunt, G. J. (1983). "Radioactivity in Surface and Coastal Waters of the British Isles, 1981", Aquat. Environ. Monit. Rep., MAFF Direct. Fish. Res., Lowestoft, (9).

4. Hunt, G. J. (in press). "Experience of ALARA in controlling radiocaesium discharges to the Irish Sea", Proceedings of the 2nd European Scientific Seminar on Radiation Protection Optimisation, Luxembourg, 8-9 November 1983.

5. MAFF, National Food Survey Committee (1983). "Household Food Consumption and Expenditure: 1981", HMSO London, 232 pp.

INVESTIGATION OF INDIVIDUAL RADIATION EXPOSURES: COMPARATIVE USE OF INTERVIEW AND LOGGING TECHNIQUES IN HABITS SURVEYS OF THE CUMBRIAN COASTAL FISHING COMMUNITY

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INTRODUCTION

The International Commission on Radiological Protection (ICRP) recommends the selection of appropriate critical groups as a basis for the assessment of dose to individual members of the public. This procedure is used by the MAFF Directorate of Fisheries Research in the assessment of individual exposure from discharges of radioactivity into the marine environment. Surveys of people's habits and consumption rates provide the basic data for the selection and combination of these critical groups, and are directed towards those who are likely to be the most highly exposed. Usually, quantitative information is derived from talking to fishermen, their families, friends and customers. Verification of this information is important, especially for the more significant exposure pathways. To enable such a verification an alternative method was devised, based on individuals weighing and recording details of their consumption of seafood, a technique known as a logging or diary study. This paper describes a comparison of the interview and logging techniques for two important pathways those via fish consumption and via mollusc consumption by the fishing community in West Cumbria, near the British Nuclear Fuels Limited (BNFL) Sellafield works.

METHODS

The two information collection techniques were applied to a common group of consumers.

Interview Technique

Consumers in the West Cumbrian fishing community who had previously been identified as critical group members[1] were revisited. Following our standard procedure[2], each person was asked to estimate the quantities of all sea-food eaten during the preceding twelve months. Factors of particular importance were the source of supply, seasonal availability and method of preparation as well as frequency of consumption and weight of individual portions. Wherever possible sample portions were weighed during the interview or an estimate was made with the aid of photographs. Each interview would typically last an hour, providing data on not only the critical group consumer but also their families. This method primarily relies on the skill of the interviewer in terms of judgement and interpretation; thus people's memories sometimes had to be guided to giving a representative account of their regular consumption rather than what that person could eat if a species were available ad libitum. Formal questionnaires were not used as they offer too rigid a format. Instead, the conversation concentrated on the most significant pathways, namely consumption of fish, crustaceans and molluscs. Sources of supply were divided into (a) within 15 km of the Sellafield pipeline (b) outside the pipeline area, but still within the northern Irish Sea and (c) convenience seafoods. A species list was also used to check that the person had not omitted to mention some species which were only eaten very occasionally. The data obtained were reduced using our standard procedure[2].

Logging Technique

After the interview, the cooperation of a representative of the household was requested in recording all the seafood eaten for a two-week period during each

quarter for the next calendar year. The logging study covered 12 people out of the original 21 people who had constituted the two critical groups[1]. Each household recorded the weight of seafood portions just before cooking, with comments such as how much was eaten by each person, source of supply etc. Records were kept of all seafood eaten including shop-bought wet and convenience products. Regular visits were made to participants' homes during the survey; people were asked to comment on what they had written down compared with what they had stated at the interview. In addition, people were asked to complete a questionnaire asking whether the consumption rates recorded during the two weeks were considered typical for that quarter. Examples of factors affecting the results given were illness, availability of species, weather conditions, engine breakdown, adverse tidal conditions and periods of absence. However, the majority of people thought their consumption pattern was typical of that quarter; thus the data have been extrapolated linearly.

The log records were analysed to tabulate individual edible flesh weights for the different species eaten by each consumer and to convert the data to consumption rates for each logging period for fish, crustaceans and molluscs coming from the same three sources as in the interview survey.

RESULTS

Due to limitations of space, only the results for fish and molluscs within 15 km of Sellafield are presented here.

In comparing results from the two techniques it is to be noted that the periods to which the interview and logging studies apply are different. Thus there are two variables to consider, a genuine change in people's habits and a possible difference between the two survey methods. Precautions were taken to detect any changes in habits, the problem being discussed with the consumers during the logging period; it is thus considered that a valid comparison can be made.

Table 1 Consumption rates (g d^{-1}) of fish and molluscs caught within 15 km of Sellafield as obtained during quarterly logging compared with those from interviews

Observation number	1st Qtr	2nd Qtr	3rd Qtr	4th Qtr	Mean	Interview
Fish consumers						
1	0.0	78.1	96.0	30.2	51.1	54.0
2	0.0	113.3	128.9	NR	60.6	61.6
3	28.4	24.3	58.7	46.9	40.3	34.9
4	0.0	194.5	68.2	81.9	86.2	155.5
5	48.6	20.3	54.3	77.0	50.1	34.9
Mollusc consumers						
6	10.3	12.1	18.2	10.1	12.7	10.4
7	0.0	0.0	32.4	12.1	11.1	5.2
8	10.3	0.0	10.1	8.1	7.1	5.2
9	10.3	26.7	0.0	6.1	10.8	5.2
10	0.0	0.0	0.0	0.0	0.0	0.0
11	0.0	0.0	0.0	0.0	0.0	0.0
12	0.0	0.0	0.0	0.0	0.0	0.0

NR = No return: mean estimated on basis of statement that fish eaten in this quarter were caught more than 15 km from Sellafield.

Table 1 presents the consumption rates for each of the two methods for the twelve persons who took part in both techniques. The persons represented by observations 1, 2, 3, 5 and 6 thought their consumption rate to be the same as that of the previous year: for them there is good agreement between the results of logging and interview (statistically, by paired t-test, $p = 0.30$). The others stated that their habits had changed. The person represented by observation 4 had been regularly supplied with fish by a local fisherman during the preceding year, but not during the logging year when he had to supplement his supply with fish from beyond the 15 km area. For observations 7, 8 and 9 variations in consumption rate between the quarters were attributable to periods spent away from home. In the case of the people identified as observations 10, 11 and 12, who were members of the previously identified critical group for molluscs, their habits had clearly changed. This was alleged to be due to concern about radioactive contamination, serving as an example of one way in which people's habits may be influenced.

CONCLUSIONS

Both techniques have advantages and disadvantages. These are summarised in Table 2, from which we conclude that the two techniques are complementary.

Table 2 Advantages and disadvantages of interview and logging techniques

Aspect addressed	Interview	Logging
Timescale	Quick guidance on overall picture of individual annual consumption rates for the preceding year	Individual annual consumption rates require four quarters of data before calculation of consumption rate for the current year
Effort	Annual visit	6-10 visits per year
	Data rapidly analysed	Data analysis more time-consuming
Objectiveness	Relies on the skill of the interviewer and judgement of the interviewee	Relies on a person meticulously following the instructions
Cooperation	Given verbal cooperation, all data compiled by the interviewer	Requires person to fill in forms
Accuracy	Only verbal or visual estimate of portion weights often possible	All portions weighed before cooking
	No data other than interview record	Questionnaire gives guidance when converting data to quarterly consumption rates
Comprehensiveness	Indicates consumption of species which may only take place for a short time	Species may not be consumed during logging period

Thus a combination of both methods is recommended, whereby interviews provide the bulk of the data and identify those people who are potential critical group members, and logging is used on those people to help confirm the interview data. The combined information can then be used for the final selection of critical groups for subsequent dose assessments and assist in selection of the appropriate environmental monitoring.

REFERENCES

1. Hunt, G. J., Hewett, C. J. and Shepherd, J. G. (1982). "The identification of critical groups and its application to fish and shellfish consumers in the coastal area of the North-East Irish Sea." Hlth Phys. 43 (6): 875-889.

2. Leonard, D. R. P., Hunt, G. J. and Jones, P. G. W. (1982). "Investigation of individual radiation exposure from discharges to the aquatic environment: techniques used in habits surveys." Proc. 3rd Int. Symp: Soc. Radiol. Prot. 2: 512-517.

LIMITATION DES DOSES EN RADIOPROTECTION OPERATIONNELLE DANS UNE USINE DE RETRAITEMENT

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La COGEMA, en fonction de l'expérience acquise lors du fonctionnement des usines actuelles, a défini des principes directeurs qui permettent de satisfaire aux recommandations de la CIPR n°22.

- garantir le respect des limites de dose,
- éviter l'emploi de sources d'exposition inutiles,
- constituer un moyen pour le contrôle opérationnel du retraitement afin que les doses qui en résultent soient les plus faibles que l'on puisse raisonnablement atteindre, compte tenu des aspects économiques et sociaux,
- constituer un cadre plus général pour garantir que ces doses peuvent se justifier en termes d'avantage qui n'aurait pas pu être obtenu d'une autre manière.

Les problèmes se situent à différents niveaux :

- exploitation normale,
- opérations d'entretien courant,
- interventions pour réparations, modifications ou à la suite d'incidents.

L'optimisation des doses suppose un choix face à tous les problèmes qui se posent et en particulier la recherche d'une méthodologie spécifique pour les interventions pour lesquelles les risques sont les plus importants.

ANALYSE CRITIQUE DE LA JUSTIFICATION D'UNE INTERVENTION

Le déclenchement des interventions lorsqu'il n'est pas lié à un choix délibéré de modification, d'amélioration ou de construction, est consécutif à la mise en évidence d'une anomalie par des moyens d'exploitation ou par des mesures radiologiques.

Les moyens d'exploitation susceptibles de mettre en évidence une anomalie sont, outre le dispositif de surveillance radiologique, les moyens classiques de maintenance du matériel : entretien, essais, visites périodiques, etc...

L'évolution radiologique des locaux en général est suivie grâce aux contrôles périodiques des surfaces et à la dosimétrie de zone.

L'efficacité de ces opérations de dépistage est parfois réduite par l'existence de situations radiologiquement confuses, sources d'irradiation, ambiance élevée, sources de contamination. Ce contexte est souvent propice à une augmentation des nuisances dans une installation, génératrice d'une diminution de la surveillance qui entraine alors une augmentation des risques.

Il est donc important d'éviter ce genre de situation et cela doit se faire par une suppression organisée des sources.

- La formalisation de la recherche doit être stricte :

  . Suivi périodique des mesures sur des points préalablement matérialisés localement.

  . Exploitation et archivage des résultats au sein de documents tenus à jour par le Service de Protection contre les Rayonnements (SPR).

Il convient de rechercher pour les points sensibles à haut débit de dose une automatisation de la mesure.

- L'information est rendue systématique par la création d'un registre de liaison SPR-Exploitant.

- L'action curative doit être vérifiée sur ce même registre.

Il faut également prendre des mesures pour éviter les dérives possibles sur ce genre de disposition.

La reconnaissance radiologique doit fournir une bonne évaluation des nuisances, notamment la séparation des composantes irradiation et contamination, et se traduire par l'établissement de cartes précisant les "points chauds".

En ce qui concerne les modifications, améliorations que l'on peut qualifier d'actions volontaires leur déclenchement doit être précédé de l'analyse critique de la justification de l'opération. Quelques exemples montrent que ce type de démarche a parfois conduit à l'abandon de l'intervention.

Cette analyse doit, quoiqu'il en soit, être une préoccupation permanente et se faire par concertation entre les responsables d'exploitation et le SPR avec la participation des spécialistes concernés (unités d'intervention - services techniques, etc...).

PREPARATION DE L'INTERVENTION

La prise en compte des interventions au niveau de la conception intervient pour beaucoup dans l'importance des difficultés rencontrées. Les résolutions adoptées ultérieurement à cet étage sont même considérées comme relativement peu déterminantes quant à la réduction des coûts radiologiques.

Il est ainsi nécessaire de :

- constituer des équipes conceptuelles regroupant les spécialistes des différentes parties,

- suivre la conception sous l'angle intervention avec l'assistance de correspondants représentant les futurs intervenants,

- établir en cours de conception, des modes opératoires avec estimation des doses associées pour la mise en compétition des différentes solutions possibles.

Le support documentation doit comprendre les plans de base regroupés en un lieu unique.

L'historique doit regrouper tous les éléments "du vécu" d'une partie d'installation.

Certaines situations peuvent ne pas avoir été prévues par le mode opératoire ou par les autorisations accordées.

En face d'une opération dangereuse ou non réglementaire, sauf si le maintien en l'état représente un risque immédiat et manifeste plus important que la poursuite, la situation est rapportée au Chef d'Installation et au responsable local de radioprotection qui établissent un nouveau mode opératoire.

TECHNIQUES D'INTERVENTION

1 - LES MATERIELS

On peut distinguer 4 familles de matériel d'intervention :

. l'instrumentation de radioprotection,
. l'outillage et les moyens d'intervention,
. les moyens de protection contre l'irradiation,
. les moyens de protection contre la dissémination de la contamination.

Il est apparu nécessaire de créer des disponibilités. En effet, si les matériels peuvent être différents selon les interventions, il se trouve tout de même qu'une partie importante de ces moyens, d'utilisation courante, peut constituer un lot de base d'intervention qui présente l'avantage :

- de réduire les déchets,
- de réduire les délais de mise en place,
- d'inciter à la mise en place de ces moyens ou de la faciliter par le fait de leur existence même et de leur disponibilité.

L'instrumentation de Radioprotection

Des modules d'intervention pour assurer un certain nombre de fonctions, ont été conçus.

- Observation du déroulement des opérations sur écran "télé",
- Liaisons phoniques,
- Télédosimétrie,
- Surveillance atmosphérique,
- Mesure du débit de dose.

L'outillage et les moyens d'intervention

Les critères d'adaptation aux interventions doivent être pris en compte au niveau de l'approvisionnement :

Vulnérabilité à la contamination.
Facilités de décontamination.
Gains de temps en cours d'utilisation.

Ce matériel, comprend des :

- appareillages de travail à distance,
- appareillages d'utilisations courantes (ventilateurs, filtres...),
- moyens phoniques de liaison entre opérateurs,
- caméras de télé, récepteurs, magnétoscopes,
- moyens télécommandés (transports de caméra ou d'instruments de mesure...).

Les moyens de protection contre l'irradiation

Les besoins en écrans de protection sont fortement fonction des situations rencontrées, toutefois il est très avantageux de pouvoir disposer du matériel de base, d'utilisation fréquente.

Les moyens de protection contre la dissémination de la contamination

Le confinement dynamique et statique de la contamination en pratique est effectué la plupart du temps par l'intermédiaire de sas construits à la demande autour des lieux d'intervention.

Il a été nécessaire de s'orienter vers la spécification et l'approvisionnement de sas réutilisables : parois souples et ossature télescopique, panneaux rigides.

2 - L'EVACUATION DU MATERIEL ET DES DECHETS CONTAMINES

Il faut particulièrement veiller à éviter les difficultés résultant des relations entre unités : coordination des parties, étiquetage, élingage, transports....

PRINCIPES DE PROTECTION

Il est absolument indispensable qu'un effort soit fait pour l'agencement général, le balisage, la signalisation, ainsi qu'en ce qui concerne leur respect. Les protections individuelles doivent être adaptées aux risques.

En particulier les règles de mise en place des protections collectives et individuelles doivent être telles qu'il ne puisse y avoir :

- de transferts de contamination surfacique,
- de transferts de contamination atmosphérique (confinement statique ou dynamique toujours performant),
- d'exposition à un risque d'inhalation même inférieur aux limites d'incorporation,
- de dose intégrée d'une manière non concertée.

Par ailleurs suivant la catégorie d'agent une formation spécifique du personnel est nécessaire :

Formation du personnel intéressé,
Formation des décontamineurs,
Formation ponctuelles pour une intervention particulière.

Il est également souvent utile de renforcer, au niveau local, la formation théorique et surtout pratique des agents de radioprotection, compte tenu du rôle important qu'ils doivent jouer dans le cadre des interventions.

INDIVIDUAL MONITORING OF EXPOSURES TO EXTERNAL RADIATION - OBJECTIVES, METHODS, INTERPRETATION

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INTRODUCTION

According to ICRP /1/ and restricting to exposures to external radiation, monitoring means the measurement of radiation for reasons related to the estimate or control of exposure to radiation, and includes the interpretation of the measurements. Monitoring of the workplace is used as measure to show, that the working environment is acceptable and is routinely repeated to justify continued operations. Individual monitoring can confirm that the general conditions at the workplace are under satisfactory control and establish whether operational changes have improved or worsened the working conditions. These statements are fairly general and require only, that it be possible to compare the various monitoring results to each other with acceptable uncertainty and to relate them to the established limits of radiation exposure. To achieve this, it is practical to state the monitoring results as dose equivalents. But this requirement is not sufficient to secure the unambiguity of measurements. It is the conditions of measurements which are crucial, i.e. the point of measurement in the exposed object, the object itself (person or phantom), the irradiation conditions. What makes the influence of these conditions so important is the attenuation of the incoming radiation in the exposed object and even more so the scattering and other secondary radiation contributions arising in the object, which in the case of neutron radiation, for example may be responsible for the greater part of the dose equivalent in an anthropoid object.

Situations like the one just described are familiar in physics, and hence a rationale for metrology i.e. the quantitative assessment of physical phenomena, has long been provided for. This is a necessary prerequisite in order to quantitatively formulate the relations between various physical phenomena. The basis of this rationale is the quantity concept (cf. e.g. /2/). Its consequent application to particular problems of measurement leads to the definitions of so-called "qualified quantities" in which standardized conditions of measurement are included. Examples are the "effective voltage" of an alternating current, with ionizing radiation, the "effective dose equivalent" /3/ and the various "dose indices" /4/. However, neither the effective dose equivalent nor the dose equivalent index can in general be measured (cf. par. 5 of /1/ and /5/), nor have other suitably defined qualified quantities of the dose equivalent category which would be measurable either directly or indirectly hitherto been defined at an international level. Failure to do this has severely hampered the evaluation of international intercomparisons of individual monitors /6/ and has provoked useless discussions on the interpretation of measurements /7/.

OBJECTIVES

Before discussing the selection of appropriate measurable quantities the objectives of individual monitoring should be clear. These are by no means self-evident, as a review of the literature reveals. According to the ICRP (par. 35 of /1/), individual monitoring means the carrying out of measurements with equipment worn or carried by

workers. It is further stated, that the primary purpose of such monitoring is to obtain an estimate of the mean dose equivalents in significantly exposed tissues and of the effective dose equivalent. This information is then considered useful in limiting radiation doses to individual workers and in demonstrating compliance with the system of dose limitation recommended by the ICRP, and with authorized limits. The ICRP further states that this goal is usually achieved indirectly by the measurement of quantities that can be related to the deep and shallow dose-equivalent indices. The expedience of this inference has been seriously questioned /8/.

More modestly and more realistically, the IAEA /9/ states that the main objectives of monitoring the exposure of radiation workers are to keep the exposure as low as reasonably achievable and to ensure that the authorized limits are not exceeded. Many more vague formulations of the objectives can be found in the literature, including the assessment of radiation risks, whatever that may mean. But such statements evade any interpretation in terms of measurable quantities.

Before methods and procedures can be discussed however, a clear decision on the objectives is needed. Here, it should be remembered that the ICRP based the system of dose limitation on a linear relation between the dose equivalent and the probability of detrimental effects for the so called no-threshold stochastic effects. This results in a substantial overestimate of this probability for low dose, low dose-rate, low LET irradiation. The question then arises of what profit can be obtained from an explicit determination of mean organ dose equivalents and of the effective dose equivalent in the case of radiation exposures well below the limits? It appears more reasonable, to confine oneself to measurements of operational quantities, which ensure observance of the limits, permit control of radiation exposure and are suitable for record keeping.

Methods and Procedures

In principle it is possible to assess all relevant organ doses and the effective dose equivalent from measurements performed on the body of the exposed person /10/, notwithstanding the particular exposure conditions. The necessary expenditure in achieving this goal depends on the acceptable uncertainty and is in general considerable. Hence, the rather intricate procedure required will be restricted to special applications. Radiation protection routine calls for simple, cheap but reliable methods.

Extensive Monte Carlo calculations in the anthropoid MIRD phantom /11/ for photons /12/ and neutrons /14/ have demonstrated that satisfactory observance of the system of dose limitation can be ensured by "measuring" the dose equivalents near the most exposed parts of the body surface at a depth of 70 μm (limiting the dose equivalent in the basal layer of the skin), of 3 mm (limiting the dose equivalent in the lens of the eye) and of 10 mm (limiting the effective dose equivalent). These dose equivalents at the three depths on the most exposed parts of the body of the exposed individual would then constitute the qualified quantities for individual monitoring.

But is this really what can be done routinely in individual monitoring and what has been done up to now? Without going into the details of existing procedures, it can be taken for granted that individual monitors are calibrated in one way or another, then are exposed on the individual, are read after exposure, and the readings then multiplied by the calibration factor to yield what at least in the routine case are considered as the monitoring results.

Conceding that the dose equivalent response function of most individual monitors is not perfect, either with respect to energy dependence or to directional dependence, and considering that the influence of secondary radiations arising in the exposed individual may be substantial, it is essential to choose an adequate phantom and appropriate irradiation conditions for calibrating. Following the lines prescribed by the ICRU /4/, a spherical phantom 30 cm in diameter, and of standardised soft tissue composition (density 1 g cm^{-1}) may be chosen. For this phantom, too, extensive Monte Carlo calculations have been performed (e.g. /12, 13/). The MIRD and the sphere results demonstrate that the sphere can be considered as a sufficient approximation for whole body exposures.

In order to arrive at a realistic procedure for individual monitoring, the following steps should then be taken.

1. Definition of specified depth dose equivalents in the ICRU sphere as qualified quantities. This includes irradiation in a field, which is homogeneous over the volume of the sphere in the absence of this sphere ("expanded field"). The point of measurement is on the central axis for parallel radiation. Specified depths are 10 mm and 0,07 mm, with a possible 3 mm in addition. For monitoring the exposure of extremities, e.g. of the hands, special concepts and techniques must to be used.
2. Establishment of derived limits for these operational quantities. For the exposure model considered here, these may be taken as numerically equal to the effective dose equivalent limit, the skin dose equivalent limit and the eye dose equivalent limit, respectively.
3. Derivation of instrument response requirements. For this purpose, the instrument and the sphere have to be considered as a single system. The spectral response function (in relation to a free field quantity causing this response) for this system designed to measure one of the above quantities is then prescribed to a good approximation by the established relation of this quantity to the field quantity (e.g. exposure, neutron fluence). This relation is given in the form of a conversion function. The analogous procedure is to be followed for tailoring the instrument's directional response function.
4. According to the definition of the quantities to be measured, calibration would in principle have to be performed on the ICRU sphere. However, this phantom is rather awkward for practical work on account of its curved surface. The choice of more conveniently shaped phantoms has been shown to be acceptable /16/.

INTERPRETATION

When an individual monitor is exposed on the body of a person, the irradiation geometry is, of course, somewhat different from that for which the quantities are defined, namely the ICRU sphere. Consequently, multiplication of the reading with the calibration factor does not lead back exactly to the sphere quantity as defined. In general, it is not possible to correct for this systematic error, as this would require knowledge of the spectral and angular distributions in the radiation field during exposure, information, which is rarely available. Moreover, the labour necessary to ascertain such a correction is scarely worthwhile, considering the other admissible and inevitable uncertainties.

The result arrived at by the simple procedure just described is appropriate for record keeping and serves well to attain the objectives of individual monitoring as outlined above. For legal reasons, and in order to avoid any judicial argument, it appears expedient to separately define the quantity arrived at by multiplying the monitor's reading and the calibration factor and to name it the "administrative individual dose" or "individual dose for record-keeping" and have this term legalized.

This paper is an attempt to treat individual monitoring in a conceptually consistent way, leading to simple, practicable methods and procedures of measurement. The results arrived at by following this path show the usual measurement uncertainties, but these do not originate in conceptual vagueness nor in incomplete definitions of quantities.

A final word of caution: none of the results of the current routine monitoring programs yield effective dose equivalents or mean organ doses, rather they demonstrate the effectiveness of radiation protection measures and compliance with the system of dose limitation. Any estimate of these dose equivalents would be affected by large uncertainties which could hardly be assessed. This together with the reservations pointed out in the section on "objectives", makes routine monitoring results, which are chiefly in the low dose range, quite questionable as input data for epidemiological studies.

REFERENCES

/1/ ICRP Publication 35: Ann. ICRP 9(4), Pergamon Press, Oxford 1982
/2/ Allisy, A.: Scuola Internazionale di Fisica "Enrico Fermi". Corso LXVIII in Metrology and Fundamental Constants, p.14. North Holland Publ. Co. 1980
/3/ ICRP Publication 26: Ann. ICRP 1(3), Pergamon Press, Oxford 1977
/4/ ICRU Report 33: International Commission on Radiation Units and Measurements, Washington 1980
/5/ Operational Quantities for Use in External Radiation Protection Measurements - An Investigation of Concepts and Principles. Commission of the European Communities. Radiological Protection-27. EUR 8346 EN. Luxembourg 1983
/6/ Fourth Information Seminar on the European Radiation Protection Dosemeter Intercomparison Programme. Bilthoven NL, 25-27.10.1982
/7/ Wagner, S.R.: Health Phys. 40, 569 (1981). Sims, C.S.:ibid.p.570.
/8/ Johns, T.F.: Rad. Prot. Dos. 4, 66 (1983)
/9/ Basic Requirements for Personal Monitoring. Safety Series No. 14, 1980 Edition. International Atomic Energy Agency, Vienna 1980
/10/ Siebert, B.R.L. et al. Phys. Med. Biol. 28, 521 (1983)
/11/ Snyder, W.G. et al. J. Nucl. Med. 10, Supplement No. 3 (1969)
/12/ Kramer, R. et al. GSF-Report S-885. München- Neuherberg 1982
/13/ Morhart, A. et al. GSF-Report S-880, München-Neuherberg 1983
/14/ Nagarajan, P.S. et al. Proc. Fourth Symp. on Neutron Dosimetry. EUR 7448 EN, Vol. I, p. 49. Luxembourg 1981
/15/ Chen, S.Y., Chilton, A.B.: Rad. Res. 78, 335 (1979)
/16/ Schwartz, R.B., Eisenhauer, C.M.: Procedures for Calibration Neutron Personnal Dosimeters. NBS Special Publication 633. Washington 1982

EIN KONZEPT ZUR INKORPORATIONSÜBERWACHUNG DES PERSONALS VON KERNKRAFTWERKEN

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Grundlagen und Anforderungen

Der gesetzliche Rahmen für die Inkorporationsüberwachung in Kernkraftwerken wird in Deutschland durch die Strahlenschutzverordnung (StrlSchV) sowie die Richtlinie für die Physikalische Strahlenschutzkontrolle (§§ 62 und 63 StrlSchV - Bek. d. BMI v. 05.06.1978 -, GMBl 1978, Nr. 22, S. 348) abgesteckt. Danach sind Inkorporationsmessungen erforderlich, wenn die Möglichkeit einer Inkorporation oberhalb 1/20 der Grenzwerte der Jahresaktivitätszufuhr nach § 52 StrlSchV nicht auszuschließen ist. Für die in diesem Zusammenhang vordringlich zu betrachtenden Nuklide Co-60 bzw. Cs-137 entspricht dieses 1/20 - Kriterium Aktivitäten von 1,1 µCi bzw. 1,8 µCi. Die zur Inkorporationsüberwachung verwendeten Meßmethoden sollen gewährleisten, daß eine Inkorporation zu Beginn einer Überwachungsperiode oberhalb 1/20 der Grenzwerte auch an deren Ende noch erkannt wird.

Zwar ist es nicht vorgeschrieben, daß die Messungen im Kraftwerk durchzuführen sind, jedoch besteht ein starkes Eigeninteresse des Kraftwerksbetreibers daran, nachweisen zu können, daß Inkorporationsdosen klein im Vergleich zum 1/20-Kriterium sind. Dieser Nachweis hat besondere Bedeutung für Fremdpersonal, das normalerweise nur für kurze Zeit der Strahlenüberwachung im Kraftwerk unterliegt und bereits mit einer Inkorporationsvorbelastung eintreffen kann.

Auch im Hinblick auf mögliche Zwischenfälle ist es sinnvoll, über eigene Möglichkeiten für eine schnelle Inkorporationsmessung zu verfügen, obwohl solche Situationen in der Vergangenheit nur äußerst selten aufgetreten sind.

Für das Überwachungskonzept wurden folgende Randbedingungen festgelegt:

Alle Personen, die erstmalig zur Arbeitsaufnahme im Kontrollbereich das Kraftwerk betreten bzw. nach Beendigung der Arbeit die Anlage verlassen, sind auf das Vorhandensein einer Inkorporation zu überprüfen. Dabei muß das Meßkonzept zu Spitzenzeiten einen hohen Personendurchsatz ermöglichen.

Daneben soll im Bedarfsfall eine Einrichtung zur quantitativen Bestimmung der Körperaktivität vorhanden sein.

Inkorporationen im Kraftwerk sollen möglichst umgehend festgestellt werden. Hierzu können ohnehin vorhandene Meßeinrichtungen mitbenutzt werden.

Angesichts der Gegebenheiten im Kraftwerk wird als Inkorporationspfad nur die Inhalation betrachtet, im Regelfall wird die Aktivität der Lunge gemessen.

Überwachungskonzept

Aufgrund der genannten Anforderungen wurde folgendes Überwachungskonzept entwickelt (vergl. Abb. 1)

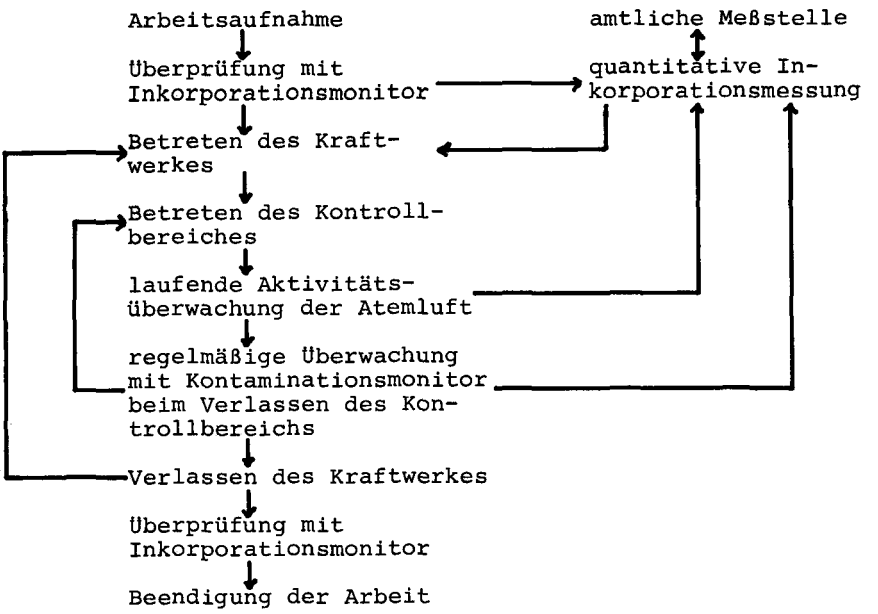

Abb. 1: Ablaufschema der Inkorporationsüberwachung

Bei Arbeitsaufnahme (z. B. Revisionsbeginn) wird beruflich strahlenexponiertes Personal mit einem integral messenden Inkorporationsmonitor auf das Vorhandensein einer Inkorporation kontrolliert. Diese Überprüfung wird im Rahmen der ohnehin erforderlichen strahlenschutztechnischen Abwicklung (Abgabe des Strahlenpasses, Aufnahme der Personalstammdaten etc.) durchgeführt. Als Schwelle für die Ja-/Nein-Entscheidung wird dabei eine Aktivität von 1/100 der Grenzwerte der Jahresaktivitätszufuhr (entsprechend 200 nCi bezogen auf Co-60) angesetzt. Die zum sicheren Nachweis dieser Aktivität erforderliche Meßzeit soll 10 sec nicht überschreiten, damit die gerade bei Revisionsbeginn in großer Zahl ankommenden Personen (bis zu einigen 100 pro Tag) zügig abgefertigt werden können.

Die Überschreitung des Schwellwertes zieht unverzüglich eine quantitative Bestimmung der Lungenaktivität mit Hilfe eines nuklidspezifisch messenden Inkorporationsmeßplatzes nach sich. Für diesen Meßplatz wird eine Nachweisgrenze von höchstens 1/500 der Grenzwerte der Jahresaktivitätszufuhr, entsprechend 40 nCi bezogen auf Co-60, gefordert. Dabei ist angesichts der vermutlich geringen Meßfrequenz (Messung nur nach zuvor festgestellter Erfordernis) eine Meßzeit von bis zu 10 min akzeptabel.

Zum frühzeitigen Erkennen von Inkorporationen werden die Ergeb-

nisse anderer im Kraftwerk vorhandener Meßgeräte herangezogen:

Im Kontrollbereich werden an ausgesuchten Arbeitsplätzen mobile Meßgeräte eingesetzt, die neben der Ortsdosisleitstung kontinuierlich die Aerosol- und die Jod- oder Edelgasaktivität der Luft messen. Mit diesen Geräten läßt sich jederzeit die zugeführte Aktivität auf Grenzwerte hin überwachen. Bei Überschreitung eines Grenzwertes entsprechend dem für den Inkorporationsmonitor aufgeführten Kriterium wird ebenfalls der nuklidspezifische Inkorporationsmeßplatz eingesetzt.

Außerdem lassen sich die Kontaminationsmonitore am Kontrollbereichsausgang zur Inkorporationskontrolle nutzen. Die Gamma-Empfindlichkeit der Kontaminationsmonitore gestattet den Nachweis einer Körperaktivität von größenordnungsmäßig 200 nCi, bezogen auf Co-60. Dieser Wert entspricht der Alarmschwelle des Inkorporationsmonitors. Bei Alarm eines Ausgangsmonitors wird man nach erfolglosen Dekontaminationsmaßnahmen deshalb auf eine Inkorporation schließen und weitere Nachforschungen mit Hilfe des Inkorporationsmeßplatzes anstellen.

Nach Beendigung der Tätigkeit im Kraftwerk werden alle Personen im Zuge der Entlassungsprozedur (Entgegennahme des Strahlenpasses etc.) erneut mit einem integral messenden Inkorporationsmonitor kontrolliert.

Durch das engmaschige Netz von Überwachungsstellen ist ein rechtzeitiger und hinreichend empfindlicher Nachweis von Inkorporationen und damit die Erfüllung sowohl der gesetzlichen als auch der betrieblichen Anforderungen gewährleistet.

Wird eine Inkorporation von mehr als der Hälfte des Grenzwertes der Jahresaktivitätszufuhr festgestellt, so wird zusätzlich eine amtliche Meßstelle eingeschaltet.

Ausführung Des Konzeptes und Eigenschaften Der Eingesetzten Inkorporationsmeßgeräte

Das geschilderte Konzept wird zur Zeit im Kernkraftwerk KRB II realisiert.

Als integral messender Inkorporationsmonitor kommt dort ein Gerät mit zwei in Brusthöhe übereinander angeordneten großflächigen NaJ(Tl)-Detektoren zum Einsatz. Die Detektoren sind in ein Stahlgehäuse eingebaut, mit 30 mm Blei gegen die Umgebungsstrahlung abgeschirmt und auf den Lungenbereich kollimiert. Die nachgeschaltete Elektronik gestattet es, den registrierten Energiebereich, die Meßzeit sowie die Alarmschwelle frei zu wählen.

Eine zu überprüfende Person tritt dicht vor die Stirnfläche der Detektoren und startet die Messung durch Handbetätigung zweier Kontakte. Die Kontakte müssen während der gesamten Meßzeit geschlossen bleiben und sind seitlich am Gehäuse so angeordnet, daß eine definierte und reproduzierbare Meßgeometrie erzielt wird.

In KRB II werden zwei baugleiche Geräte im Eingangsbereich installiert, und zwar je eines für kommendes und gehendes Personal. Die Konstruktion der Monitore mach ein Eingreifen des Strahlenschutzpersonals - abgesehen von administrativen Hinweisen (Sicht- und Sprechkontakt zu den Aufstellungsplätzen) - nicht erforderlich, wodurch der

Strahlenschutz entlastet und der Meßablauf beschleunigt wird. Es wird erwartet, daß max. 200 Personen pro Monitor und Stunde überprüft werden können.

Das Ergebnis des Messung ("Inkorporation: Ja/Nein") sowie System- und Fehlermeldungen werden auf einer separaten Signaleinheit im Strahlenschutzschalter des Eingangsbereiches zur Anzeige gebracht. Das zuständige Strahlenschutzpersonal erteilt daraufhin entweder die Freigabe oder veranlaßt eine weitere Messung mit dem nuklidspezifischen Inkorporationsmeßplatz. Eine Dokumentation der Ergebnisse der Monitormessungen erfolgt nicht.

Nach Angaben des Geräteherstellers wird bei einer Meßzeit von 10 sec eine Nachweisgrenze von $<$ 50 nCi, bezogen auf Co-60, erreicht. Die im vorhergehenden Abschnitt genannten Anforderungen werden damit erfüllt; vermutlich wird man zur Überprüfung der festgelegten Alarmschwelle mit kürzeren Meßzeiten als 10 sec auskommen.

Der nuklidspezifische Inkorporationsmeßplatz befindet sich ebenfalls im Eingangsgebäude und damit in unmittelbarer Nähe des auch die Monitormessungen überwachenden Strahlenschutzpersonals. Er ist für sitzende Personen ausgelegt und als seitlich und rückwärtig mit aktivitätsarmen Stahl abgeschirmter Stuhl ausgeführt.

Mit dem Stuhl verbunden ist eine Vorrichtung zur Aufnahme eines Detektors, die es gestattet, den Detektor in einer reproduzierbaren Position vor der Person zu fixieren. Der Detektor ist mit Blei abgeschirmt und kann wahlweise auf die Lunge, den Magen-Darm-Trakt oder den (Ganz-)Körper kollimiert werden. Auch eine Messung der Schilddrüse kann ermöglicht werden.

Es wird ein Reinstgermanium-Detektor verwendet, mit dem die Nuklidzusammensetzung von Inkorporationen detaillierter untersucht werden kann, als dies mit den bislang fast ausschließlich verwendeten NaJ(Tl)-Detektoren möglich ist. Damit wird das Ziel verfolgt, zu einer genaueren Kenntnis über die Herkunft und Ursache der Inkorporation zu gelangen, um in Zukunft noch gezielter entsprechende präventive Maßnahmen ergreifen zu können.

Zwar ist die Nachweisempfindlichkeit des Germanium-Detektors deutlich niedriger als die eines für Inkorporationsmessungen üblicherweise eingesetzten NaJ(Tl)-Detektors, jedoch wird dieser Nachteil in bezug auf die Nachweisgrenze durch die bessere Energieauflösung zu einem großen Teil kompensiert. Der Gerätehersteller gibt die Nachweisgrenze für Co-60 und Cs-137 mit ca. 10 nCi bei einer Meßzeit von 10 min an, womit die Anforderungen an den Meßplatz erfüllt sind.

Mit Hilfe des dem Detektor und der Analogelektronik nachgeschalteten Vielkanalanalysators und Kleinrechners wird aus der gemessenen Körperaktivität nach Eingabe des in der Regel hinreichend genau eingegrenzten Inkorporationszeitpunktes die zugeführte Aktivität bestimmt. Anschließend werden den so festgestellten Bruchteilen der Grenzwerte der Jahresaktivitätszufuhr die gleichen Bruchteile der Grenzwerte der Körperdosis nach § 49 StrlSchV zugeordnet. Unter Zugrundelegung der angegebenen Nachweisgrenzen kann man auf diese Weise noch Inkorporationsdosen von wenigen mrem ermitteln. Soweit erforderlich, können diese Dosen anschließend zur Bilanzierung der Dosiswerte aus äußerer und innerer Exposition herangezogen werden.

PROBLEMS OF RADIATION PROTECTION IN X-RAY DIFFRACTION APPARATUS

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Analitical X-ray diffraction equipments are widely used in scientific research laboratories and in education establishments. Nowdays, these apparatus are being used also in industrial laboratories for quality and process control purposes.

The progresses in electronics and in technology have brought in the recent past to a great sophistication of X-ray diffraction equipments.

Since X-ray equipments are likely to be dangerous if mishandled, particular attention has been recently paid by the manufacturers in designing such apparatus.

While the industrial applications of X-ray diffraction, which require routine procedures, are carried out with modern equipments with high standard safety characteristics, in the research work the apparatus and their mode of operation may be different. In fact frequent modifications are required in order to obtain better performances suitable for the specific scientific work. This fact may represent a real health hazard for the operator particularly for the very high exposure rates close to the windows of the tube-shield. A certain number of accidents have been reported in the literature with consequent severe damage to the operator (1-3).

The present communication deals with an investigation on several (about 30) X-ray diffraction apparatus utilized in typical research laboratories, in order to obtain more detailed informations for a better guideline regarding some aspects of radiation safety.

The results of our analysis has allowed to locate and to weight different kinds of potential radiation sources.

Although the radiation produced in these apparatus is comparatively soft (5-50 keV) and the primary beam dimensions are rather small (because collimated with pinholes or slits), the intensity is very high.

The primary beam is particularly dangerous, since it represents the most concentrated radiation source.

A rough evaluation ($\pm$25%) of the primary beam exposure rate is given by the following empirical formula (14):

$$X = \frac{50\ ZVI}{74\ D^2}\ R/sec$$

where
$\dot{X}$ = exposure rate
Z = atomic number of the target material
V = tube voltage (kV)
I = tube current (mA)
D = distance from the focus (cm)

This formula is obeyed in the range 25-80 kV using 1 mm Be-filtration; other filters may change the exposure rate by a factor of 2.

It is to be noticed that without collimator, the primary beam diverging from the window of the tube-shield has an apex angle of 15-30° (depending on the dimension of the window hole). It follows that, close to the tube shield window, the exposure rate amounts to some thousands of R per second. A few seconds exposure to the hands of the operator exceedes the value of 60 rem, the maximum permissible annual dose, for this part of the body by causing serious somatic damages and reaching, some times, the amputation of the fingers.

Among the potential sources of radiation the direct beam is the most hazardous one.

The direct beam irradiation may be caused by:
- shutters failure;
- operator's error;
- inadeguate instruction to the operator.

Other sources of potentially hazardous radiation are analyzed, even though the dose rate involved are very much smaller than those associated with the primary beam.

1. The leakage through gaps and cracks of shielding can reach values of the same order of magnitude of the primary beam. Such leakage radiation was never observed in our surveys and natural background values were measured around the shielding.

2. The faulty coupling between the tube-shield windows and cameras or collimators is observed mainly when the X-ray generator and the diffraction accessories are produced by different manifacturers. We have found such situation in several installations with an exposure rate of about hundreds mR/h. The problem is particularly serious not only for the relatively high exposure rate but also because the beams are very narrow and irregularly directed.

3. The secondary emission from the sample and other materials is uniformly distributed around the generator bench and presents a relatively low value of exposure rate (some mR/h).

4. The X-rays diffracted from the sample and other materials can reach values of the order of some R/h. The diffracted beams are particularly insidious since they exhibit small sections, irregular shape and can be oriented in whichever direction. Both secondary emission and diffracted rays constitute a real problem in the open cameras. These cameras are, therefore, the most hazardous

apparatus.

5. Exposure rate from 0 to 100 mR/h was found to be due to the nature and the number of the beam shutters and to their running mode. This hazard is typical of shutters of the old tube-shields.

Particular attention has been dedicated to the monitors used in this analysis. In order to obtain a reliable survey the monitor must have good sensitivity to low energy X-radiation (down to 10 keV and a good energy independence at very low photon energies. Moreover, because of the difficulty to locate the very narrow and irregularly directed escaping beams, a detector with a small window area is desiderable.

In our survey we have employed the following detectors:
- an IONEX 2500/3 dosimeter with a thin window Nuclear Enterprises ionization chamber 2532/3;
- a Victoreen survey meter mod. 471 with a thin window ionization chamber, used to calibrate the Eberline Geiger-Müller probe, mod. HP-190 and the Eberline NaI (Tl) scintillation counter mod. PG-2.

The experience gained in this survey tell us that the exposure rate measurement around the X-ray diffraction equipments can give rough estimates of dose rate, by employing the commercially available monitors. Therefore it is necessary to choose a suitable detector when actual measurements dose rate are required.

Moreover it is our opinion that the X-ray diffraction equipments are among the most potentially hazardous pieces of laboratory apparatus. They must be carefully overseen and handled with caution.

Bibliography

1) R. JENKINS, D.J. HAAS: Incidence, detection and Monitoring of radiation from x-ray analytical instrumentation. X-Ray Spectometry 4, 33, 1975.

2) G.S. LINSLEY: A review of the problems of safety control in x--ray diffraction and spectrometry. Proc. of the 4th Intern. Congress of IRPA, Paris, April 1977, Vol. 2, 511.

3) K.D. STEIDLEY, R.J. STABILE, L.Y. SANFILIPPO: A case hystory of severe radiation burns from 50kV x-rays. Health Physics 40, 399, 1981.

4) BO LINDELL: Occupational hazard in x-ray analitical work. Health Physics 15, 481, 1968.

TWENTY YEARS OF TRITIUM INTERNAL MONITORING IN JAPAN ATOMIC ENERGY RESEARCH INSTITUTE

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INTRODUCTION

The Japan Atomic Energy Research Institute(JAERI) was established in 1956, and research activities at the Tokai Research Establishment were started in 1958. Among many facilities at Tokai Research Establishment, the following facilities are concerned with tritium exposures:

a. Tritium production laboratory; recently, for the fusion research and development program, production of tritium on 100 Ci level are carried out. During the purification process, tritium was handled as a gas. At the present time, annual production of tritium is several hundred curies.
b. Heavy water reactors, JRR-2 and JRR-3; At Tokai Research Establishment, there are two heavy water research reactors, that is, JRR-2 and JRR-3. The concentration of tritium in the reactor water increases every year. Presently, the concentration is about 1 mCi/ml for the both reactors. During the past 20-23 years of reactor operation, leakage of the heavy water occurred occasionally. The largest leakage rate was about 3.5 l/week for JRR-2.
c. Van de Graaf accelerator; tritium is frequently used as a target. In most cases, the amounts of tritium handled are on the curie level. Leakage of the tritium sometimes occurs in the target room.
d. Other facilities; low level contamination with tritium is also observed occassionally in the waste treatment plant.

In JAERI, the chemical form of tritium handled is tritiated water or tritium gas. We have no experiences of internal contamination other than by DTO and HTO. At Tokai Research Establishment, the number of radiation workers is about 1,800(in 1983), of which about 220(in 1983) are workers handling tritium or working under a tritium atmosphere.

MONITORING OF TRITIUM AT WORKPLACES AND FOR INDIVIDUALS

Monitoring of The Workplace

Several techniques are used to monitor tritium concentrations. An air flow type ionization chamber monitor is used for continuous measurement of total tritium (tritium gas and tritiated water) in air at the workplace. In addition to this continuous monitoring method, cold trap method and adsorption method are also used. When contamination is suspected, surface monitoring or exhaled air monitoring for the workers is of course carried out.

Protection of The Worker from Internal Contamination

An air supplied suit with a full-face mask is used at high level air contaminations. For the workers, when internal contamination is suspected, tritium concentration of the exhaled air is then measured for preliminary individual monitoring. If an internal contamination of more than 10 mrem is estimated by the exhaled air measurement, then the worker is sent to Bioassay Division, where the internal contamination is monitored in detail.

INDIVIDUAL MONITORING OF INTERNAL CONTAMINATION

The Bioassay Division carried out individual monitoring, on a routine and special monitoring basis.

Routine Monitoring

The purpose of routine monitoring is to check for the presence of significant internal contamination. At present, for tritium, 10 mrem (50 years committed dose) is adopted as the significant exposure level for internal contamination.

The number of workers who have a risk of tritium contamination is about 220, however, except for the test production group, the level of tritium handled is very low, and furthermore, the workplace monitoring is carried out very strictly. For these reasons, individual monitoring is not carried out on a routine basis for all workers who have a risk of tritium contamination. At present, about 60 workers are selected as subjects of routine monitoring. This routine monitoring is carried out every three months.

Special Monitoring

The purpose of special monitoring is to estimate the body burden, and a committed dose if necessary. This special monitoring is carried out for the following workers,

a. workers found by routine monitoring to have a significant contamination,
b. workers who have had an accidental intake of tritium,
c. workers who are suspected of having a tritium contamination.

Method of Bioassay for Tritium

In general, urine is taken as a sample. Water contained in exhaled air and saliva is often used as the sample. For urinalysis, 0.2 ml of urine is mixed with 15 ml of scintillator solution(PPO-POPOP/toluene-ethanol). After several hours, the tritium activity is measured with a low background liquid scintillation counter for 10 to 30 minutes. For the exhaled air, sample water is collected by condensing exhaled air directed through small test tube cooled by an ice bath. In the case of 10 minutes counting, the detection limit is about 10 pCi/0.2 ml for urine or other water samples.

DOSE CALCULATION

Based on the recommendation of ICRP, the committed dose equivalent is calculated by the following formula,

$$D = 51.2 \cdot (\mathcal{E}/m) \cdot q_0 \cdot \int_0^{50y} R(t)\, dt \qquad \text{----------(1)}$$

where D: committed dose equivalent(rem/50 years),
$\mathcal{E}$: effective energy of tritium(0.006 MeV),
m: mass of the critical organ(body water, 43 Kg),
q_0: amount of intake(μCi)
R(t): a retention function($e^{-0.693t/T}$), and
T: effective half-life(day).

Based on the assumption that the tritiated water is distributed uniformly in the body water, the amount of intake, q_0, is calculated as

$$q_0 = U_0 \cdot m \qquad \text{----------(2)}$$

where U_0: initial concentration of tritium in the body water such as urine.
Combining equations (1) and (2),

$$D = 51.2 \cdot (\mathcal{E}/m) \cdot U_0 \cdot m \cdot (T/0.693) = 0.44\, U_0 \cdot T \qquad \text{----------(3)}$$

In equation (3), T is usually determined by follow-up analysis of the body water. If the follow-up is not possible, the ICRP recommended value of 10 days is used.

EXPERIENCE OF TRITIUM MONITORING FOR INTERNAL CONTAMINATION AT JAERI

Results of Routine Monitoring

At JAERI, routine monitoring of tritium by urine analysis started in 1970. The results of the monitoring are shown in Table 1.

Table 1 The results of routine monitoring at JAERI

Year	Number Monitored	Number found Contaminated*1
1970	38	0
71	55	4*2
72	52	0
73	61	0
74	105	1*3
75	98	1*2
76	175	0
77	206	0
78	211	0
79	182	0
80	207	0
81	228	1*4
82	254	0
1970-82	1872	7

*1: Significant exposure level is 10 mrem(committed dose)
*2: Van de Graaf accelerator
*3: JRR-2
*4: JRR-3

During the past 13 years, about 1900 measurements were carried out. As a results, only 7 subjects were found out to have significant contamination. All these subjects were sent to special monitoring.

Results of Special Monitoring

Special monitoring at JAERI for tritium started in 1963. The results of the special monitoring are shown in Table 2.

In the past 20 years, a total of 358 measurements were carried out. As a result, we found about 70% of the subjects to have tritium contamination. However, the evaluated committed doses are very small for most of the subjects, and a committed dose of over 250 mrem was observed only in one subject(380 mrem).

Biological Half-life of Tritium

In the past 20 years, 255 subjects were found to have tritium (tritiated water) contamination. The half-life of tritium retention was observed for 50 subjects by follow-up measurement. Careful investigation of their working conditions were performed, and 41 subjects were selected as having no repeated contamination of tritium. The observed half-lives for 41 subjects are shown in Fig.1.

As shown in Fig.1, the observed half-lives are in the range of 5 to 17 days, an average value is about 10 days. These findings are in good agreement with those reported by ICRP(4 to 18 days, a typical value is 10 days).

Because the observed cases are so few, both the seasonal and age variations of the biological half-life are not obtained from the data.

Table 2 The results of special monitoring at JAERI

Year	Number of Cases	Number of Subjects	No Contamination	Estimated Committed Dose (mrem)			
				< 10	10-250	250-1000	>1000
1963	1	1	0	0	1	0	0
64	4	6	6	0	0	0	0
65	4	80	22	49	9	0	0
66	8	18	2	16	0	0	0
67	9	55	30	20	5	0	0
68	6	22	0	9	13	0	0
69	7	26	4	15	7	0	0
70	8	33	6	12	14	1	0
71	6	23	5	6	12	0	0
72	3	8	6	2	0	0	0
73	1	3	0	3	0	0	0
74	4	13	5	7	1	0	0
75	3	4	0	2	2	0	0
76	1	3	0	1	2	0	0
77	2	5	0	1	4	0	0
78	2	9	5	4	0	0	0
79	5	35	8	26	1	0	0
80	4	4	3	0	1	0	0
81	3	8	1	3	4	0	0
82	1	2	0	1	1	0	0
Total	82	358	103(29%)	177(49%)	77(22%)	1	0

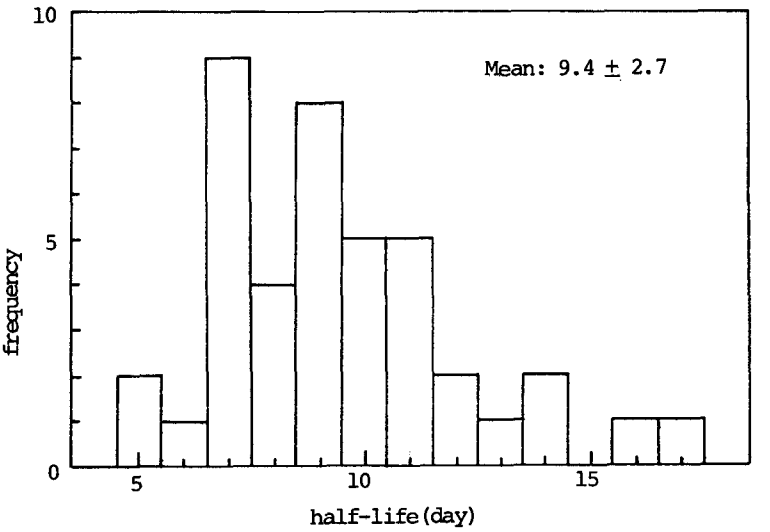

Fig.1 Biological half-lives of tritiated water observed for Japanese adults

PREPAREDNESS IN INTERNAL DOSE ASSESSMENT AT THE CENTRAL RESEARCH INSTITUTE FOR PHYSICS, BUDAPEST

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Introduction

The committed effective dose equivalent is the basic quantity in internal dose limitation. Therefore results measured in any other terms shall be converted into committed effective dose equivalent. The actually measurable quantities are the activities in whole body, in body tissues or in excreta. The main problem in the estimation of internal dose is now compare the measured quantity to the limited one. There are two possible ways to overcome this difficulty. One of them is the comparison of the measured data to the reference value derived from the ALI /Annual Limit on Intake/ the secondary limit recommended by the ICRP [1]. These ALI values are calculated from the committed effective dose equivalent limit by using the ICRP recommended Reference Man data, metabolic models and retention patterns. Thus numerous assumptions and simplifications are involved in this procedure. In the second method the activity time integral /U_S/ for every source tissue /S/ is first determined from the directly measurable quantity. The committed dose equivalent /$H_{50,T}$/ in tissue T can then be expressed as

$$H_{50,T} = \sum_S U_S \cdot \mathrm{SEE}\ /T \leftarrow S/$$

where SEE /T ← S/ is the specific effective energy originating from tissue S and absorbed in T. This latter procedure can be used in cases of higher internal contaminations and assumes the availability of sophisticated methods of measurement and calculations.

In our whole body counter laboratory severe efforts has been made in the last few years to be prepared for applying both procedures in the present and future monitoring programmes.

Instrumentation

The whole body counter of the institute was put into operation in 1964 and has reconstructed and improved in the last years. Presently, the device enables us to carry out whole body activity measurements in tilting chair, arc and one-, or two detector scanning geometries and activity distribution /organ-, tissue activity/ measurements by two detector profile and rectilinear scanning using different collimator systems [2].

The detector signals are processed by a multichannel amplitude analyser. Autonomous control electronics developed in the laboratory is used for controlling the detector movement and the measurement. This control is helped by a PDP 11/34 computer and necessary CAMAC electronics in the most complicated measurements [3]. This computer equipped with peripherials and having extended memory capacity is at our disposal for collecting and storing measured data, for quantitative evaluation of gamma spectrometric and activity distribution measurements, for metabolic model calculations and for dose estimation.

For calibration purposes different types of phantoms viz. Presdwood, BOMAB, REMCAL and a thorax phantom are used.

Methods

Computer programs COMPLIST [4] and DCOM 3 have been developed for the calculation of the expected organ or whole body activities vs. time after the single intake of a given radionuclide. The calculations can be performed using the models recommended by the ICRP [1] or any other metabolic models assuming first order kinetics. From these calculations the temporal variation of the derived reference levels /e.g. Derived Investigation Levels/ can be obtained for measurable quantities like activities in whole body, in organs and in excreta. These calculated results are also used in routine monitoring to determine the proper monitoring frequency taking into account the minimum detectable activities and many other factors.

If more precise dose assessment is necessary, then activity distribution and retention measurements can be carried out for the given individual case to determine the activity time integral U_S, as indicated above. The whole procedure can be followed in Fig. 1 where the necessary steps are shown in case of quantitative profile scanning. For computation of activity time integral from the measured count rate profiles, a set of programs has been worked out and used in different computers [5]. For decomposition program DECOMP /DAS2/ and WDEC were developed. These codes after proper calibration calculate the most probable organ activities by numerical solution of a linear equation system using the weighted least squares method. If activities in different organs have already been obtained at the times of measurements the retention functions for organs and the activity time integrals can be computed by the programs COMPFIT1, COMPFIT2, DCOM16, SUPEXP and INTIME. These programs fit the numerical solution of a differential equation system determined by an arbitrarily chosen compartment model to the organ activity values; or fit a sum of exponential functions; and/or calculate the numerical integral of a series of organ activity data.

Conclusions

The preliminary investigations and test measurements show, that organ activities can be determined by the methods available in the range of 1-10 kBq even in the case of overlapping organs, but very sophisticated calibration measurements are needed. According to the results obtained by investigations of accidental internal contamination cases, the doses estimated by the two ways mentioned above are found to be in good agreement in most cases [6] . Dose assessment based on sophisticated measurements of individual parameters are necessary only when the level of internal contamination and consequently the accuracy requirement is higher.

References

[1] ICRP Publication No.30, Ann. ICRP 2, 3/4 /1979/
[2] A. Andrási, Éva Beleznay, Report KFKI-1983-43 /in Hungarian/
[3] J. Urbán, A. Andrási, Report KFKI-1982-45 /in Hungarian/
[4] A. Andrási, Éva Beleznay, Gy. Kötél, Proc. of the Symposium on Biomedical Dosimetry: Physical Aspects, Instrumentation, Calibration, IAEA, Vienna, 1981, p. 159.

[5]A. Andrási, Éva Beleznay, Acta Phys. Acad. Scient. Hung., 52, 3/4 p. 303 /1982/
[6]A. Andrási, Éva Beleznay, Proc. of the XI. Regional Congress of IRPA on Recent Developments and New Trends in Radiation Protection /Vienna, Sept. 20-24, 1983/ /in press/.

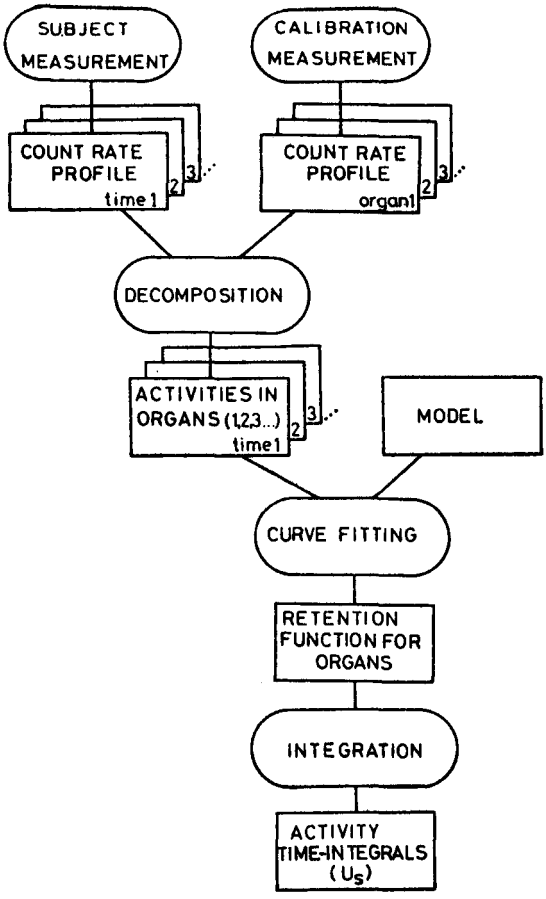

Fig. 1. Scheme of activity time integral determination by quantitative profile scanning.

A STUDY OF THE DECONTAMINABILITY OF SURFACE MATERIALS FOR USE IN NUCLEAR INSTALLATIONS

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Introduction

The use of surface materials allowing the easy removal of contaminations, which may occur during routine operation or during accidents is an important part of the efforts directed to the safety of nuclear installations. However, requirements and methods of test influencing the selection of such surface materials have always been subjects of controversial developments and discussions at national and international level. The Federal Institute for Materials Testing (BAM) has, at both levels, contributed to an improvement of the situation by an own research project and by participating in the troublesome work of relevant standardization. This work was furthered by the Federal Ministry of the Interior.

Terms and Definitions

Some of the more important terms to be used in the following need a careful definition, since they are often handled in an ambiguous way.

In practice contamination of a surface is simply defined as the presence of a radioactive material soiling the surface. According to this decontamination is the removal or reduction of the soiling radioactive material. Both definitions do not take into account the existence or strength of an interaction between surface material and radioactive material. As soon as the decontaminability of surface materials is considered, such interactions become the main point of interest and the above definitions are insufficient, if used in the context of a test method. All these methods are based on the following definition: Decontaminability is the ability of a surface material (being in contact with radioactive material)

to resist fixation

and (should a fixation have occured)

to allow easy reduction of the fixed radioactive material as a result of cleaning efforts.

From this definition the two components of decontaminability Contamination Resistance and Ease of Reduction are clearly visible. There have been repeated attempts to measure both components seperately, e.g. in the French standard method, but neither an adequate experimental method nor the necessity for doing so could be demonstrated. Since decontaminability is a specific quality of a surface material, care has to be taken, that the cleaning efforts mentioned above do not comprise single steps which tend to reduce fixed radioactive material in an unspecific manner by use of abrasive or corrosive agents.

The numerical description of decontaminability has mainly been done in terms of the decontamination factor DF, obtained after more or less well defined contamination and decontamination procedures.

Related terms of similar structure, e.g. residual activity fraction or percentage as well as decontamination percentage have also been used. The decontamination factor is defined as

$$DF = \frac{\text{initial activity}}{\text{residual activity}} = \frac{\text{initial pulse rate}}{\text{residual pulse rate}}$$

The initial activity as it is defined in practice and by most of the test methods is the activity brought into contact with the surface during contamination

Problems

A careful consideration of the term "initial activity" immediately leads to difficulties. In most cases only a small activity is finally fixed to the surface by various physical and chemical processes. The bulk of the activity remains in a state free of interactions with the surface and will not contribute to the residual activity. Concentrations being kept constant, the fixed activity is a constant for a specific surface material. An increase of the initial activity will then not result in an increase of the fixed activity or the residual activity. In the calculations, however, an increased initial activity leads to an increased DF-value. The contribution of the fixed activity, reflecting by quantity and behavior the specific qualities of the surface material, to the DF-value is in this way arbitrarily masked and diluted. The test method is loosing its discriminating power and will produce high DF-values for the whole range of surface materials. In this connexion it is unimportant, whether the contaminating material on the surface is a liquid or a salt crust, obtained by evaporation of a liquid. Expressed in terms of crystalline monolayers of the Co- and Cs-compounds used for contamination, three of the test methods offer between 1.6 and 14 monolayers for fixation. The highest value for a fixation measured in the author's laboratory was around 0.3 monolayers, but a material like this would never be considered for use in nuclear installations. Typical fixation values for surface materials of excellent decontaminability are two orders of magnitude lower.

In a test method the size of the initial activity, representing a quantity of the contaminating element (active plus inactive), is normally derived from other experimental parameters. For example, a certain minimum quantity of the contaminant solution is needed to cover a constant and sufficiently large area of the test specimen. Furthermore, care has to be taken, that sufficiently high residual count rates are obtained in case of surface materials of excellent decontaminability. The initial activity, therefore, is not a free variable which may be adjusted in accordance to the aspects discussed above. It would be more appropriate to consider, whether there is a real need for terms like decontamination factor and initial activity in the assessmant of decontaminability.

Another principal problem arises from the need for cleaning a test specimen prior to testing. It has been demonstrated by use of autoradiography, that dust particles and other types of dirt picked up by the test specimens during transport will spoil the test result, if not removed by an effective cleaning process. This has to comprise the use of cleaning mixtures for degreasing as well as wiping and

rinsing steps. It has been argued, that such efforts are not only far away from practice, but furthermore tend to change surface properties due to abrasion, chemical attack or pickup of solvents by the surface material. Such aspects are indeed of importance and should be carefully considered. The fact, that in practice surfaces are normally not cleaned with similar care cannot be a reason to accept dirt of unspecific composition and quantity as a component of a specific surface material to be tested. Abrasion can be avoided, otherwise wiping with soft cellulose tissues would decrease the decontaminability, which is not the case. Chemical attack by organic solvent mixtures used for degreasing is a possibility that would only be excluded, if the cleaning agents could be carefully adjusted to the chemical resistance of the material to be tested. This, however, is a difficult task and in a standard the choice of the cleaning agent should not be left to the judgement of the testing center. The use of detergent solutions for degreasing has been recommended, but there is experimental evidence, that detergent residues on the surface will also influence decontaminability results.

The choice of the decontamination agent for the test is a third complex of different approaches. In the French standard method the contaminated area of the test specimen is rinsed with 1.7 litres of water and no reduction of the fixed activity is expected as a result of this treatment. In Germany water is a favourite decontamination agent for a number of reasons and our own experiments indicate, that water is able to remove up to 90 % of the fixed activity, depending on the type of surface material. Other decontamination agents may be used instead of or in addition to water, but their effect is, as a rule, proportional to the effect of water and in some cases smaller than to be expected. This has been demonstrated by the systematic evaluation of the results obtained with the German standard method. This method applies water, an acid detergent solution and 1 m hydrochloric acid in three successive decontamination steps. - A German round robin experiment concerning the effect of three commercially available decontamination agents on four different contaminated surface materials resulted in a mean value of 4 for the factor by which the products were better than water. Extreme values for the factor were 0.6 and 16.

Proposed test method

Starting in 1980 Working Group 10 of ISO/TC85/SC2 first considered the current national standards for the test and assessment of decontaminability in search of a possible ISO standard method. The British and the German standard were then selected for comparison in international round robin experiments. The results for both methods were unsatisfactory due to their low interlab reproducability. The German method proved to be only slightly better than the British one. Therefore, it was decided to develop a compromise method from the best elements of both national methods. This compromise method was tested and improved in three round robin experiments with nine participating institutions from four countries. Finally a sufficient interlab reproducability was achieved and the main sources of error were identified.

In December 1983 a first draft proposal on the basis of the compromise method was completed and circulated among SC2-members to

collect comments.

In the following a short listing of the essential features of the proposed method is given together with a description of the advantages to be expected while applying the method:

1. Test specimen dimensions of 50 mm x 50 mm allow transport in slide storage containers without contact of the surface areas to be tested.
2. Separate application of ^{60}Co- and ^{137}Cs-contaminant solutions allows use of different types of detectors with solutions from a common source and increases the value of the information obtained.
3. Composition, production and storage of the decontaminant solutions are carefully described in order to minimize the contributions from this most important source of errors.
4. A four-step wiping and rinsing procedure on the basis of a petrol ether-isopropanol mixture and demineralized water was included to reduce the impact of dirt on the measurement results.
5. Demineralized water is applied as the sole decontamination agent for the sake of simplicity and the ease of waste management. Being the most widely used decontamination agent water is likely to be a good standard material for this purpose.
6. Contamination is done by help of a special holder for test specimen, allowing for the first time the contamination of a constant area by a constant volume of contaminant solution.
7. Five test specimens are decontaminated simultaneously in a cage-stirrer apparatus under identical conditions. Waste volumes and the time spent for a test are considerably reduced; the influence of differences between operators is brought to a minimum.
8. The terms decontamination factor and initial activity are no longer used. The measurement results are expressed in terms of Standardized Mean Residual Pulse rates (SMRP) for the single radionuclides and and of the Final Residual Pulse rate (FRP) for the combination of both radionuclides.
9. The assessment of the decontaminability or ease of decontamination is done on the basis of the FRP-values by means of a classification table which has been empirically compiled.

In the course of the round robin experiments some highly sensitive experimental parameters have been identified: pH-values and carrier concentrations of the contaminant solutions, effectiveness of the cleaning procedure, chemical resistance of a silicone rubber ring being in contact with the test specimen and the contaminant solution and the way of handling the apparatus used for contamination.

Variations in some other parameters were of minor importance: temperatur and volume of the contamination and decontamination agents, the length of time spent for contamination and decontamination, the stirrer speed of the cage-stirrer apparatus, the type of detector and the counting geometry used.

In a discussion of sources of errors it should be noted that inhomogeneities in a group of test specimen often occur to such an extent that experimental errors contribute only slightly to the deviations obtained.

DECOMMISSIONING OF A TUNNEL USED AS A SHIELD FOR A HIGH-ENERGY PROTON BEAM

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INTRODUCTION

When high-energy particle accelerators and related facilities are designed, considerable effort goes into ensuring that they are adequately shielded and that any radioactivity induced by high-energy particle interactions can be safely handled. A radiation problem that will arise in the future but which receives relatively little attention at the design stage is the eventual disposal of the large quantities of low-level radioactivity, particularly in the shielding, that will have to be faced when the accelerator comes to the end of its useful life. We recently had a chance to study this problem when it became necessary to remove an obsolete proton beam tunnel to make way for the injector for the electron positron collider (LEP) being constructed at CERN. This gave an opportunity to study the residual radioactivity that remains in shielding material after the more usual isotopes have had time to decay away and which may ultimately determine the storage time that will be necessary before those materials can be freely recycled.

HISTORY OF THE TUNNEL

The plan of the tunnel is shown in Fig. 1. It was constructed in 1966 to house an extracted proton beam from the CERN 28 GeV/c proton synchrotron. This beam, together with a suitable target and decay path, enabled a neutrino beam to be produced for physics experiments. This so-called neutrino tunnel was used for about ten years during which time approximately $3 \cdot 10^{19}$ high-energy protons were consumed. The tunnel has long since been stripped of all reusable beam elements and equipment and only the shielding remained. This shield consisted of a corrugated steel cylinder of 2.2 m internal diameter surrounded by earth shielding of dimensions shown in Fig. 1. The floor of the tunnel was a layer of concrete filling the bottom segment of the otherwise cylindrical tunnel.

RADIOACTIVITY IN THE SHIELD

Although the tunnel had not been used for more than six years, dose rates up to 60 μSv/h were still measurable near the former target position. This radiation was due to residual activity in the steel lining, the concrete floor and the earth shielding. Samples of these materials, taken from different parts of the tunnel, were analysed by gamma-ray spectrometry. The isotopes found and an estimate of the quantities of activity involved are given in Table 1. The distribution of the isotopes longitudinally, in the concrete on the tunnel floor, is shown in Fig. 2 and the radial distribution, through the earth shield, is given in Fig. 3. Another long-lived isotope of possible interest that will be formed in hadron interactions in the shield is tritium. Measurements were made of soluble tritium in the earth shield and a total activity of 100 MBq was estimated. The activity remaining in the steel of the tunnel lining was mainly Co-60 but also still contained about 20 MBq of Mn-54 (with a 290 day half-life).

DISCUSSION

The isotopes that have been found fall into two distinct categories, those produced by high-energy particle spallation interactions with nuclei (i.e. Na-22 and H-3), and those produced by thermal neutron activation of trace elements (Co-60, Cs-134, and Eu-152). It is of interest to note that Cs-134 was found in concrete but was barely detectable in the earth and that the Co-60 and Eu-152

concentrations are more than an order of magnitude lower in concrete than in earth, relative to the Na-22 concentration. The lateral distribution of the Na-22 in the shield is consistent with an attenuation mean free path of the high-energy particles of 100 g/cm^2 whereas the thermal neutron activation reduces by nearly 2 orders of magnitude over the first 100 g/cm^2 of shield.

By comparing the number of protons used in the facility with the activity induced in the earth shield, a rough estimate can be made of the number of isotope-producing interactions per proton used. These estimates are given in Table 1. Although there are many factors influencing this relation the figures obtained could serve as a guide in making crude estimations under other conditions.

The trace elements in question appear to be widely distributed and vary with type of geological formation (Ref. 1). Typical values of the abundance of the trace element isotopes with high neutron capture cross-section are given in Table 2, together with the appropriate cross-section. The radiological importance of the induced isotopes depends on many factors. Some judgement can be made based on the gamma dose-rate constant of the isotope (dose rate at 1 m from 1 MBq) and its radiotoxicity classification. For comparison purposes the data in the Swiss national radiation code (Ref. 2) has been used where isotopes are classed from 1 to 9 in decreasing radiotoxicity order. As can be seen, Eu can be considered more radiologically important than the other isotopes even though its gamma dose rate will be relatively low. As regards tritium, its external dose rate is negligible and its radiotoxicity is class 9. Coupled to the fact that it could be extracted from the earth by washing, it is not expected that this isotope will contribute to the long-term radiation problems of activated shielding materials.

DISPOSAL OF RADIOACTIVE MATERIALS

In the case being considered the absolute levels of the radioactivity remaining in the shield are relatively low and the average level in the shield is below that which need be considered as radioactive (Ref. 3). Even so, the active part of the earth and concrete were separated and buried in the shield of a new neutrino tunnel what was being constructed on the other side of the PS machine.

The above exercise could be considered a pilot experiment in the disposal of unwanted beam-tunnel shielding. Present-day extracted proton beams are at least an order of magnitude higher in intensity than they were ten years ago and the disposal problem may not be so easily solved in the future.

CONCLUSIONS

The measurements that have been made of the long-lived isotopes found in the shield of a high-energy particle beam line indicate that activation of trace elements in earth, and to some extent in concrete, could control the long-term radioactivity in these materials. The most important of these isotopes that was found in the earth was Eu-152 which has a 12.7 year half-life. The experience gained in decommissioning of the neutrino beam tunnel suggests that attention should be paid to possible long-term problems with this activity when selecting materials for shielding high-energy particle accelerators.

REFERENCES

1. Encyclopaedia Universalis (Tome 7), p. 600 (1970)
2. Ordonnance concernant la protection contre les radiations, Le Conseil Fédéral Suisse (Article 63, 111).
3. In Ref. 2 (Article 3, 73, 111).

Table 1

Isotopes found and their activity (in MBq)

Isotope with half-life / Material	Co-60 5.3 y	Cs-134 2.1 y	Eu-152 12.7 y	Na-22 2.6 y
Steel	100	-	-	2.5
Concrete	15	5	30	110
Earth	40	<2	130	85
Atoms/proton (in earth)	3.6×10^{-4}	-	1.5×10^{-3}	4×10^{-4}

Table 2

Trace elements in shielding materials

Isotope found	Co-60	Cs-134	Eu-152	Na-22
Parent isotope	Co-59	Cs-133	Eu-151	-
Abundance ppm	23	7	0.5	-
Thermal neutrons X section barn	19	28	5900	-
γ dose at 1 m μSv/h/MBq	0.35	0.24	0.14	0.32
Toxicity group	8	6	5	6

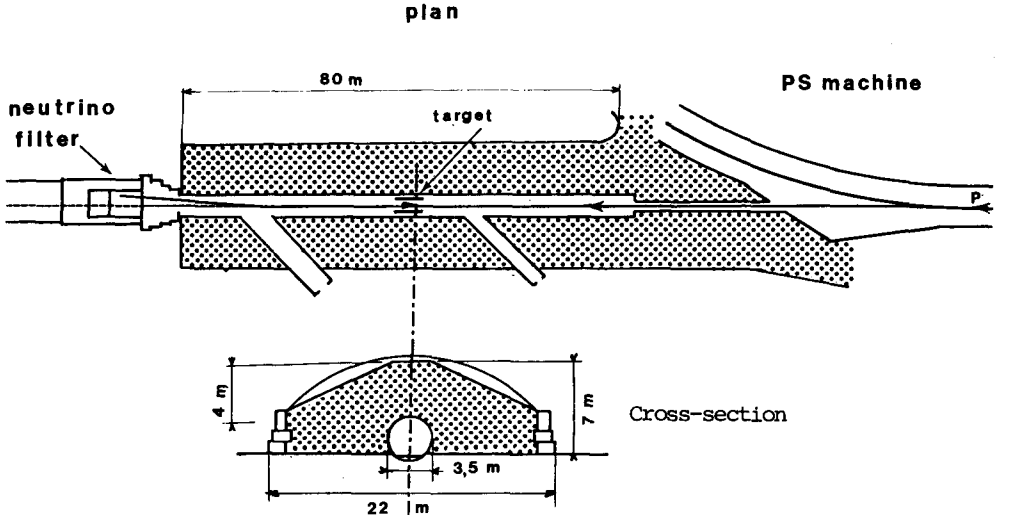


Fig. 1. Layout of neutrino tunnel shield

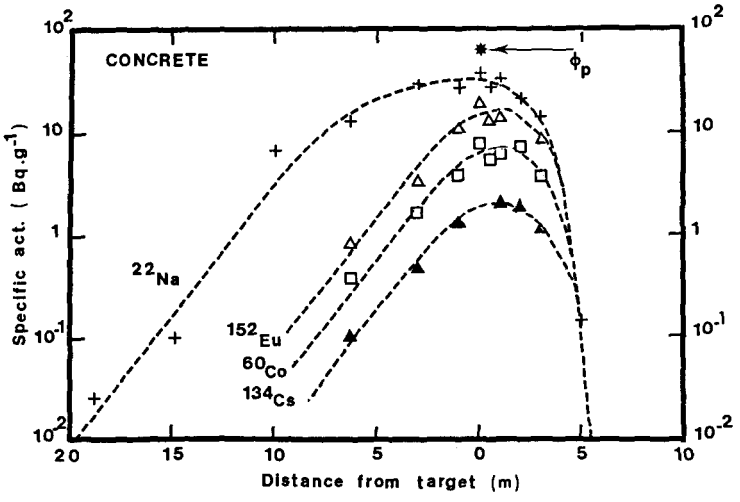
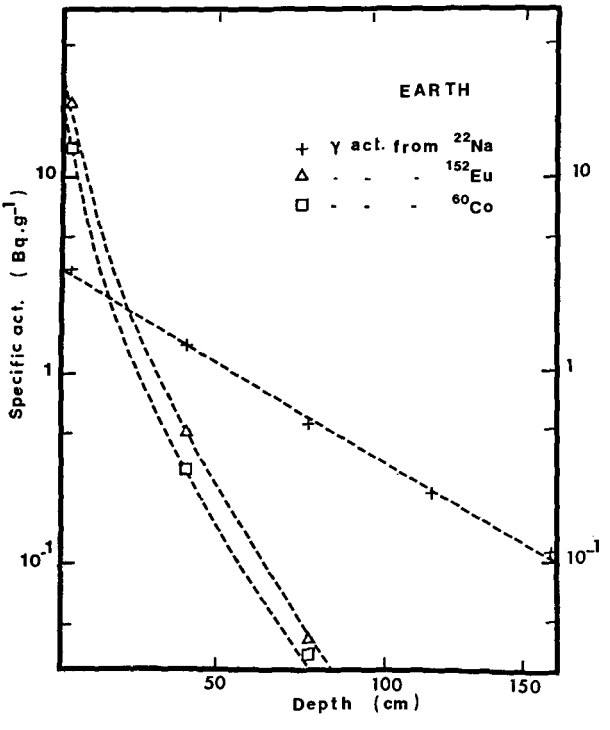


Fig. 2. Longitudinal distribution of isotopes along the concrete floor of tunnel.

Fig. 3.

Radial distribution of isotopes through the earth shield.



RADIATION PROTECTION OF PERSONNEL IN DECONTAMINATION OF HOT LABORATORIES

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1. Introduction

In the Tokai Research Establishment of JAERI, there are two hot laboratories for post-irradiation examinations of nuclear fuels and materials. One is used for monitoring Magnox fuels from the gas cooled reactor (Calder Hall Type Reactor) and irradiated materials, and the other for post-irradiation examinations of nuclear fuels from light water reactor power plants.

In cutting the fuels or polishing the samples for metallurgical examination, radioactive particulates containing fission products are dispersed in the cell, and surfaces of the equipment, floor and walls are radioactively contaminated. In the respective cells, at the times of periodical maintenance of the in-cell equipment and repair of defective machines, decontamination is made for minimizing personnel exposure doses.

This paper describes the radiation monitoring and protection in cell decontamination, and discusses the data obtained in radiation monitoring.

2. Radiation Monitoring

Decontamination in the cells is usually conducted in the following steps: 1) removal, storage or disposal of irradiated fuel or material, cuttings etc. by remote operation, 2) wiping off cuttings etc. by manipulator with rags, 3) measurements of the exposure rate and surface contamination density in the cell with a manipulator, 4) repeated decontamination by hands with personnel entry into the cell, and 5) repeated measurements of the exposure rate and surface contamination density.

Low levels of the exposure rate and surface contamination density are set for decontamination operation so that the personnel can conduct maintenance or repair work in light protective outfit and their exposure doses may be as low as reasonably possible. In many cases, about 2×10^{-5} $\mu Ci/cm^2$ is set for the surface contamination density and a few mR/h for the exposure rate, except for hollows and crevices of the equipment and gaps between the floor and machines which are hard to decontaminate.

In decontamination, the procedure is first planned and then the methods and the reduction levels of exposure rate and surface contamination are determined. Prior to the operation, a radiation work permit is issued, and the techniques of radiation monitoring and personnel protection including the outfit and the preventive measures for spread of contaminants are determined.

During the decontamination work with personnel entry into the cell, the radioactive concentration and any abnormality are constantly monitored in the cell, isolation room and service area. In the course of decontamination, the exposure rate and surface contamination density are measured in the cell at suitable stages of operation, and the decontamination and radioactivity measurement are

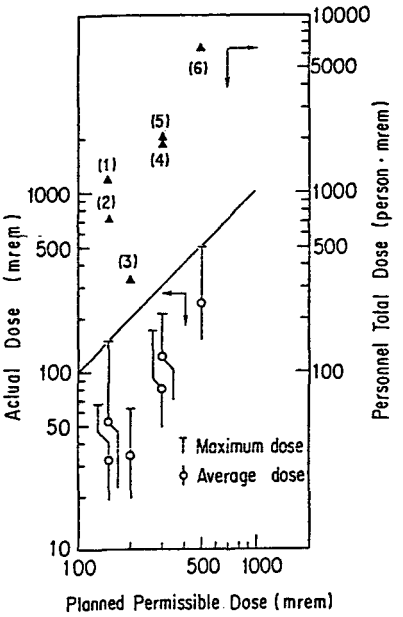

Fig.1 Individual and total dose of personnel vs. planned permissible dose in cell decontamination
Decontam. operation: (1) 24 persons x12 days, (2) 23x12, (3) 14x20, (4) 24x18, (5) 17x30, (6) 26x80.

repeated until the planned reduction levels are attained.

Personnel wear personal dosimeters, and their exposure doses are controlled by limiting the entry time not to exceed the planned permissible dose. Figure 1 shows individual and total dose of personnel vs. the planned permissible dose in recent decontamination works. As seen in the figure, nobody received the doses more than the planned permissible leves, and the ratio of the actual dose to the planned permissible level was between 1 and 1/2. The ratio of the maximum to the average dose is nearly between 2 and 3.

The protection for inhalation was made according to the guide in Table 1. Internal contamination was not detected in the decontamination personnel by the survey of the noses or the whole body counting after work. This means that the protection method was suitable.

3. Discussion of Radiation Monitoring Data

Decontamination Factor

When irradiated fuel or material was handled in the cell, the exposure rate was very high from hundreds R/h to tens of thousands R/h. After removal from the cell, including the cutting etc. and then the suction of cuttings with a vacuum cleaner or wiping them off by manipulator with rags, the exposure rate and surface contamination decreased to about 500 mR/h and $10^{-2}\mu Ci/cm^2$, respectively. Subsequently, the decontamination was made by hands with mops until the planned reduction levels were attained. As shown in Table 2, the exposure rate decreased only by one order at most but the surface contamination decreased by 2 ~ 3 orders. In cell decontamination with personnel entry, the decontamination factor, F, for exposure rate and surface

Table 1 Guide for respiratory protective devices to be used in JAERI

air concentration	surface contamination	respiratory protective device
$< (MPC)_a^{40}$	α: $< 10^{-4}$ μCi/cm^2 β: $< 10^{-3}$	half mask (if required)
$1\ (MPC)_a^{40} \sim 10(MPC)_a^{40}$	α: $10^{-4} \sim 10^{-3}$ β: $10^{-3} \sim 10^{-2}$	half mask or full-face mask
$10(MPC)_a^{40} \sim 100(MPC)_a^{40}$	α: $10^{-3} \sim 10^{-2}$ β: $10^{-2} \sim 10^{-1}$	full-face mask, SCBA*, SAM* or PAPR*
$100(MPC)_a^{40} <$	α: $10^{-2} <$ β: $10^{-1} <$	PAPR* or SAS*

* SCBA: self-contained breathing apparatus
PAPR: powered air-purifying respirator (suit)
SAM: supplied-air mask, SAS: supplied-air suit

Table 2 Decontamination factors for cell decontamination with personnel entry

item facility	Hot Lab.A	Hot Lab.B
dose rate		
pre-decontam., mR/h	70 ~ 1500	10 ~ 500
post-decontam., mR/h	10 ~ 120	1 ~ 50
surface contam. level		
pre-decontam., $\mu Ci/cm^2$	$2x10^{-3} \sim 2x10^{-2}$	$2x10^{-4} \sim 2x10^{-3}$
post-decontam., $\mu Ci/cm^2$	$2x10^{-6} \sim 1x10^{-5}$	$1x10^{-6} \sim 3x10^{-5}$
decontam. factor		
for dose rate	10 ~ 20	5 ~ 10
for surface contam.	2000 ~ 10000	100 ~ 200

contamination were 10 and 100~1000, respectively. The nuclides of contaminants were fission products of Cs-137, Sr-90, Y-90, Ru-106, Rh-106, Ce-144, Pr-144, Zr-95, Nb-95, Sb-125, etc., activated products of Mn-56, Co-60, etc. as β-emitters, and transuranic elements of Pu-238, Pu-239, Pu-240, Am-241, Cm-242, Cm-244 etc. as α-emitters. The nuclide Co-60 originated in cutting of the fuel coating materials. In the cell where the coating materials were cut, Co-60 accounted for about 90% of the total radioactivity.

Resuspension Factor

Due to decontamination operation and to the action of ventilation air turbulence, the contaminant particles deposited on the floor and machines surfaces are exfoliated and blown up into the air, leading to the air contamination. To know the extent of this dispersion is very useful for estimating preliminarily the extent of air contamination and for considering the personnel protection for inhalation.

Resuspension factors of particles as an indication of the air contamination, i.e. the ratio of the air concentration to the surface contamination density, are shown in Fig. 2 in frequency distribution, where the values were calculated for surface contamination densities over $10^{-4}\mu Ci/cm^2$ inclusive. As seen in the figure, the resuspension factors in decontamination works ranged between $10^{-8}cm^{-1}$ and $10^{-4}cm^{-1}$ the geometric mean was $K_g = 1x10^{-6}\ cm^{-1}$ and the geometric standard deviation was $\sigma_g = 8$.

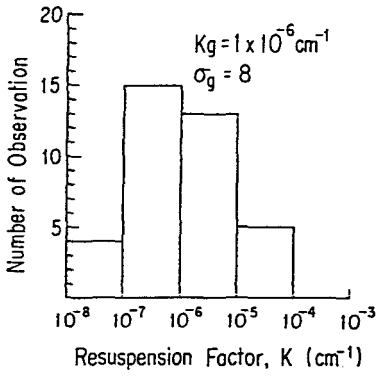

Fig.2 Frequency distribution of resuspension factors obtained in decontamination work

Particles deposited in the cell were produced in cutting or polishing the irradiated fuel or material. The particle size distributions measured by a cascade impactor are shown in Table 3. The activity median aerodynamic diameters were AMAD = 6 μm ~ 17 μm and the standard deviations were $\sigma_g = 1.2 \sim 2.7$; these AMADs are considerably large in the respect of possibility of their inhalation. These values are nearly the same as those (AMAD = 2.5 μm ~ 11 μm, $\sigma_g = 1.7 \sim 3.2$) previously measured by Ikezawa et al. [1] in one of the hot laboratories.

The resuspension factors of deposited particles in cell decontamination are fairly large, by four orders or more, larger than the value

Table 3 Size distribution of radioactive particles blown up into air during cell decontamination measured by cascade impactor

cell No.		1	2	4	5	6
particle size distribution	AMAD(μm)	16	6	17	8	13
	σ_g	2.4	2.7	1.3	1.9	2.2

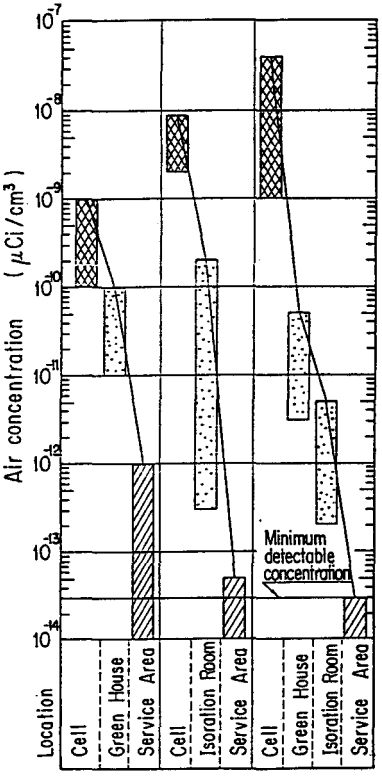

Fig.3 Radioactive air concentration in cell, green house, isolation room and service area during cell decontamination

of $K = 2 \times 10^{-8}$ cm^{-1}, which was obtained by Dunster [2] and which has been widely used as the base for the limits of surface contamination density.

Prevention of Contamination Spread with an Isolation Room

An isolation room is set up behind the cell to prevent the spread of contaminants during decontamination, to bring in equipment into the cell, and to take out the contaminated equipment from the cell or temporarily store it and also intermidiate-level solid wastes.

In Fig. 3 are shown radioactive air concentrations in the cell, the isolation room (or a green house temporarily built) and the service area, which were obtained by air monitoring during decontamination. As seen in the figure, the air concentrations in the isolation room or the green house decreased to 1/10 ～ 1/100 of those in the cell but not to nil. This is ascribed to the back door of cell being entirely open during decontamination. However, this room decreased extremely the leakage of contaminants to the service area. When a green house was set up within the isolation room, the leakage of contaminants to the service area was reduced to almost zero; and the air concentration there was undetectable.

References

(1) Y. Ikezawa, T. Okamoto, A. Yabe: Proceedings of the 5th Congress of the International Radiation Protection Society, p. 495-498, Jerusalem (1980)
(2) H.J. Dunster: Health Phys., 8, 353-356 (1962)

PRELIMINARY DECOMMISSIONING MEASUREMENTS FOR RESEARCH REACTOR RB - 1.

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To plan the decommissioning operations of the RB-1 reactor there have been made some activation measurements of the steel reactor container to determine the exposition of personal during the dismantling of the inner structure of the reactor, with Uranium already taken out, and to decide if the steel container must be sotred as a radioactive waste. There has been analyzed the radioactive contamination of the inner structures, with a special consideration to the graphite parts.

On the basis of such data and of activation and contamination measurements made on the reactor housing (for instance of concrete and other materials) it shall be possible to decide if the decommissioning operation shall be of SAFSTOR, ENCOMB or DECON type.

Occupational radiation exposure of the reprocessing plant and the plutonium fuel fablication facilities at PNC Tokai Works.

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I. Introduction

At the PNC Tokai Works, the fuel reprocessing plant has been operating since 1977, and 172 tons of spent fuels from PWR and BWR plant have been processed. While the plutonium fuel fabrication facilities have been operating since 1967, Pu and U mixed oxied fuels for FBR (Joyo) and ATR* (Fugen) have been manufactured continuously.

This paper describes the occupational exposure data on annual collective doses, collective doses per unit energy generated and individual dose distributions of the workers.

We have been making intense effort to reduce the exposure in these facilities based on the principle of ALARA and dose limits recommended by ICRP. The dose limitation system at PNC Tokai Works is given in Table 1. The doses to workers are evaluated quarterly or monthly with TLD badge and TLD finger-ring dosimeter for the measurement of gamma, beta and neutron exposure.

Table 1. Dose limitation system at Tokai Works

Category	investigation level	action level	dose limit
A	300mrem/3months	1.3rem/3months	3rem/3months
B	100mrem/month	400mrem/3months	1.5rem/year

A. workers who recieve occupational dose continualy
B. workers who enter the controll area occasionaly

II. Fuel reprocessing plant

1. Occupational doses

The occupational doses to fuel reprocessing workers for the period 1977 - 1982 are summarized in Table 2.

It was the pre-operational reprocessing period during 1977 - 1979. The regular operations were carried out in 1980 - 1982, and nine reprocessing runs using fuels from seven different reactors have been performed. In this 3 years, annual collective doses were 60 - 70 man·rem and annual average doses to workers with measurable doses were about 100 mrem. This could be attained with the efforts to reduce both collective dose and individual doses associated with the operational works and with the special radiation works such as repairs and inspections.

The operational radiation control programs have been performed on the special radiation works, including the reasonably permitted

* Advanced Thermal Reactor of HWR type

Table 2. Doses at the PNC Fuel Reprocessing Plant

Fiscal year	Number of workers monitored	Number of ** workers with measurable doses	Annual *** collective dose (man rem)	Average measurable dose per workers (mrem)
1977	768	447	1.9	4
1978*	968	435	27.6	63
1979*	1489	709	88.1	124
1980	1217	597	62.9	106
1981	1570	593	64.0	108
1982	1751	722	71.1	98

* repair of the acid recovery evaporator ** minimum measurable dose is 10 mrem
*** summation of measurable doses

dose limit authorized for each worker.

There were repair works of the acid recovery evaporator from 1978 - 1979, and this source of exposure appears to be dominating the doses. The total integrated working time assoicated with this repair was 7452 man·day and the collective dose was 63 man rem.

2. Relation of collective dose to reprocessing load

The collective dose per unit energy generated is given in Table 3. The average value over the 3-year period of 1980 - 1982, the regular operations were carried out, was 104 man rem[GW(e)a]$^{-1}$. This values is overestimates for reprocessing alone because of the other maintenance works during the intervention periods. The avereage value of the annual collective dose per unit energy generated associated with the reprocessing runs alone was 54 man·rem[GW(e)a]$^{-1}$.

Table 3. Annual collective dose per unit energy generated at the PNC Fuel Reprocessing Plant

Fiscal year	Annual collective dose (man rem)			Energy * generated (GW(e) a)	Annual collective dose per unit energy generated (man rem (GW(e) a)$^{-1}$)		
	Total	during reprocessing runs	during intervention periods		Total	during reprocessing runs	during intervention periods
1977	1.90	1.90	—	0.047	41	41	—
1978	27.6	13.4	14.2	0.120	230	112	118
1979	88.1	9.8	78.3	0.190	463	52	411
1980	62.9	38.6	24.3	0.662	95	59	36
1981	64.0	39.0	25.0	0.642	100	61	39
1982	71.1	24.7	46.4	0.603	118	41	77

* derived from following ; (Burn up (MWD/Ton·U) × Reprocessing (Ton·U))
+ 365 (MWD/MWY) × 0.3 (GW(e)/GW(t)) × 10^{-3} (GW/MW)

3. Doses received by various groups of workers

There are six groups of workers at the reprocessing plant, and the exposure data for each groups are given in Table 4. The annual average doses for chemical analysis group were 120-144 mrem during the regular operation period 1980-1982. These were about 2 -4 times that estimated for other groups. This was due to increasing requirements for analytical services during reprocessing runs. Repair works of the acid recovery evaporator resulted in dominant doses to maintenance personnels in 1979.

Table 4. Doses received by various groups of workers at the PNC Fuel Reprocessing Plant

Group	1979		1980		1981		1982	
	Number of workers	Annual average dose (mrem)	Number of workers	Annual average dose (mrem)	Number of workers	Annual average dose (mrem)	Number of workers	Annual average dose (mrem)
Operation	328	40	353	74	430	37	451	55
Waste treatment	296	73	110	39	122	29	313	21
Chemical analysis	125	59	146	138	193	120	206	144
Maintenance	465	93	305	29	484	40	486	15
Health physics	58	34	66	47	59	36	63	43
Administration & Others	217	4	237	3	282	<10	232	<10

III. Plutonium fuel fabrication facilities

The occupational doses to Pu fuel fabrication workers for the period 1977 - 1982 are summarized in Table 3.

Table 5. Doses at the PNC Pu Fuel Fublication Facilities

Fiscal year	Number of workers monitored	Number of workers with measurable doses	Annual collective dose (man rem)	Average measurable dose per workers (mrem)
1977	311	114	12.9	90
1978	606	134	10.2	76
1979	555	225	23.1	103
1980	561	198	33.3	168
1981	909	200	50.4	252
1982	584	266	57.5	216

The cumulative amount of fuels fablicated for ATR and FBR was 37.6 tons and 1.2 tons respectively, and cumulative collective dose was 188 man rem in this six years.

In the period of 1980 - 1982 the reactor grade plutonium recoveried from highly burn-up spent fuel has become to be treated in place of the standard grade plutonium, and especially the amount of fuels fablicated for FBR with high plutonium content has been increased. According to these situation both annual collective dose and annual avarage doses have increased since 1980. The annual collective doses were 33 - 58 man rem, and annual average doses were 168 - 252 mrem in this 3 years.

IV. Dose distribution

Figure 1 shows the probability vs dose of the individual annual doses at PNC Tokai Works for fiscal year 1982 in log-normal distribution. About 60 % of the individuals over the minimum measurable dose were below 100 mrem and 95% of them were below 500 mrem. It was generally founded individual doses have been indicated a log-normality. [Ref. 1] But in the case of Tokai Works the dose distribution depart from the straight line. At the plutonium fuel fabrication facilities Pu fuels were handled routinaly in glove

boxes and workers were normally exposed in the relatively constant radiation field. As individual doses were controlled to be kept below the investigation level (300 mrem per 3 months) by the radiation control programme depending on the dose limitation system, the individual highest annual doses have been get down to in a dose range between 0.5 and 1.0 rem. At the fuel reprocessing plant, same control programme was applied in the routine radiation works, while in the case of the special radiation works the investigation level were occasionally varied up to the reasonably permitted dose which did not exceed the action level in Table 1. The difference of the dose distribution between these two facilities was reasonable according to these situation.

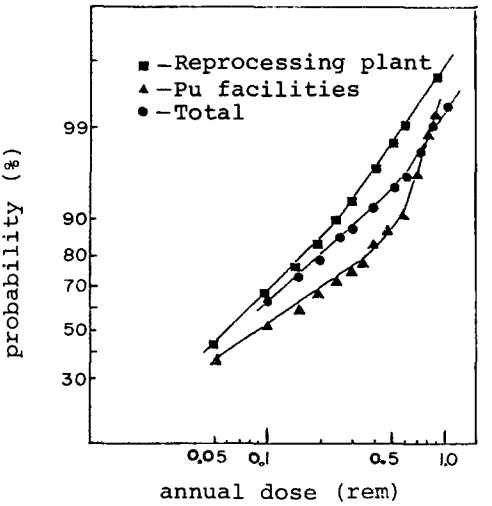

Fig. 1 The log-normal distribution of annual individual doses

V. Conclusions

The following results have been obtained by the operational experience of fuels facilities at the PNC Tokai Works.

1. The collective dose per unit energy generated at the fuel reprocessing plant was 104 man rem $[GW(e)a]^{-1}$.
2. Annual average doses with measurable dose to fuel reprocessing workers were about 100 mrem for the regular operation. Annual average doses to Pu fuel fabrication workers were about 200 mrem for the routinaly fuel fablication in the last three years.
3. There were two major sources of collective dose at fuel reprocessing plant and the plutonium fuel fabrication facilities; they were the routine operations and special maintenance. The former was dominant at the plutonium fuel fabrication facilities and the latter was dominant at the fuel reprocessing plant.
4. The investigation level of 300 mrem per 3 months for routine radiation workers, which aims to inspect unforeseen exposure sources, acted on reducing the individual annual dose below 1.0 rem.

References

1. UNSCEAR 82 Occupational exposures, Annuex H, 1982 (New York; United Nations)

DEVELOPMENT OF AIRBORNE RADIOIODINE MONITORING TECHNIQUE IN JAERI

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Radioiodine is usually an important object of airborne monitoring in nuclear facilities. It was found that radioiodine in the exhaust air of hot laboratory, radioisotope production plant, fuel reprocessing test facility and nuclear reactors in JAERI consisted of particulate iodine, inorganic iodine and organic iodine in different proportions from facility to facility [1,2]. Studies have been made to establish techniques required for monitoring airborne radioiodine rapidly and precisely. Described in the present paper are (1) the monitoring techniques for airborne radioiodine currently used in JAERI, (2) characterization of radioiodine in the exhaust air of the facilities, and (3) selective collection of radioiodine from the high concentration of radioactive noble gas-containing atmosphere in nuclear reactor accidents.

1. Current Monitoring Technique for Airborne Radioiodine in JAERI

Based on the monitoring experiencies of inorganic and organic iodine species in the exhaust air from the facilities, a charcoal loaded filter paper (CP-20) and a charcoal cartridge (CHC-50) have been used in JAERI [3,4,5]. Collection performances of CP-20 and CHC-50 for airborne radioiodine were studied by laboratory experiments and in situ experiments at the facilities. From experiments on the collection of organic iodides in moist air, impregnation of chemicals such as TEDA on charcoal and heating sampling air were found to be very effective for improving collection efficiency [3]. In Table 1 are shown the collection efficiencies of CP-20 and CHC-50 in relative humidity range of 50% to 90% [4]. The air heating technique is useful at relative humidities above 90% and for long-time sampling. No dependency of the collection efficiency on the methyl iodide concentration was found in the TEDA impregnated charcoal cartridge (1cm thickness) in the concentration range of 10^{-19} to 10^{-8} g/cm^3 [5].

A sampling head of the monitor is shown in Fig.1 which can collect radioactive particles on a fibrous filter and radioiodine

Table 1 Collection efficiencies of adsorbents for airborne iodine*1

Adsorbents	Chemical form of iodine	Range of efficiencies % (representative value)		Remarks
		CP-20	CHC-50	
Unimpregnated charcoal	Inorganic Organic	40-90(50) 10-40(20)	90-100(95) 60-95 (80)	*1 Relative humidity: 50%-90%, Face velocity:20cm/sec (Sampling period 8hr) *2 10wt%TEDA impregnated charcoal *3 Heated to 70°C-90°C
Impregnated charcoal*2	Inorganic Organic	70-98(80) 40-95(70)	~100(100) 90-99 (95)	
Air heated*3	Inorganic Organic	~ 90(90) 70-90(80)	~100(100) 90-99 (95)	

gas on CP-20 and CHC-50, and measure their activities with radiation probes [4]. Though CP-20 of thickness 2mm has rather low collection efficiency, it can be used for continuous monitoring with a GM counter. CP-20 and CHC-50 are always set in series in the sampling head as shown in the figure. When the activity on CP-20 exceeds a given level, CHC-50 is demounted and its activity is measured precisely with gamma-ray spectroscopy. The minimum detectable concentration of the iodine monitor is $10^{-10}\mu Ci/cm^3$ for one hour sampling at a flow rate of 50 l/min.

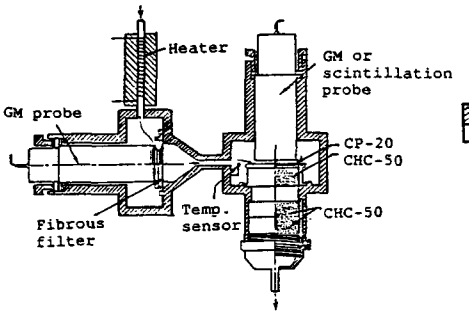

Fig.1 Sampling head of iodine monitor

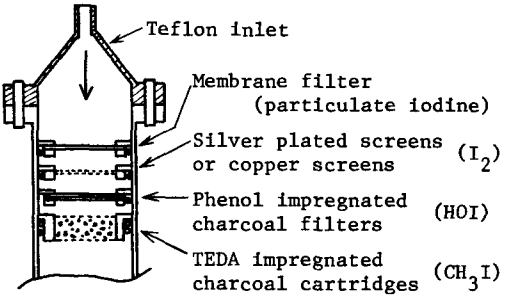

Fig.2 Diagram of modified May pack sampler.

2. Characterization of Airborne Radioiodine

Airborne radioiodine species in the exhaust air at the ^{131}I production plant and the fuel reprocessing test facility were analyzed by a radio-gaschromatograph [1] and a radioiodine species sampler [2] which consisted of a particulate filter, caustic scrubbers and activated charcoal beds. In both the plant and facility the fraction of inorganic and organic forms were high and the fraction of particulate form was low. In both cases, radioiodine was mostly in organic form downstream of the charcoal filters. These results were in good agreement with those from the routine radioiodine monitoring using CP-20 and CHC-50. The results also provided basic data for evaluation of the iodine collecting performances of CP-20 and CHC-50.

In order to analyze the forms of airborne radioiodine species more easily and precisely, May pack sampler was modified and examined [6]. A major point of modification is that phenol impregnated charcoal filter papers are used to collect hypoiodous acid (HOI) [7]. This HOI collection method is based on a principle similar to that of the radioiodine species sampler developed by Keller et al. [8]. A diagram of the modified May pack sampler is shown in Fig.2. In the performance test, air containing aerosol, $^{131}I_2$, $HO^{131}I$ or $CH_3{}^{131}I$ was passed through the sampler at a face velocity of 10cm/sec. The membrane filters collected aerosol completely. Fractions of I_2 and HOI were adsorbed on the filters but most were desorbed by flowing clean air through the filters after the adsorption. Only about 1% per membrane filter remained, respectively. I_2 collection efficiencies of silver plated screens and copper screens were more than 70% per screen and about 100% per 10 screens, respectively. These screens also adsorbed HOI. HOI collection efficiencies of

silver and copper screens were about 20% and 3% per 5 screens, respectively. HOI collection efficiency of the phenol impregnated charcoal filter paper was about 90% per sheet. Though this filter paper adsorbed a fraction of CH_3I, most was desorbed by flowing clean air. The remaining CH_3I fraction was less than 1% per sheet. CH_3I collection efficiency of the TEDA impregnated charcoal cartridge (bed depth 2cm) was about 100%. The formulas for determining the radioiodine species distribution from the radioactivities on respective adsorbents were obtained on the basis of the above performance of the modified May pack sampler. These formulas were applied to easy and precise estimation of change of chemical form of CH_3I due to light exposure. For example, the fractions of radioiodine were 0.3% particulate, 90% I_2, 2% HOI and 7% CH_3I after 40 J/cm^2 light exposure [9].

3. Radioiodine Monitoring Method in Nuclear Reactor Accidents

Radioiodine is released together with large amounts of radioactive noble gases in nuclear reactor accidents. The radioiodine monitoring using a charcoal filter as an adsorbent leads to overestimation of the radioiodine concentration in air due to the adsorption of radioactive noble gases. Therefore, the adsorbents for radioiodine monitoring in nuclear reactor accidents must have high retention for radioiodine and low retentions for radioactive noble gases. Several adsorbents having such properties are reported [10,11].

In JAERI, to establish the radioiodine monitoring method in nuclear reactor accidents the retention efficiencies for methyl iodide and xenon were investigated for several adsorbents shown in Table 2 by flowing the air containing $CH_3{}^{131}I$ or ^{133}Xe through the adsorbent at constant relative humidity and flow rate [12].

Fig.3 shows the relation between retention efficiency of adsorbents for ^{133}Xe and loading time. In all the adsorbents, efficiency tends to decrease with increase of the loading time. It is seen that ^{133}Xe is adsorbed by TEDA impregnated charcoal and silver loaded zeolite A, but little by silver loaded silica gel A and silver loaded alumina. Table 2 shows retention efficiencies for methyl iodide and retention ratios defined as the ratio of retention efficiency for methyl iodide to that for xenon. The retention efficiency for methyl iodide is in the order of TEDA charcoal, silver

Table 2 Retention efficiency for methyl iodide and retention ratio under the conditions: face velocity 19 cm/sec, loading time 5 to 60 min and bed depth 2 cm.

Adsorbent	Impregnant Chemicals	Impregnant Content (%)	Shape	Retention efficiency for methyl iodide(%) Relative humidity(%) 20	Retention efficiency for methyl iodide(%) Relative humidity(%) 90	Retention ratio * (%) Relative humidity(%) 20	Retention ratio * (%) Relative humidity(%) 90
TEDA Charcoal	TEDA	5	granule	99+	99+	$1x10^1$-$2x10^2$	$5x10^1$-$8x10^3$
Ag silica gel A	$AgNO_3$	12	bead	98-99+	99	$3x10^5$-$1x10^6$	$6x10^5$-$1x10^7$
Ag silica gel D	$AgNO_3$	12	granule	99+	92-92+	$7x10^3$-$2x10^4$	$9x10^3$-$8x10^4$
Ag zeolite A	Ag	35	bead	99+	89-99	$9x10^0$-$1x10^2$	$8x10^1$-$7x10^6$
Ag zeolite B	Ag	38	bead	99+	82-94	$2x10^4$-$3x10^5$	$6x10^4$-$3x10^5$
Ag alumina	$AgNO_3$	9	bead	99+	97-98	$4x10^5$-$7x10^6$	$4x10^6$-$3x10^7$

* Retention ratio is defined as the ratio of retention efficiency for CH_3I to that for Xe.

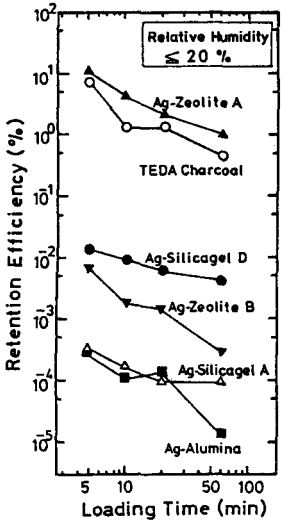

Fig.3
Retention efficiency for xenon as a function of loading time under the conditions: face velocity 19 cm/sec and bed depth 2 cm.

loaded silica gel A and silver loaded alumina. The retention ratio is in the order of silver loaded alumina, silver loaded silica gel A and silver loaded zeolite B.

From these results, silver loaded alumina and silver loaded silica gel A, which have high retention efficiencies more than 97% for methyl iodide and high retention ratios more than 10^5, are suitable for as the adsorbents for emergency radioiodine monitoring.

A silver loaded alumina cartridge for iodine monitoring was made in trial. After collecting methyl iodide from the air containing 10^{-4} μCi/cm^3 ^{133}Xe and 10^{-10} μCi/cm^3 $CH_3{}^{131}I$ by the cartridge, the spectrum was measured with a Ge(Li) detector. The results showed that the silver loaded alumina cartridge could selectively collect $CH_3{}^{131}I$ from the air containing ^{133}Xe.

REFERENCES

[1] E. Tachikawa and M. Naritomi : J. Nucl. Sci. Technol., 8, 371 (1971).
[2] M. Naritomi and S. Fukuda : J. At. Energy Soc. Japan, 14, 531 (1972).
[3] M. Naritomi, Y. Yoshida and S. Fukuda : J. Nucl. Sci. Technol., 10, 292 (1973).
[4] Y. Yoshida : Radioisotopes, 26, 508 (1977).
[5] T. Ohata, H. Matsui, M. Naritomi and Y. Yoshida : JAERI - M-8158 (1978).
[6] H. Noguchi, M. Murata, Y. Tsuchioka, H. Matsui and M. Kokubu : JAERI-M-9408 (1981).
[7] M. Naritomi, J. Takada and S. Kitani : Preprint 1977 Fall Meeting AESJ, p.169 (1977).
[8] J. H. Keller, F. A. Duce and W. J. Maeck : Proc. 11th AEC Air Cleaning Conf., CONF-700816, p.621 (1970).
[9] H. Noguchi, H. Matsui and Y. Yoshida : J. At. Energy Soc. Japan, 24, 381 (1982).
[10] C. Distenfeld and J. Klemish : NUREG/CR-0314 (1978).
[11] J. E. Cline : Health Phys., 40, 71 (1981).
[12] S. Kato, H. Noguchi, M. Murata, H. Imai, H. Matsui and M. Kokubu : Hoken Butsuri, 17, 427 (1982).

Auswirkung differenzierter Oberflächenkontaminationsrichtwerte auf die Einsatzmöglichkeiten verschiedener Kontaminationsmessgeräte

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EINLEITUNG

In vielen Ländern werden heute bei den Richtwerten für Oberflächenkontaminationen nur α-Strahler und die restlichen Nuklide, sog. β/γ-Strahler, unterschieden. Die Richtwerte für beide Gruppen wurden je für den kritischen Pfad des kritischen Nuklids berechnet. Für viele Nuklide mit geringer Radiotoxizität, die z.B. in der Medizin oder in der Forschung angewendet werden, sind diese Grenzwerte sehr restriktiv. Die meisten dieser Nuklide sind auch nicht leicht messbar, so dass oft ein grosser Aufwand zur Messung eines ungerechtfertigt tief liegenden Richtwertes erforderlich ist. Die Einführung risikogerechter Oberflächenkontaminationsrichtwerte wurde verschiedentlich vorgeschlagen und im Bericht NRPB-DL-2 (1) wurden Beispiele für neue Richtwerte publiziert. Entscheidend dür die praktische Bedeutung dieser neuen Werte sind ihre Auswirkungen auf die erforderlichen Messverfahren bei der Kontaminationskontrolle. Vereinzelt wurde die Befürchtung geäussert, die Einführung nuklidspezifischer Richtwerte führe zu einer Verkomplizierung der Ueberwachung, insbesondere in Labors, in denen mit vielen verschiedenen Nukliden gearbeitet werde. Die tatsächliche Auswirkung der Vorschläge in NRPB-DL-2 auf die Praxis wurde für 58 Nuklide und drei Detektortypen bestimmt.

BESTIMMUNG DER GERAETEEMPFINDLICHKEITEN

Für die Beurteilung von Oberflächenkontaminationsmessungen ist die Impulsrate pro Richtwert eine geeignete Grösse. Dieser Wert wurde mit Flächenquellen von 10 cm x 10 cm der Nuklide C-14, Fe-55, Co-60, Sr/Y-90, I-125, Cs-137, Tl-204 für je ein Gerät mit Endfensterzählrohr, Xe-Proportionalzähler und Butan-Durchflussproportionalzähler bestimmt. Der Abstand zwischen Quelle und Detektor- resp. Sondenunterkante betrug jeweils 0,5 cm. Für die weiteren aufgeführten Nuklide wurde die Empfindlichkeit berechnet.

DISKUSSION DER ERGEBNISSE

Durch die Einführung der neuen Richtwerte würde die Kontaminationskontrolle für einige wichtige Nuklide erleichtert. Die Zahl der Nuklide, für die mit einfacher Messung ein Richtwert nicht mehr nachgewiesen werden könnte, wäre kleiner. Wie aus den Tabellen ersichtlich, ist es ein Trugschluss anzunehmen, ein einheitlicher Richtwert ergebe auch einigermassen einheitliche Richtwertimpulsraten. Es ist im Gegenteil feststellbar, dass die Richtwertimpulsraten bei Anwendung der neuen Richtwerte mit wenig Ausnahmen sogar geringere Nuklidabhängigkeit zeigen.

EMPFINDLICHKEIT ips/Richtwert	HEUTIGER RICHTWERT (10^{-4} µCi/cm²)	RICHTWERT NACH VORSCHLAG IN NRPB-DL-2 (10^{-2},10^{-3} resp. 10^{-4} µCi/cm²)
< 5	Cr-51,Mn-54,Fe-55,Zn-65,Nb-93m,Pb,210	Pb-210
5 - 15	C-14,S-35,Ga-67,Sr-85,Tc-99m,I-125, Cs-131	Nb-93m
15 - 50	Ca-45,Co-57,Co-58,Co-60,Se-75,Sr-87m, Cd-109,Ag-110m,In-111,In-113m,I-123, Sb-125,Ba-133,Eu-152,Gd-153,Er-169, Yb-169,Hg-197,Tl-201,Hg-203,Bi-207	Ca-45,Mn-54,Co-58,Co-60,Zn-65,Sr-87m, In-111,In-113m,Sb-125,Eu-152,Gd-153, Er-169,Yb-169
50 - 150	Na-22,Na-24,P-32,Cl-38,K-42,Sc-46, Ca/Sc-47,Mn-56,Fe-59,Br-82,Rb-86,Sr-89, Sr/Y-90,Cd-115,Cd-115m,I-131,Cs-137, Ce-141,Re-189,Ir-192,Au-198,Tl-204	C-14,Na-22,Na-24,P-32,S-35,Cl-38,K-42 Sc-46,Ca/Sc-47,Mn-56,Fe-59,Ga-67,Br-82, Sr-85,Rb-86,Sr-89,Sr/Y-90,Tc-99m, Cd-115,Cd-115m,I-125,I-131,Cs-137, Ce-141,Re-189,Ir-192,Au-198,Tl-204
150 - 500		Cr-51,Fe-55,Co-57,Se-75,Cd-109,Ag-110m I-123,Cs-131,Ba-133,Hg-197,Tl-201, Tl-203,Bi-207

TABELLE 1: Richtwertimpulsraten für die Geräte Contamat-X2 der Firma Frieseke & Hoepfner und LB 1210 B der Firma Berthold mit Xe-Proportionalzählern

	Contamat	LB 1210 B
Fläche des Detektors:	160 cm²	100 cm²
Flächengewicht des Fensters:	6,75 mg/cm² (Titan)	5,0 mg/cm² (Titan)
Nulleffekt:	15 ips	7 ips

EMPFINDLICHKEIT ipm/Richtwert	HEUTIGER RICHTWERT (10^{-4} µCi/cm²)	RICHTWERT NACH VORSCHLAG IN NRPB-DL-2 (10^{-2},10^{-3} resp. 10^{-4} µCi/cm²)
< 300	Cr-51,Mn-54,Fe-55,Co-57,Zn-65,Se-75, Sr-85,Nb-93m,I-125	Nb-93m,I-125
300-900	Ni-63,Tc-99m,In-111,I-123,Bi-207, Pb-210	Mn-54,Zn-65,In-111,Pb-210
900-3000	Co-58,Ga-67,Sr-87m,Ag-110m,In-113m, Ba-133,Gd-153,Tl-201	Co-57,Co-58,Se-75,Sr-85,Sr-87m,In-113m, Gd-153
3000-9000 (50-150 ips)	C-14,Na-22,Na-24,P-32,S-35,Cl-38, K-42, Ca-45,Sc-46,Ca/Sc-47,Mn-56, Fe-59,Co-60,Br-82,Rb-86,Sr-89, Sr/Y-90,Cd-109,Cd-115,Cd-115m, Sb-125,I-131,Cs-137,Ce-141,Eu-152 Er-169,Yb-169,Re-189,Ir-192,Hg-197, Au-198,Hg-203,Tl-204	Na-22,Na-24,P-32,Cl-38,K-42,Ca-45,Sc-46, Ca/Sc-47,Cr-51,Fe-55,Mn-56,Fe-59,Co-60, Br-82,Rb-86,Sr-89,Sr/Y-90,Tc-99m,Cd-115, Cd-115m,I-123,Sb-125,I-131,Cs-137,Ce-141, Eu-152,Er-169,Yb-169,Re-189,Ir-192,Au-198, Tl-204,Bi-207
9000-30000		C-14,S-35,Ni-63,Ga-67,Cd-109,Ag-110m, Ba-133,Hg-197,Tl-201,Hg-203

TABELLE 2: Richtwertimpulsraten für das Gerät TMB 2.1 mit Durchflussproportionalzählrohr DDB 2002 der Firma Münchner Apparatebau

Fläche des Detektors: 175 cm²
Flächengewicht des Fensters: 0,9 mg/cm²
Nulleffekt: 500 ipm
Spülgas: Butan (bei Verwendung einer Mischung von 90 % Argon und 10 % Methan ergeben sich höhere Empfindlichkeiten für Nuklide mit weichen X-Strahlen).

EMPFINDLICHKEIT ips/Richtwert	HEUTIGER RICHTWERT (10^{-4} µCi/cm²)	RICHTWERT NACH VORSCHLAG IN NRPB-DL-2 (10^{-2},10^{-3} resp. 10^{-4} µCi/cm²)
< 0,3	Cr-51,Mn-54,Fe-55,Co-57,Zn-65,Se-75, Sr-85,Nb-93m,I-125,Cs-131	Mn-54,Zn-65,Nb-93m
0,3 - 1	Co-58,Ga-67,Tc-99m,In-111,I-123, Tl-201,Bi-207,Pb-210	Co-57,Co-58,Se-75,Sr-85,In-111,I-125, Pb-210
1 - 3	C-14,S-35,Ca-45,Sc-46,Sr-87m,Cd-109, Ag-110m,In-113m,Sb-125,Ba-133, Eu-152,Gd-153,Er-169,Yb-169,Hg-197	Ca-45,Sc-46,Cr-51,Sr-87m,In-113m, Sb-125,Cs-131,Eu-152,Gd-153,Er-169, Yb-169
3 - 10	Na-22,Na-24,P-32,Cl-38,K-42,Ca/Sc-47, Mn-56,Fe-59,Co-60,Br-82,Rb-86,Sr-89, Sr/Y-90,Cd-115,Cd-115m,I-131,Cs-137, Ce-141,Re-189,Ir-192,Au-198,Hg-203, Tl-204	Na-22,Na-24,P-32,Cl-38,K-42,Ca/Sc-47, Mn-56,Fe-59,Co-60,Ga-67,Br-82,Rb-86, Sr-89,Sr/Y-90,Y-90,Tc-99m,Cd-115, Cd-115m,I-123,I-131,Cs-137,Ce-141, Re-189,Ir-192,Au-198,Tl-201,Tl-204, Bi-207
10 - 30		C-14,S-35,Fe-55,Cd-109,Ag-110m,Ba-133, Hg-197,Hg-203

TABELLE 3: Richtwertimpulsraten für das Gerät X5W mit GM-Endfensterzählrohr der Firma Graetz

Fläche des Detektors: 6,4 cm²
Flächengewicht des Fensters: 1,5 - 2 mg/cm² (Glimmer)
Nulleffekt: 0,4 ips

LITERATUR

(1) Derived Limits for Surface Contamination, A. Wrixon, G. Linsley, K. Binus and D. White, 1979, NRPB-DL-2 und NRPB-DL-2 Supplement, 1982

EVALUATION OF THE RADIOACTIVE AEROSOL CONCENTRATION IN AREAS OF FILTRATION, DRYING AND PACKAGING OF YELLOW CAKE IN A BRAZILIAN URANIUM MILL

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INTRODUCTION

The radiological hazards associated with uranium mining and milling have long been recognized. The Brazilian Nuclear Energy Comission (CNEN) understands the need to control those hazards and has been working with the uranium industry to have this task accomplished.

The CNEN has monitoring and auditing programs and in this way it can help the industry to overcome any lack of sufficient number of equipment necessary to have an adequate radiological control. Having the programs also gives the CNEN the assurance that the health physics procedures it recommends are being followed.

One of the areas of most concern is the yellowcake filtration , drying and packaging sections of the mill. This is because during the start of commercial operations begining the first semester of 1982, the air sampling results showed high levels of airborne uranium. The objective of this paper is to show those results and compare them with data acquired after the yellow cake area was closed down for improvements.

MATERIALS AND METHODS

The Complexo Minero-Industrial do Planalto de Poços de Caldas (CIPC) installed by NUCLEBRAS (Empresas Nucleares Brasileiras)consists of the mine mill and auxiliary installations, located at Caldas, Minas Gerais.

The mine is of the open-pit type and one can find uranium starting from 40 m to 300 m down, The ore grade being mined is 847 ppm but there are parts where the average grade reaches 1500 ppm. The area is about 500000m^2 (NUCLEBRAS 82). The mill has nominal capacity of 500 metric ton/yr of U_3O_8, and the uranium concentrate produced comes as ammonium diuranate, ^{3}ADU.A flow diagram of the area is shown in Figure 1.

The CNEN occupational monitoring program (Mo83) at the CIPC consists of air, external radiation surface contamination and drinking water monitoring at the various facilities. A routine bioassay program is also performed (Li83). Due to the surface contamination and inhalation hazards involved, the results presented in this paper focus

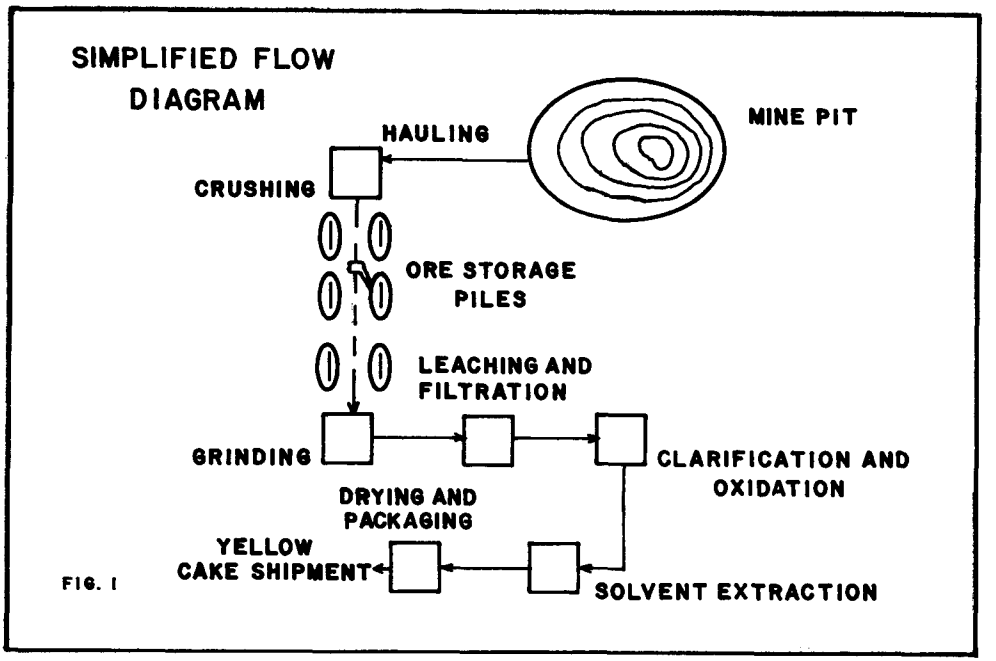
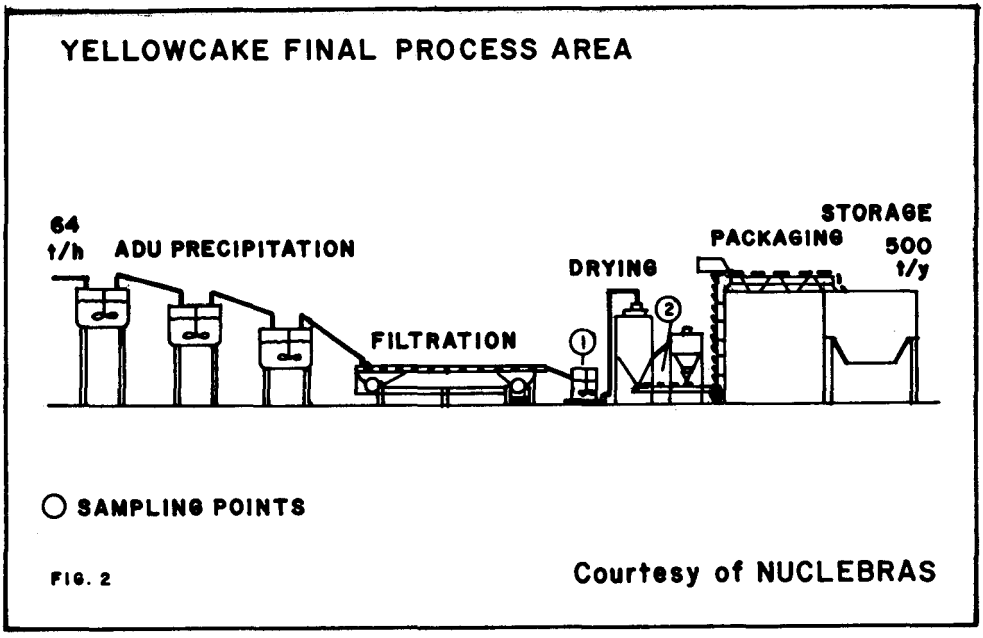


FIG. 1

only on the air monitoring of the yellow cake area.

The yellow cake area has sections where the uranium is preciptated with ammonia, filtered to separate the concentrate from any residual liquids,dried, atomized and packed for shipment. The sampling locations are shown in Figure 2. For each section the sampling location represents the point where the worker is more likely to stay and not necessarily the points where the uranium levels in the air could be the highest. Samplers wereplaced at a height representative of the breathing zone of the section worker.

The monthly air sampling at the yellow cake building was performed using a Eberline RAP-1 continuous air sampler during a 8 hr-day shift. Only routine operation was considered,that is,the two filtration lines and packaging sections were operational and no repair or maintenance was being performed during sampling. Glass fiber filters were used to collect the airborne yellow cake and counted for gross alpha activity one week after the end of sampling with a Eberline scintillation alpha counter. Two filters per location per month were used from march to december 1982. During July all the facilities were shut down and no sampling was performed. Sampling was not performed at the packaging section, due to engineering problems in the line which takes the dried yellow cake to the hopper before it is poured into the drums, Packaging was being done in the dryer section, straight into the drums.



RESULTS AND DISCUSSION

The results of the filtration area were very high from March to June, the average uranium concentration and standard deviation being $2.1 \pm 0.6 Bqm^{-3}$. During July the area was shut down for modifications in order to solve engineering related problems and try to improve some health physics aspects in the area. Modifications included changes of pipes which were not appropriate because of yellow cake leaking; floor finishing which was made smooth and easier to clean; and the color of some yellow painted surfaces which were changed to make the identification of spills possible by visual inspection. The results after restart (August to December) showed an average concentration of 0.78 ± 0.37 Bqm^{-3}. The analysis of variance showed a significant difference ($\alpha = 0.05$) between the first and second semester of 1982.

No statistical difference was found between the results before and after shut down for the packaging section (dryer location) the average for the year being 0.23 ± 0.13 Bqm^{-3}. This result is no longer representative of the actual working conditions since the correct line is now operating.

The air sampling results presented were utilized to estimate the effective dose equivalent based on ICRP publication 30 (ICRP79). An AMAD equal to 1 μm and solubility class W were used. The dose calculated for the filtration worker using the results of the first semester was 9.54×10^{-3} Sv. For the levels encountered on the second semester the dose was 3.48×10^{-3} Sv. For the packaging worker the dose calculated was 10.47 x 10-4 Sv.

CONCLUSIONS

The high levels of airborne uranium encountered at the filtration plataform, which operates with two open filters, has warranted the use of masks by the workers. According to the IAEA guidelines (IAEA83) masks should be worn while corrective measures are taken. The use of masks is also recommended for emergency situations, repair and maintenance and in special short-term ocurrences. Proper engineering control measurements were requested in order to improve the working conditions in the area.

The overall conclusions are that the inspection program helped to improve radiation safety aspects; train personnel; maintain a good relations hip between regulatory agency and the industry; get data for better analysis on the extent of the monitoring program; and gather working experience necessary for a good health physics control that will be needed more and more with other mine and mill installations that will be operating in the near future.

REFERENCES

IAEA83 International Atomic Energy Agency, "Radiation Protection of Workers in the Mining and Milling of Radioactive Ores", S.S. nº 26, IAEA, Vienna (1983).

ICRP79 International Commission on Radiological Protection, pub. 30, "Limits for Intakes of Radionuclides by Workers", Pergamon Press, Oxford (1979).

Li83 Lipsztein, J.L., Ramalho, A.T., Fonseca, A., Wrenn, M.E., and Cohen, N., "A Mathematical Model for Uranium Biokinetics and its application to the Internal Dosimetry Bioassay Program of the IRD", Meeting on Radiological Protection and Dosimetry, Centrecon, Itaipava, R.J., Brazil (March 1983).

Mo83 Mouço, C., Carvalho, S., Estrada, J. and Ney, C., 1983, "Occupational Radiation Protection Fiscalization Program at a Brazilian Uranium Mine and Mill", Health Phys. Society Annual Meeting, Baltimore, M.D., (June 1983).

NUCLEBRÁS82 Empresas Nucleares Brasileiras. Folheto Informativo do CIPC (May 1982).

ESTIMATING MAINTENANCE DOSE AT A MESON FACTORY

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ABSTRACT

By far the largest fraction of the dose to personnel at TRIUMF is due to work done in areas of high residual radiation fields. In order to be able to plan such work effectively in accordance with the ALARA principle it is necessary to be able to predict the induced fields for a variety of operating conditions followed by arbitrary cooling periods. We have measured the gamma-ray spectra at various locations in the cyclotron, the cyclotron vault, the beam lines and near associated meson production target stations using a small high purity germanium detector. Analysis of these spectra and a knowledge of the operating fields which induce this activity allows us to calculate the yields of the radioactive species induced by an arbitrary set of operating conditions. These yields in turn allow a good estimate of future induced fields to be made. By folding in data on man-hour requirements to meet maintenance schedules and planned future development work it is possible to project dose loads in light of higher beam production scenarios. We can thus identify areas or devices that need improvement in reliability or remote handling capability. Several such problem areas have been identified and corrective action taken to lower the man-dose during repair or maintenance of critical components. The estimates of future dose-loads are reviewed periodically as better measurements of radiation fields and statistics on maintenance and repair become available.

INTRODUCTION

At TRIUMF intense beams of 500 MeV protons extracted from the cyclotron are used to produce secondary beams of mesons for nuclear physics research. The primary radiological problem at TRIUMF is the relatively high (by accelerator standards) residual field encountered during the maintenance and repair of the cyclotron structures and associated beam lines and production target stations. Exposure rates shortly after shutdown range from 1 to 1000 mGy hr^{-1} and decay by one to two orders of magnitude during maintenance periods lasting up to two months. It is TRIUMF policy to try and limit the annual dose to any individual staff member to 10 mSv and at the same time to minimize the total man-dose to the staff. In order to comply with this policy we have developed a procedure for predicting maintenance and repair doses based on field measurements and analysis of maintenance and development requirements. Figure 1 shows a flow diagram outlining the method. From in situ measurements of gamma-ray spectra and exposure rates and the operating field history we are able to calculate the radionuclide production rates at a number of field points. Once the production rates are known the exposure rates are calculated for an arbitrary operating scenario. The exposure rates at the measurement points are then used to scale more extensive detailed residual radiation surveys in order to estimate future dose from the maintenance schedules. If these estimates conflict with our guidelines for limiting individual and collective dose we can study the effect of improved procedures and reliability of components or the effect of changes in the operating schedules in order to minimize these conflicts.

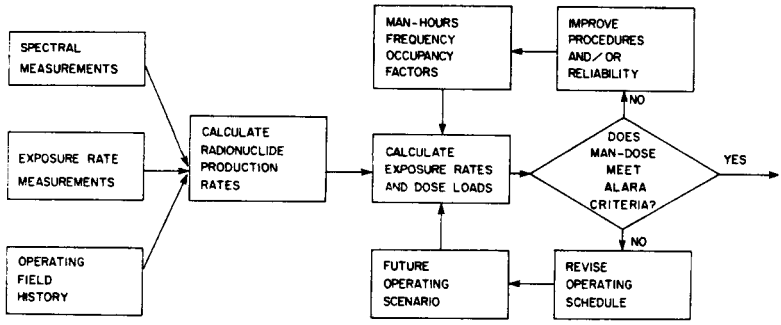

Fig. 1. Flow diagram illustrating dose planning procedure.

1. THE FIELD MEASUREMENTS

a) Induced Fields

The distribution of long-lived radionuclides induced in the accelerator structures were measured at eleven locations with a pure Ge detector having a detection efficiency of 0.48% relative to a (76.2 × 76.2 mm) NaI(TL) scintillator. A detector of such low sensitivity was chosen so that it could be operated in fields up to 1 mGy/h without appreciable counting losses. At the time of the measurement the accelerator had been shut down for 5 weeks. Measurements of the exposure rate at each location were made simultaneously using a portable G-M survey meter. Because of the small size of the Ge detector and the complex source distributions involved all geometry effects were neglected. The amount of a given radionuclide measured at a given location was calculated as an equivalent source at 25 cm using the measured efficiency of the Ge detector. The partial exposure rates due to the identified radionuclides were then calculated from the known decay schemes[1] and the sums normalized to the G-M survey meter measurements.

The most common materials present in the irradiated surroundings are concrete, stainless steel, aluminium and iron although the composition varies from location to location. Despite this the distribution of radionuclides in the induced activity is remarkably similar at all locations measured. Figure 2 shows the average fractional composition of the induced activity for the more prominent radionuclides, the error bars indicating the rms spread in the measurements. The distribution is also similar to that found by other workers.[2] The ^{59}Fe and ^{60}Co activity was especially high in the basement of the cyclotron vault where the operating low energy neutron fields are known to be high and the measurement points are close to the large mass (~3000 tonnes) of the cyclotron magnet.

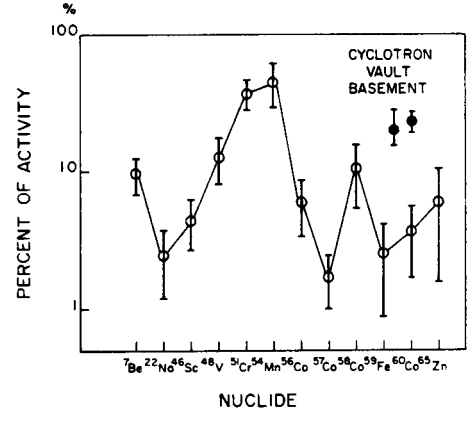

Fig. 2. Average distribution of radionuclids in the induced activity at TRIUMF.

The contribution of the short lived radionuclides was obtained from an analysis of the decay curves of the residual field as measured with TRIUMF beamspill monitors.[3] The contribution of ^{56}Mn at E.O.B. was approximately equal to the sum of the total exposure rate due to the long-lived components. In the cyclotron vault ^{24}Na contributed about 25% of the total.

b) Operating Fields

For most operating modes the ambient radiation fields are proportional to the extracted beam current. We therefore use the beam current as a measure of the operating fields. This procedure does not take into account variations in the fields due to mistuning of the cyclotron or beam lines but has the advantage of being the variable of direct interest in planning.

2. PREDICTION OF RESIDUAL FIELDS

From the measured distribution of radionuclides and the historical record of the operating fields as approximated by the beam current history we calculated the production rates for the various radionuclides using the data on half-lives in the literature.[1] These in turn allow us to calculate the residual fields for an arbitrary beam current scenario and decay period. In Fig. 3 the calculated long-lived component of the residual field at a typical location is compared to the measured values of the exposure rate. Differences between the measured and calculated values are thought to be due to beam tuning effects.

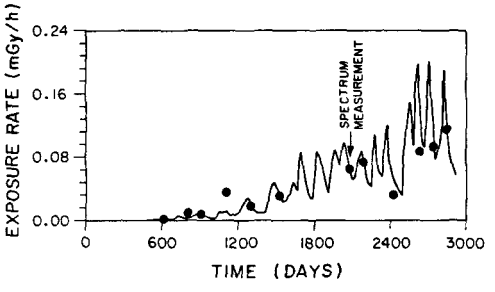

Fig. 3. Exposure rate in the cyclotron vacuum tank calculated from the beam current history. Points are measured values.

3. DOSE ESTIMATES

In order to estimate the dose which would be incurred during maintenance and repair in the active areas, the facility was divided into a number of subsystems. For each subsystem the most dose intensive tasks were identified and estimates of the frequency of maintenance and the number of man-hours per task were made by supervisors responsible for carrying out the task. In areas where there were large spatial gradients in the fields an occupancy factor was also estimated. 120 separate tasks were identified in this way and were categorized into those requiring a lengthy shutdown period (with a corresponding relief due to a long decay period) and those requiring a period of less than four days. The projected dose to personnel for each task was then calculated from the above data and detailed radiation surveys of the active areas. The frequency versus dose relationship for these tasks is shown in Fig. 4. Diagonal lines on this plot are lines of constant man-dose per year. The total man-dose predicted in this way for each of the first three years of the program was within 20% of that eventually incurred.

It became apparent that when the scaling of the residual fields as calculated for the intended future long term beam schedule were applied to these tasks those to the right of the 10 man-mSv yr^{-1} line could not be carried out after 1985 without severely compromising our dose guidelines.

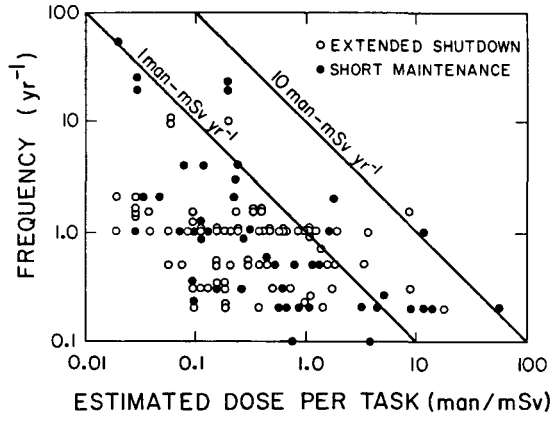

Fig. 4. Frequency vs Dose relationship for maintenance tasks.

4. PROGRAM FOR REDUCING DOSE

In order to close the loop implied in Fig. 1 a list of priorities for action was derived for the tasks above the 1 man-mSv yr^{-1} line in Fig. 4. The action taken to reduce the projected dose was of five types:

i) Make the device or system remotely handleable
ii) Redesign to reduce service requirements
iii) Redesign to eliminate the device or system
iv) Abandon the device if of limited use
v) Alter maintenance procedures, e.g. use local shielding.

Flexibility in scheduling beam was limited to shifting low intensity polarized beam periods to always occur before major planned shutdown periods to allow at least two weeks for 'cool down'. Regular maintenance periods of longer duration have been scheduled less frequently to allow at least the short-lived radioactivities to decay.

CONCLUSIONS

Estimates of exposure rates from the measured radionuclide distribution in the residual fields at TRIUMF have allowed us to make accurate predictions of the dose incurred during maintenance and development of the facility. These in turn have led to the redesign of procedures and equipment so that the total annual man-dose has levelled off despite the fact that the extracted beam current has continued to rise (Fig. 5).

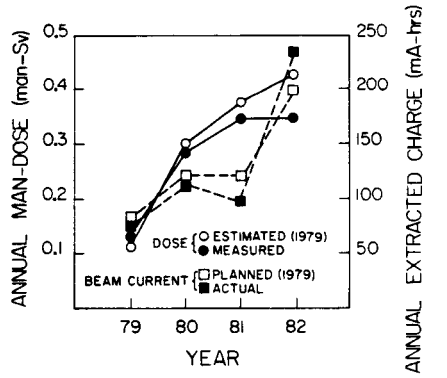

Fig. 5. Annual man-dose and extracted charge as a function of time.

REFERENCES

1. C.M. Lederer and V.S. Shirley ed., Table of Isotopes, Seventh Edition (John Wiley & Sons, New York, 1978).
2. H. Hirayama et al., Measurements of Radionuclides Induced in the Accelerator Structure of the KEK-PS, Health Physics 43, 481 (1982)
3. J. Drozdoff et al., A Microprocessor-based Radiation Monitoring System, these proceedings.

MONITORING OF RADIATION WORKERS HANDLING RADIOLUMINOUS PAINTS

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1. Introduction

Radioluminous paint is being consumed by various agencies such as watch manufacturing, telephone and defence. These paints are activated either with tritium (H-3) or promethium-147. The watch industries consume about 37 TBq of H-3 activated paint and the others use 2TBq of Pm-147 activated paint yearly. Earlier, Ra226 was in use as a luminous paint but by the end of 1969, the use of this paint was abandoned completely and H-3 and Pm-147 activated paints were introduced. This has resulted in a significant reduction in external exposure to workers. Because H-3 and Pm-147 are pure beta emitters, the assessment of internal exposure in workers poses many problems. Considerable data are available on the behaviour of H-3 inside the human system, using which it is possible to estimate the H-3 burden from bioassay results. However the Pm-147 burden is difficult to determine as the metabolic behaviour of Pm-147 activated paint is not fully understood. The published literature by Palmer(1), Arndt(2), Shipler(3) and others indicate that the inhaled activity is slowly taken by the liver and finally deposited in bone and this distribution takes years for completion. Therefore, in case of Pm-147, determination of organ activity provides additional information over the bioassay and both these results can be taken into account while determining the internal contamination in a radiation worker. Fourteen radiation workers have been monitored for Pm-147 in their livers and lungs. Bioassay was also carried out for all of these workers. Ten workers were monitored for H-3 burden by urinalysis.

2. Equipment and methods

In order to count the Pm-147 in-vivo, an attempt was made to determine the external bremsstrahlung (EB) coming out. For this purpose a phoswich detector consisting of a NaI (Tl) crystal (20 cm dia x 0.3 cm thick) backed and optically coupled to a CsI (Tl) crystal (20 cm dia x 12.7 cm thick), mounted in a 20 cm thick steel room and operating with the pulse shape discrimination technique was employed(4).

An EB spectrum includes all energies from zero to the maximum beta ray energy of Pm-147 and the percentage yield of low energy photons in 12-75 keV energy is comparatively higher than the high energy component. The counting system was calibrated in the 12-75 keV region and a minimum detectable activity of 1.11 KBq was found in liver for 40 mins counting time (fig.1). The radiation workers were counted with the system for organ activity. Simultaneously the fecal as well as urine activity of these workers were also determined. Radiation workers handling tritium activated paint were monitored on a fortnightly interval by urinalysis.

3. Results and discussion

Three workers showed significant Pm-147 uptake in liver and lung (Table I). An EB distribution obtained from one of the workers is shown in fig.2, along with the background distribution from a normal subject. The natural background count-rate in the steel room used to vary between 30 to 60 cpm in the energy band of 12-75 keV. The counts obtained with the subject in the same energy band are 194 cpm yielding 26.6 KBq of Pm-147 in liver. In the other two cases also the counts observed are quite significant and found to be 10.7 KBq and 27.4 KBq of Pm-147 in their liver. These three workers with internal contamination belong to the same

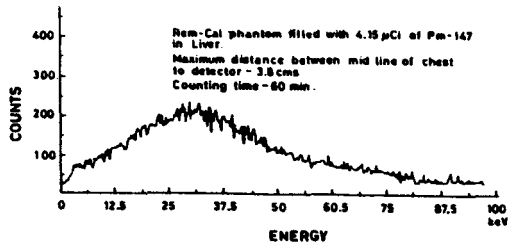

Fig.1 Bremsstrahlung spectrum of Pm-147 obtained with phoswich detector

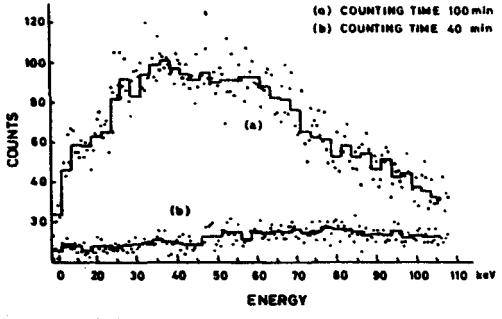

Fig.2. (a) EB detected from radiation worker'A'(b) background distribution from a normal subject of same H/W

institution where air contamination of 2.32×10^2 Bq/m^3 was determined in the workshop.

Bioassay results indicated no Pm-147 activity in urine. However, fecal activity was observed in the three workers. In order to compare the results, the liver and lung burdens were estimated from fecal results. ICRP-30 lung model was used for lung burden estimation and the model suggested by Palmer (1) was used for liver burden estimation. For both lung and liver, extreme values were calculated assuming each organ to be the sole source of fecal elimination independent of the other. Lung burden was calculated with biological half-life (Tb) of 50 to 500 days (5). Liver burden was evaluated with Tb of 660 to 1000 days (1).

The estimated organ burdens are given in brackets beneath the external counting results in Table I. Organ burdens are found to be lower than the values measured by external counting.

Since these persons were working regularly with radioactive paint, they were subjected to chronic exposure through inhalation. For calculation of dose, the chronic exposure is averaged to a single intake in the middle of the working

Table I: Results of Pm-147 monitoring of radiation workers

Radiation workers	Duration of work(months)	Time elapsed (days)	Fecal activity (dpm)	Lung activity (KBq)	Liver activity (KBq)
A	8	69	45.9	5.9 (0.08 to 0.8)	26.6 (0.7 to 1.1)
B	1	39	249.6	-	10.73 (3.9 to 6)
C	6	3	5th day 1.23×10^4	12th day 237.6	5th day 7.3	12th day 0.5	27.4 (3.8to5.7)

period (6) and lung burden is extrapolated to that point. Pulmonary deposition at the time of exposure is calculated from fecal elimination and external lung counting and are given in column 3 & 4 of Table II. Intakes in the pulmonary region are also calculated from the liver burden values of external counting, assuming

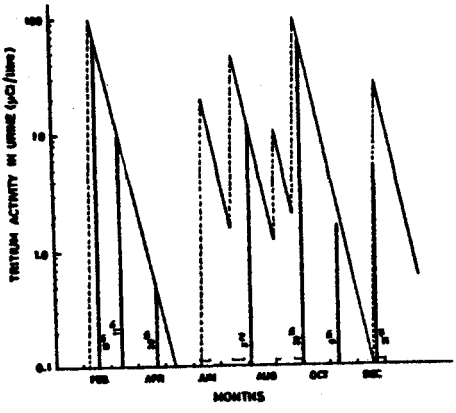

Fig 3. Estimation of H_3 intake in worker III

that 20% of pulmonary activity is transferred to blood with Tb of 50 to 500 days (5), 50% of activity is transferred to liver in 24 hrs[4] and the liver activity is eliminated slowly with a Tb of 660 - 1000 days. The long lived component of intakes evaluated from liver counting data are given in column 5 of Table II. The computed intakes from fecal results are very much less than the intakes computed from external counting. But the results computed from the individual organ activity are falling within agreeable limits.

The same workers used to handle Ra-226 paint also before the introduction of the Pm-147 paint. Therefore, they were monitored for Ra-226 burden by counting in an arc geometry with NaI(Tl) crystal detector (20.3 cm dia x 10.1 cm thick). There was no detectable activity. However, the bioassay results showed lung burden in all of the workers and it was less than 148 Bq.

Table II Estimated intake of Pm-147 in three workers

Radiation worker	Intake occured before monitoring (days)	Estimated intake from		
		Fecal KBq	Lung KBq	Liver KBq
A	160	12.3	90.3	202.9
B	50	19.4	-	123.9
C	120	64.2	131.9	216.0

Table III Results of H-3 monitoring for 4 workers

Radiation Workers	Annual intake MBq	Annual Dose mSv
I	11.3 (305.4 uCi)	12.22
II	12.9 (348.6 uCi)	13.2
III	11.1 (300.0 uCi)	11.8
IV	6.3 (170.3 uCi)	6.0

Ten radiation workers who are actively engaged in dial painting with H-3 activated paint were monitored at fortnightly intervals. The H-3 contamination in urine vary between 3.7 to 370 KBq. In one or two instances the urine concentration exceeded the limit of 1.85 MBq per litre in 1981 and 1982. The maximum concentration which was detected at any time in any subject was 2.5 MBq. These high activities were due to the reason that the exhaust system which is connected to workboxes was occasionally switched 'off' by workers during working hours, resulting in the air concentration upto 118 KBq/m^3. After restoring the normal air flow in the workshop the urine activity in workers dropped to the order of 66.6 KBq per litre.

A typical example of a subject who had exceeded the limit of 1.85 MBq per litre of urine twice in year 1982 has been represented in fig.3. The annual intake of activity and the dose for the worker is calculated from the bioassay results. Since the actual time of intake of activity was uncertain in all the cases, the annual intake was calculated on three main assumptions i) the biological half-life of H-3 in all the workers is 7 days, ii) intakes in workers occur in the middle of their working period, iii) the maximum activity in the workers at any time can never exceed 4.44 MBq (120 μCi)/ litre (as it is observed from the monitoring results). The amount of intake is determined from the urinalysis by extrapolating the elimination rate to the middle of the working period. The six peaks in the figure 3 represents the possible amount of intake while working. The Annual Intake in this case is determined by adding up the corresponding values of activity. The table III gives the annual intake as well as the calculated annual dose of four radiation workers for 1982. The results indicate high Annual Dose but well within the permissible limit.

4. Conclusion

The monitoring of radioluminous paint workers has shown that the radiation risk from working with tritium activated paint is higher than that of handling Pm-147 activated paint. The body monitoring of promethium workers by liver and lung counting provides a complementary method to bioassay, for the estimation of body burden of workers handling Pm-147 radioactive paint. The Pm-147 activity found in fecal samples indicate the slow clearance of lung activity, determination of which may provide effective means of estimating the internal contamination of workers handling Pm-147 activated paints. Workers handling H-3 activated paint need a regular programme of monitoring at short intervals. The excretion rate of tritium has been assumed to be represented by a single exponential curve for the purpose of dose calculation. Theoretically it will be wrong to consider single compartment model for a long term clearance due to two reasons i) tritium is the constituent element of body tissue and therefore, it exchanges with organically bound hydrogen and ii) the paint is an organic polymer which is insoluble in body fluid. The errors in assuming a single compartment model is yet to be evaluated. According to ICRP-30, this is not likely to be more than 10%. The annual dose calculated for H-3 workers were found to be significant even though scheme of rotation of workers was put in action. It requires continuous vigilance on the part of the health physicist to control the working conditions and ensure safe levels of body burden.

References:

1. H.E.Palmer, I.C.Nelson and G.H.Crook; Health Phys. Vol 18 (1970) P. 53
2. D.Arndt etal; Vienna, 1975, IAEA - SR-6/15
3. D.B.Shipler et al; Vienna 1975, IAEA - SR-6/28
4. R.C.Sharma et al; Vienna 1975, IAEA - SR-6/50
5. ICRP, Publication No.30, 1978, (New York:Pergamon Press)
6. ICRP, Publication No.10A, 1969, (New York:Pergamon Press)

INCORPORATION RISK OF WORKERS PRODUCING AND HANDLUNG 125-I-LABELLED RADIOIMMUNOASSAY KITS

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A survey will be presented on the occupational incorporation risk in the field of production and use of RIA-kits.

Over a period of nearly 10 years the working area and the 125-I-thyroid uptake of about 40 workers were controlled. Only in the field of hormon labelling significant incorporation levels above 5 nCi were detected. They were correlated to activity troughput, airborne contamination and contamination of the working places.

It will be demonstrated that by changing the facilities required for protection purposes, the incorporation risk was reduced in these laboratories significantly.

In the areas of manufacturing and application of RIA-kits the incorporation risk is neglegible.

125 I AIRBORNE CONTAMINATION LEVELS IN VITRO RADIOMETRY LABORATORIES

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Some recent publications have considered subjects relating to the quantity of airborne radioiodine when used in nuclear medicine, metabolic radiotherapy and protein labelling. The aim of this study is to determine the airborne concentrations of 125 I in in-vitro radiometry laboratories where millions of test tubes, containing altogether - in different solutions and concentrations - activities of the order of 4 GBq, are manipulated every year.

To evaluate the degree of radioactive contamination present in the air, a sampling device was set up, consisting of a glass cylinder within which three separate layers of active carbon were placed. Upstream, a millipore type prefilter was fitted to hold back, if necessary, any particles of 0.8 μm or larger. The active carbon filter (10-18 mesh) has a dry retention efficiency in excess of 95% (2-4).

Using the filter described, samples lasting five hours were taken on different days of the week and at various periods of the year inside a laboratory (100 m3), the number of air changes being limited to 2 per hour. The aim was to keep count of the possible variability in the quantities of manipulated radioiodine as well as of climatic conditions.

TABLE 1

PERIOD	CONCENTRATION Bq/m3 (± 10%)
APRIL	
TUESDAY	7.3 (-2)
WEDNESDAY	2.2 (-1)
THURSDAY	9.8 (-1)
FRIDAY	1.2
JULY	
TUESDAY	1.6 (-1)
WEDNESDAY	1.3 (-1)
THURSDAY	6.6 (-2)
FRIDAY	1.5 (-1)
SEPTEMBER	
TUESDAY	1.1 (-1)
WEDNESDAY	6.6 (-2)
THURSDAY	7.3 (-2)
FRIDAY	1.5 (-1)

Number in parenthesis represent powers of 10

Table 1 Concentration values in air taken in 3 different weeks during a year.

In order to effect evaluation of an equivalent dose to the thyroid following only the chronic inhalation of airborne 125 I, a discontinuous transfer function from the environment to the worker

was assumed.

The compartmental analysis carried out took into account the models presented by the ICRP 30 and, as far as inhalation is concerned, those of the 1966 Task Group (3-5-6). The compartments participating in the metabolism of iodine are shown in Fig. 1.

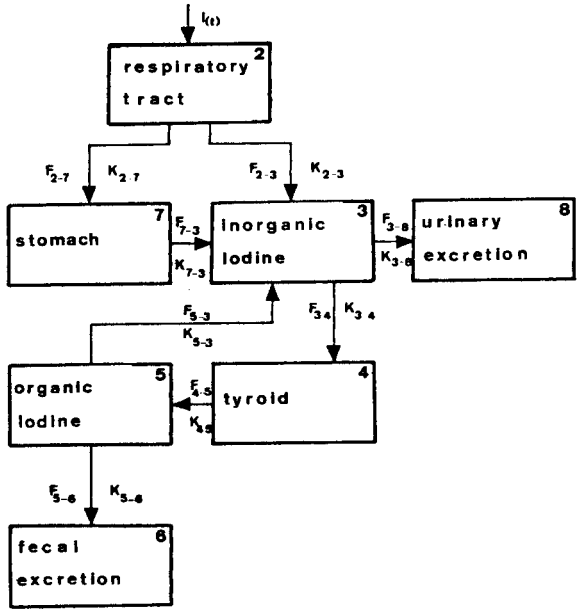

FIGURE 1 Mathematical model used to describe the kinetics of radioiodine in the body.

Transfer of the iodine, I(t), from the environment to the respiratory apparatus, is described by equation /1/ considering the prevalently gassy nature of the contaminant and its total absorption in the respiratory tract :

$$I(t) = Ca(t) * Vi * Fa \qquad /1/$$

where : Ca(t) = concentration in air of the radioisotope (Bq/m3)
Vi = inhaled volume in a unit of time (m3/h)
Fa = deposited fraction

The kinetics of iodine is described for the model presented by a system of differential equations:

$$dQ2(t)/dt = I(t) - (K2-3 + Lr) * Q2(t) \qquad /2/$$

$$dQ7(t)/dt = F2-7 * K2-7 * Q2(t) - (K7-3 + Lr) * Q7(t) \qquad /3/$$

$$dQ3(t)/dt = F2-3 * K2-3 * Q2(t) + K7-3 * Q7(t) + F5-3 * \qquad /4/$$

$$* K5-3 * Q5(t) -(Lr + K3-4) * Q3(t)$$

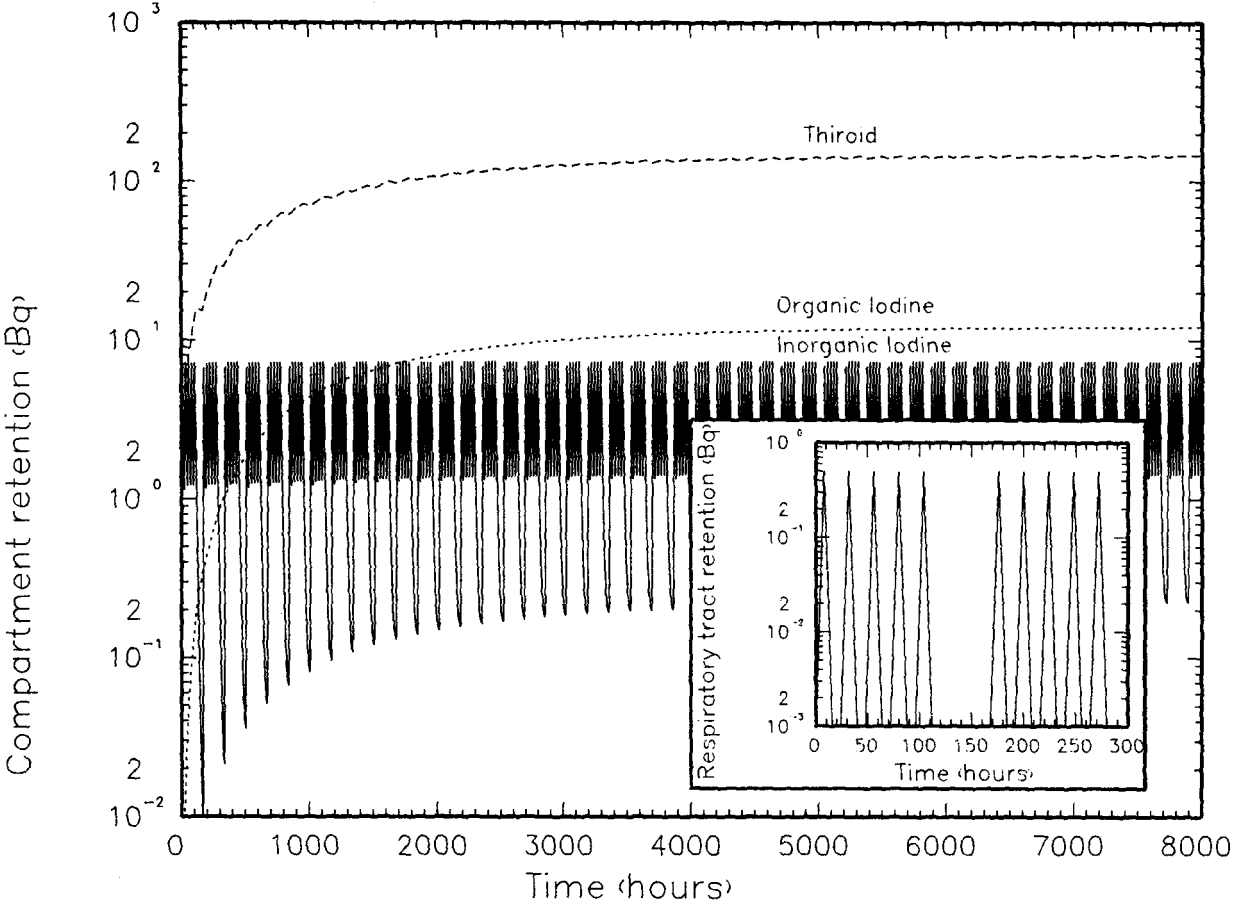

Fig 2 - Quantity of 125 I deposited in the affected compartments in terms of time

$$dQ4(t)/dt = F3\text{-}4 * K3\text{-}4 * Q3(t) - (Lr + K4\text{-}5) * Q4(t) \qquad /5/$$

$$dQ5(t)/dt = K4\text{-}5 * Q4(t) - (Lr + K5\text{-}3) * Q5(t) \qquad /6/$$

where: Lr = radionuclide decay constant.

Fig. 2 graphically represents - correspondingly to the maximum concentration value in air - the analytical solutions of the system for the most interesting compartments. The inset indicates the course of the activity in the respiratory tract in terms of time.

By means of equation /7/, knowing the quantity Qj of radio compound present at equilibrium in the j-th compartment, it is possible to calculate the annual dose in the same:

$$Dj = Kc * Qj \qquad /7/$$

where: Dj = annual dose to the j-th compartment (μSv/year);
Kc = 7.06 μSv/Bq/year, conversion factor between the present activity and the annual dose.

From what has been said above, the annual dose to the thyroid can be evaluated as 700 μSv.

In the light of the dose value obtained, it is reasonable to affirm that the radioprotection classification of the staff must, considering the almost total absence of external radiation, rule out the possibility of operators belonging to group A and, consequently, these laboratories should not be thought of as a controlled zone.

If what has been said is valild for the activities of the order of those considered by us, even more so can this classification be applied to personnel operating in laboratories where lesser quantities of radioactivity are employed.

BIBLIOGRAPHY

1) Cardozo R.L. The dispersal of radioactive matter by evaporation Healt Physics Pergamon Press, Vol. 25 (Dec.), 593-598 1973

2) Hirling J. Efficiency of various filter packings for removal of Iodine 131 and its compounds. Contained in : I.A.E.A. Proc. of Symposium treatment of airborne radioactive wastes, New York 1968

3) I.C.R.P. Report of the task group on reference man I.C.R.P. Pub. 23 Pergamon Press, Oxford, 1974

4) Murata T. Kamiya K. Removal of iodine and methyl iodine by a full sized charcoal filter. Contained in : I.A.E.A. Proc. of Symposium treatment of airborne radioactive wastes, New York 1968

5) I.C.R.P. Limits for intakes of radionuclides by workers I.C.R.P. Pub. 30 Pergamon Press; Oxford, 1978

6) I.C.R.P. Task group on lung dynamics, Deposition and retention models for internal dosimetry of the human respiratory tract Health Phys. 12, 173-207, 1966

LEVELS OF AIRBORNE CONTAMINATION WHILE HANDLING ^{125}I AND ^{131}I AND ^{99m}Tc UNSEALED SOURCES IN MEDICAL DIAGNOSTIC PROCEDURES [x]

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INTRODUCTION

The use of radioactive substances in nuclear medicine is well-known to involve some radiation hazard to both patients, staff members and a fraction of population. With commonly used radioisotopes of ^{125}I and ^{131}I the hazards from the inhalation of contaminated air are not only detectable but also significant [1].

The present studies were aimed at determining the various forms, aerosol aerodynamic size distributions, levels and localizations of airborne radioiodine and technetium in a well-equipped diagnostic nuclear medicine unit of a 1500-bed university hospital. This information helps to reduce the personnel and population radiation hazard /in agreement with the ALARA principle/.

MATERIAL AND METHODS

In RIA Lab. 1 commercially available "kits" were used for thyroid function determinations. 330 samples were automatically prepared daily, i.e. about 80,000 samples per year /total activity: 145 - 185 MBq of ^{125}I/. In RIA Lab.2 under a chemical hood proteohormones and nucleic acids were iodinated with 1480 MBq of ^{125}I per year /20 procedures/. In a ^{131}I laboratory 120 patients with thyroid cancer and other diseases were administered a total activity of 1665 MBq of ^{131}I per year in capsules and in water solution. In a ^{99m}Tc laboratory 29.6 GBq of ^{99m}Tc was used daily for scanning, i.e. about 7500 patients received 2146 GBq per annum.

Airborne radioiodine sampling was carried out:

/1/ in two RIA Labs /RIA 1 and RIA 2/ with unsealed ^{125}I sources. Simultaneously individual ^{125}I levels in the inhaled air by staff members were monitored by IDF. For comparison air under the hood was also sampled,

/2/ in the application room and in 4 out-patients diagnosed with ^{131}I, whose exhaled air as well as that inhaled /determined by the IDF/ was also monitored,

/3/ in the ^{99m}Tc pipetting room and in the scanning room.

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[x] This work was partly supported by the Alexander von Humboldt Foundation, Federal Republic of Germany.

For airborne ^{125}I, ^{131}I and ^{99m}Tc sampling the following methods were used:

/1/ a standard May-pack filter packet method [2],

/2/ personal IDF method [3], and

/3/ apparatus for measuring ^{131}I activity /in Bq m^{-3}/ in the air exhaled by diagnostic patients [4].

To determine the activity vs. aerosol size curve a five--cascade impactor was used.

In parallel measurements the air sampling time used by the standard, personal IDF and impactor methods corresponded to that spent by the personnel working with unsealed ^{125}I, ^{131}I and ^{99m}Tc sources /only standard and impactor methods/.

RESULTS AND DISCUSSION

The selected results of the mean and individual ^{125}I, ^{131}I and ^{99m}Tc concentrations at the Diagnostic Nuclear Medicine Dept., University of Göttingen are shown in Table 1 and Fig. 1.

The mean airborne ^{125}I concentrations in RIA Lab. 1 at the working sites using unsealed sources, in the centre of the labelling room at the automatic gamma counter and its vicinity ranged from 0.02 to 0.08 Bq m^{-3}. Elevated concentrations in the range of 0.2 ÷ 0.3 Bq m^{-3} were only found in the nearest vicinity of the automatic pipetting device; there the contribution from the elemental ^{125}I was also predominant /i.e. 69 ÷ 82%, mean: 77%/, while that at the automatic gamma counter was 51%. In the doctor's room the organic form of $CH_3{}^{125}I$ was predominant /86%/. ^{125}I activity distributions vs. aerosol size for two days /A_1, A_2/ are shown in Fig. 1a.

The mean airborne ^{125}I concentrations in RIA Lab. 2 during manual iodination under a chemical hood were 0.5 Bq m^{-3} /55% $CH_3{}^{125}I$/ and fell to 0.2 Bq m^{-3} 21 hrs after the completion of iodination. It was 10-fold higher than that in RIA Lab. 1 but comparable with that found in the vicinity of the automatic pipetting device in Lab. 1. Under the hood airborne ^{125}I concentrations during ^{125}I procedures ranged from 177 to 5076 Bq m^{-3} /30 ÷ 74% I_a and 24 ÷ 48% I_2/ depending on the labelling technique used. It fell rapidly to 1.8 Bq m^{-3} and 0.03 Bq m^{-3} 72 hrs after the completion of iodination. The corresponding aerosol distribution /B/ is shown in Fig. 1a.

Airborne ^{125}I concentrations for individual people handling unsealed sources in RIA Lab. 2 were found to vary from 13 to 302 Bq m^{-3} /working time varied between 0.25 and 4.7 hrs/ and were much higher than those while operating the automatic pipetting device in RIA Lab. 1, i.e. 0.3 ÷ 0.5 Bq m^{-3} for 32 to 68 hrs, resp.

The patients who had received diagnostic ^{131}I capsules constitute airborne contamination sources. Thus

/1/ Within 48 hrs a patient exhales about $3 \cdot 10^{-4}$ of the activity

administered to him, which corresponds well to the previous work of Krześniak et al. [4] ,

/2/ The mean individual airborne concentration during 24 hrs in the patient's nearest vicinity is of the order of 20 Bq m^{-3}, as measured with the personal IDF, whereas it drops to 6 Bq m^{-3} during the period between 24 and 48 hrs,

/3/ The mean airborne concentration in the room during 48 hrs is 3 Bq m^{-3} as measured with the standard method,

/4/ The percent contributions of all the three forms of ^{131}I during 48 hrs were:

	air exhaled	air in the room
I_a	1 - 27%	12 - 47%
I_2	4 - 66%	11 - 46%
CH_3I	12 - 96%	37 - 76%

The above distribution of the three forms of airborne ^{131}I depends on that found in the air exhaled by the patients. The relative aerosol probabilities in air samples also vary as a function of the aerodynamic diameter of aerosols. For the first few days after the oral administration of ^{131}I, as shown in Fig. 1 /curves A_1 and B_1/, they are similar, and the corresponding aerodynamic diameters are given in Table 1. Similar measurements for ^{99m}Tc are presented in Table 1. Airborne ^{99m}Tc concentrations were found to be of the order of 1 Bq m^{-3}. The behaviour of ^{99m}Tc in the air is not known, yet its percent contributions on the three layers of the May-pack filter are: 28.17 and 57 resp. The activity distributions vs. aerosol aerodynamic diameter in the Tresorraum and patients waiting room are shown in Fig. 1c.

RESULTS

/1/ In spite of better ventilation in RIA Lab. 2 individual airborne ^{125}I concentrations during handling unsealed ^{125}I sources are comparable to those in other laboratories,

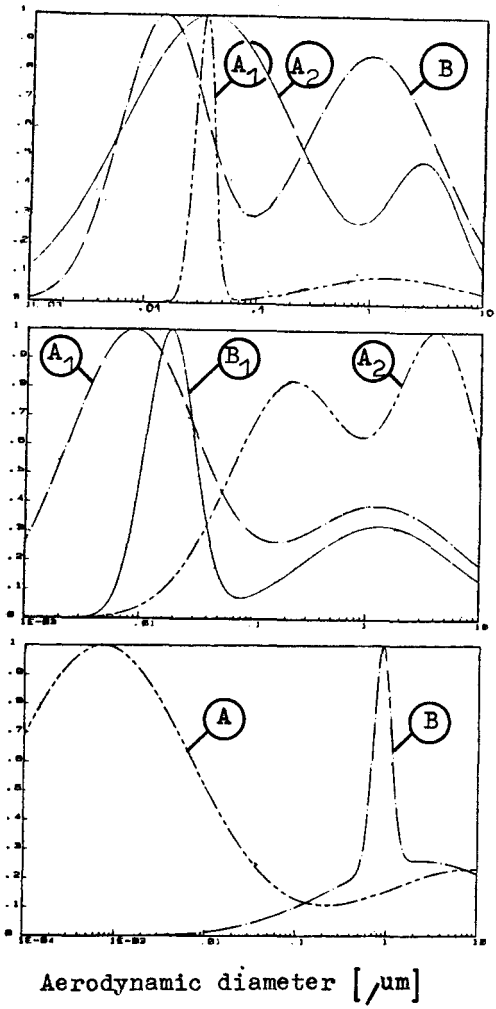
/2/ Individual airborne ^{125}I concentrations are much higher than the mean concentrations in rooms,

/3/ Workers are exposed to ^{125}I inhalation only during actual handling procedures,

/4/ An automatic pipetting device together with good ventilation allows the mean airborne ^{125}I concentrations to drop considerably,

/5/ ^{125}I and ^{131}I activity distributions as a function of the aerodynamic aerosol diameter may be described by two normal-log curves, the median values at 0.02 /um and 1 ÷ 5 /um while those for ^{99m}Tc have medians at 0.001 /um and 1 /um.

/6/ The patients who had received diagnostic ^{131}I capsules constitute airborne contamination sources.



a/ ^{125}I

A_1, A_2 - in air RIA Lab. 1 1^{st} and 2^{nd} day;

B - in air RIA Lab. 2.

b/ ^{131}I

A_1, A_2 - in air Lab. ^{131}I 1^{st} and 2^{nd} day /patient D.M./;

B_1 - as above, in air 1^{st} day /patient M.A./.

c/ ^{99m}Tc

A - Waiting room,

B - Tresorraum.

Fig. 1. Log-normal particle size distribution of radioactive aerosol in rooms as a function of the Aerodynamic Diameter.

LITERATURE

1. Glöbel et al., Proc.17 F.f.S., Aachen 8-10 Juni 1983.
2. Krześniak J.W., Porstendörfer J., Nukleonika 1979, 3, 259-269.
3. Krześniak J.W. et al., Proc.III WCNMB, Paris 1982, 2985-2988.
4. Krześniak J.W. et al., Nucl.Med.XVIII, 5, 1979, 246-251.
5. Report 23 ICRP, Oxford 1974.
6. Str.Sch.V, 13 Okt.1976, BGBI, I., S.2905.

IMPLEMENTATION OF THE NEW IONISING RADIATION REGULATIONS INTO A PRACTICAL SYSTEM OF PERSONAL DOSIMETRY

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Introduction

The United Kingdom Health and Safety Commission's consultative document, entitled 'The Ionising Radiation Regulations 198-', was published in 1982 (Ref 1). After due process of consultation, this document which is based on the Euratom Directive of 1980 and on ICRP publication 26, will form the framework of new legislation on the control of workers potentially exposed to ionising radiations. The final date for implementation of this legislation is not yet clear but we at Winfrith are planning for a target date of January 1985. This paper reports on the steps taken to date to produce a practical, tested system for personal dosimetry which should meet the anticipated requirements by that date. The paper concentrates only on those aspects of the regulations concerned with personal dosimetry.

AEE Winfrith is a medium sized nuclear research establishment of the UKAEA and has approximately 1000 radiation workers, potentially exposed to a wide range of gamma fluxes (10 keV to 6 MeV) and a small external neutron dose hazard. Internal doses may arise from a wide range of fission products, neutron activation products and from uranium and plutonium oxide fuels.

Requirements

Up to the present Winfrith, in common with most other laboratories has discriminated between external and internal personal dosimetry. Operating under the Factories Act Legislation for external dosimetry, we compile legal dose records for the individual radiation worker (Ref 2). Internal exposures are controlled under a more 'ad-hoc' basis operating under a system of Derived Secondary Limits (roughly based on ICRP 10). Records are kept in a wide range of units depending on the type of monitoring; dose, in rems, is rarely recorded.

In the future we will be required to ensure that the sum of external and internal (committed) doses do not exceed legal limits and that, lower investigation levels are also observed. The legal limits are those specified in ICRP 26.

If any component can be demonstrated to be less than 10% of the legal limit it need not be monitored. The philosophy, techniques and measurement protocol and the record keeping system will all operate under "approved laboratory status".

Philosophy

It will be acceptable practice, in order to demonstrate compliance with the dose limits, to assess secondary 'dose quantities' and to use comparative measurements. Thus the dose limit inequalities can be rewritten as the sum of the ratios of measurable qualities to the limit of such qualities.

$$\frac{H_{EXT}}{H_{EXT\ L}} + \sum_i \frac{C_i t_i}{2000\ DAC_i\ hr} + \sum_j \frac{B_j(t)}{B_j\ L(t)} + \sum_k \frac{U_k(t)}{U_k\ L(t)} + \ldots\ldots \leq 1$$

with $i \neq j \neq k$ etc: L stands for a limit; C_i is airborne concentration of isotope i whose derived air concentration is DAC_i and the exposure time is t_i: $B_j(t)$ is the amount of isotope j in the body, after allowance is made for previous intakes and $B_j.L(t)$ is the limit at time t: Similarly U refers to urinery excretion measurements and terms can be added for faecal sampling etc. If i = k = j etc i.e. when two or more methods are used for assessing the same isotope, a weighted mean term is employed.

In using the above inequality as a basis for a system of dosimetry we are depending on the validity of the biological models of behaviour as specified by ICRP 30 or by our own models. Our experience is that the higher exposures, arise from specific, recognised incidents and that it is usually possible to observe biological clearance in the individual involved. In practice we would operate the routine system at a lower level (probably at the investigation level of 3/10) and revert to individual dosimetry at higher levels.

Practical Systems

In any practical system it is necessary to discriminate between the day to day control and the end of the year legal records. Dosimetry information, in many forms, will be fed into a data base. Some of this information, especially, that from workplace monitoring, will be used for day to day control whilst other information, e.g. biological monitors, will be used for the annual records. Short term dose control is already practiced in some areas of the Winfrith Site and is already computer orientated. Similar facilities will be extended to other areas using the appropriate monitoring devices.

In principle we can allocate probable components of individual doses at the begining of the year by using past radiation records of specific work areas and specific tasks within those areas. Derived investigation levels could then be set for each component and these levels used as crude working limits to trigger off further investigations or a full dose update. The prefractionation approach could have severe reprecussions on working practices and if used it is felt essential that facilities are available to rapidly reassess the dose components during the year as the information becomes available from the data base.

Short term dose control also arises from the use of contract labour. Such labour is often used for short periods for specific tasks. These times are too short for an assessment of internal dose using the present routine urine and body monitoring schedules. Improvements in local screening of body contents will be made in some facilities prior to the target date of Jan 1985.

Plutonium intakes have long been recognised as the most difficult problem under the new regulations. The problem arises mainly because of lack of sensitivity in monitoring techniques. At Winfrith we are more fortunate than most laboratories in having a small defined work force and a high fired, high burn up oxide fuel which is relatively easy to detect in the lung and whose behaviour in the lung is fairly well categorised (Ref 3). Day to day control is via personal air samplers and installed air sampling systems. Air sampling will continue to be used to instigate faecal sampling, nose blows and lung monitoring following suspected incidents. Routine urine monitoring and lung monitoring, in the absence of recognised incidents, are used as long term back up to confirm the absence of slow chronic build up of material in the body. From the records of the laboratory over the last twenty years we are confident that no serious chronic exposure situation exists and that small intakes are a result of recognised incidents which can each be investigated and documented. The time scale before we can confidently enter a dose record may be long. This time scale, in excess of 1 year, means that retrospective modification of dose records will be required in the new system.

Dose Records

The need for a rapid response time in the system for re-evaluation of current dose status on the individual has been stressed above. Such a system is computer based. The legal, annual dose records - which will require considerable changes and enhancements - are also computer based. Similarly, with the exception of personal air samples, all the dosimetry laboratories use compatable minicomputer systems for data acquisition and storage and these computers are directly linked to the mainframe machine (ICL 2976) which holds the dose records and associated codes. Thus the basis of the enhanced record keeping systems already exists without need for major investments in hardware.

The main requirement has been identified as developing a common compatible database to support input, specific to any individual, from all laboratory systems, to be used by main frame codes on metabolic models, dosimetry calculations and record keeping and to act as a buffer for day to day information of individual radiation dose statistics. This data base, using the 2976 has been demonstrated as a pilot scheme and is now being extended to all radiation records.

Input, under a unique personal dosimetry number will arise from:-

External dosimetry laboratory	-	Legal dosimeters for γ, n, shallow and deep doses
Radiation areas	-	Secondary dosimeters

	- Chest monitoring
	- Air Sampling
	- Incident details/abnormal events
Radiochemistry laboratory	- Urine, faeces, nose blows
Body Monitoring Laboratory	- Organ and body contents

Output, from the codes using the data base will be in the form of annual dose records and interim dose status statements. The interim statements, available for interrogation, under password security, by authorised staff will give the current status on specfic individuals together with time of compilation, flags as to approach to statutory or derived working limits, the need for special sampling schedules, missed samples etc. The format of annual dose records, which are the legal requirement, have yet to be confirmed. Present thoughts are that they will consist of a three part document.

a) A summary form giving all personal data together with a statement of lifetime dose, annual dose and annual dose limits. It will be necessary to quote both effective dose equivalent and non stochastic organ doses. Although most parameters in the dose inequality expression are not measured or expressed in dose terms, this summary statement will be in sieverts.

b) The external dose, as a monthly record, will as at present, list gamma and neutron whole body doses and extremity doses.

c) The internal record will list monitoring schedule, isotopes, limits of detection together with best estimates and errors: It will be necessary under the regulations to keep other records e.g. metabolic models used, calibration data, traceability of standards. These and other data will be kept by the individual laboratories.

Conclusion

The system proposed, a refinement of existing systems, is being built up throughout 1983 and will be run in parallel with existing systems in 1984. Approval for the dosimetry laboratories will be sought in late 1984 aiming to change over from January 1985. Experience to date indicates that these target dates can be met providing the software development effort can be maintained.

References

1. HSE Consultation Document on Ionising Radiations. HMSO Nov 1982
2. Preston H E and Gill D W. AEEW R 1395. 1983
3. Ramsden D. IRPA Jerusalem 1980

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EXPOSURE TO IONIZING RADIATIONS OF HEALTH WORKERS: INDIVIDUATION OF "AT RISK ACTIVITIES".

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Radiation exposure received by workers in 24 of the Hospitals in the Tuscany Region have been analyzed basing on more then 100.000 monthly individual controls (total boby), obtained with fil-badge dosimetry since 1975 to 1983.

Data are grouped according to different creteria: working environment; classes of work ers (Physicians "P", Technicians "T", Nurses "N").

We have considered doses exceeding the maximum permissible dose for occupationally exposed workers and doses in the ranges (in mR) 10÷40; 40÷100; 100÷250; 250÷400: percentage of controls in such ranges are reported in Tab. 1, together with the total number of the controls for each class of worker. Data clearly show that among the workers subjected to periodic individual control of the X Rays exposure only a small percentage "actually" works in situation of risk possibility.

	Ph.	T.	N.
Orthopeady	19/55	0/30	5/33
Neurosurgery	5/10	==	==
Dentistry	3/21	0/2	1/10
Cardiology	17/48	0/24	2/43

If we want to identify among various radiological activities, the "at risk activities" it is necessary to define then,for example, as "activities" for which the number of the individual controls >250 mR/month exceeds the 1% of the total controls.If we accept this definition, we identify as "at risk activities" the radiological activities in the Departments of Orthopeady, Neurosurgery, Cardiology. (Only in a first time also Dentistry). For this departments we report in over table the number of people which during 7 years, had controls with doses > 250 mR respect to the total mean number of people in the class: only a small group of people working in "at risk" departments shows relevant doses and that physicians are the most exposed. Moreover a long time analysis has shown a decrease in the number of significant exposures because of improvement of professionality of workers, of use of new diagnostic methods (i.e.CT) and principally because of radioprotection instructions.

MONTHLY INDIVIDUAL CONTROLS

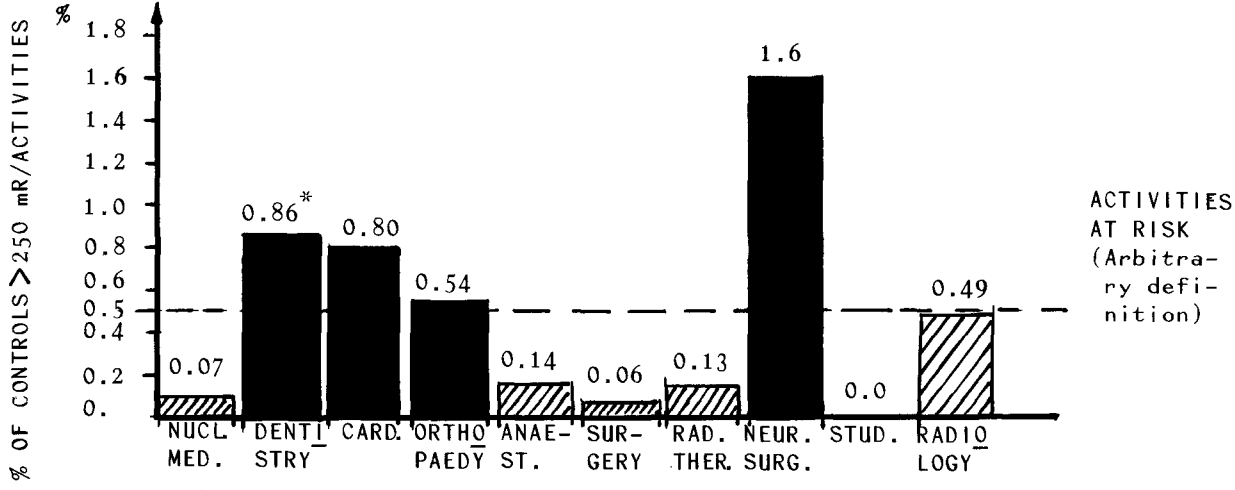
WORKER CLASS	EXPOSURES (MILLIROENTGEN)					LOST	TOTAL
	< 40	40÷100	100÷250	250÷400	> 400		
PHYSICIANS (P)	46.638 25,4%	1.766 3,2%	856 1,6%	104 0,2%	220 0,4%	4.991 9,2%	54.575 100%
TECHNICIANS (T)	35.858 90,1%	1.074 2,7%	358 0,9%	37 0,2%	111 0,3%	2.315 5,8%	39.753 100%
NURSES (N)	39.129 90,1%	753 1,7%	208 0,5%	15 0,1%	48 0,1%	3.246 7,5%	43.396 100%
TOTALS	121.622 88,3%	3.593 2,6%	1.422 1,0%	156 0,1%	379 0,3%	10.552 7,7%	137.724 100%

EXPOSURE CLASS

Tab.1

INDIVIDUATION OF RISK ACTIVITIES

	NUCL. MED	DENTI-STRY*	CARDIO-LOGY	ORTHO-PAEDY	ANAE-ST.	SUR-GERY	RAD. THER.	NEUR. SURG.	STU-DENTS	RADIO-LOGY
PHISICIANS (P)	2788	1893	3987	7178	5970	6871	2522	1019	9	22348
TECHNIC.(T)	2285	193	2379	4729	3033	4390	2591	46	158	19949
NURSES (N)	3059	954	3310	3567	1033	5125	7267	26	2289	16766
TOTAL	8122	3040	9676	15474	10036	16386	12380	1091	2456	59063

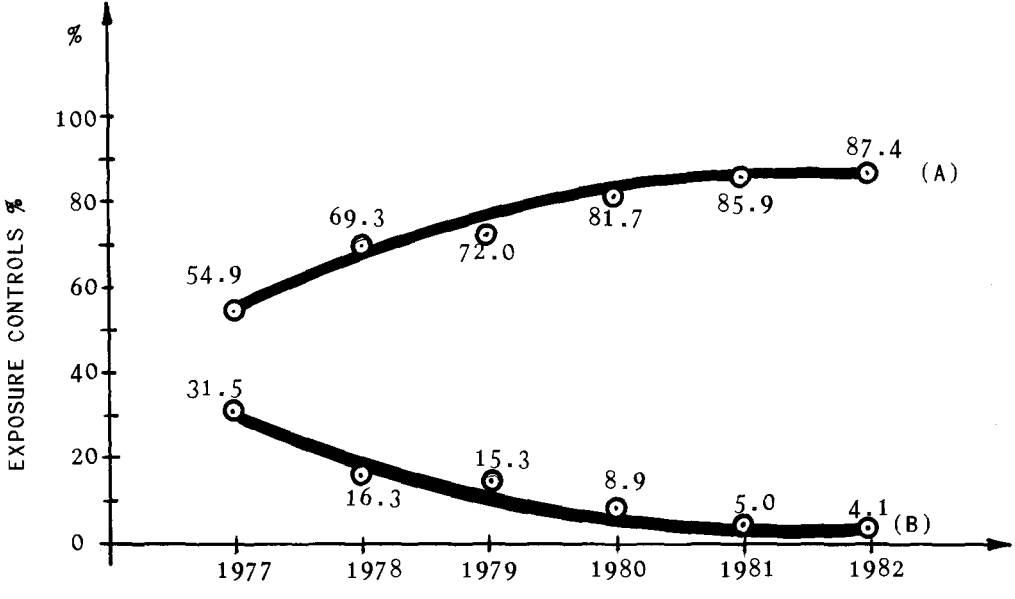


(*) Only for the first control period. Actually no at risk activitie.

Tab. 2

(A) EXPOSURES $\leqslant$ 10mR

(B) 10 mR $<$ EXPOSURES $\leqslant$ 40 mR



Tab. 3

PREDICTIVE VALUE OF PLETISMOGRAPHY IN CASES OF EXTREMITIES PARTIAL EXTERNAL IRRADIATION

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INTRODUCTION

External irradiation in extremities, particularly in hands, is the most frequent radiological accident (1-3). These accidents have predominantly occurred among workers in industrial and medical uses of radiology. Under these circumstances, the estimation of the dose distribution on the irradiated body segment becomes prioritary. For such purpose, some physical dosimetry techniques are applied, such as the processing of ring thermoluminiscent dosimeters and the reconstruction of the accident with an equivalent tissue phantom. A chronometrical analysis of clinical events in skin (depilation, erythema, etc.) serves to complete the estimation. Additionally, external irradiation in extremities calls for the evaluation of the degree of severity of the damage suffered by the connective-vascular tissue. Studies on peripherical circulation serve to evaluate the vascular harm and, consequently, the depth degree of the irradiation.

Various techniques allow for evaluating changes in peripherical circulation: radioisotopic studies on vascular permeability, cutaneous and subcutaneous temperature measurements, etc. Other useful techniques -such as ultrasounds and thermography- allow for detecting deep tissue harm, although they cannot be used in evaluating peripherical circulation. In order to overcome these limitations, pletismography has been applied as a technique serving to record a physiological variable of peripherical circulation that, in turn, serves to evaluate the magnitude of the harm and to predict the evolution of non-stochastic effects produced by external irradiation on extremities. Some results have already been published of preliminary records obtained by means of photoelectrical pletismography during studies performed on neoplastic patients treated with radiotherapy (4).

MATERIALS AND METHOD

A pletismographic system was used composed with a photoelectrical transducer connected to a recording step (5). The principle of its operation is based on the reflection of a white light beam collimated on a sufficiently irrigated body area, next to the epidermis (6). The variations of the blood volume due to the mechanic action of the heart muscle produce changes in the optic density of the iluminated area, which are detected by the photoelectrical transducer. The analogical signals thus obtained are filtered through a integrating-differentiating system and then processed by means of an operational amplifier equipped with feedback. The resulting information is shown in a high-persistence oscilloscope and/or a high-velocity recorder adequated for processing bioelectrical signals.

Assuming that the hands of workers risking overexposure are the the areas of the body most frequently analyzed, considering their

tasks, the transducers were adapted for a pletismographic monitoring of the fingers. They are small devices, easy to apply and may also be used for surface measurements in arms, legs and thorax. Pletismographic recording is a periodical process, whose basic model is progressively modified as the wave flows along vessels of different widths. This is mainly due to the elastic properties of the vessel, in addition to the blood flow and to friction and damping phenomena (7). Angles tend to disappear in the peripheral vessels, while the record contour finally adopts a triangular conformation. The exact nature of the vessels originating the blood pulse volume in skin is still under discussion. However, some authors postulate that the record originates at the bed adjacent to the transducer, in vessels with diamaters ranging between 40 and 300 µm(8).

A pletismographic recording was performed for every accidental situation, even in those cases in which overexposure was suspected to have occurred on extremities. The record is obtained by placing the transducer on the skin surface, in which arterial pulsation has been previously palpated. Usually, a pletismogram involves drafting at least 100 pulsation waves. The recorded analysis was compared with:

a) the expected pulsation wave;

b) pulsation waves obtained in homologous areas of both extremities;

c) pulsation waves recorded at the same position at different times.

The following parameters were analyzed for each record:

a) Amplitud, in millivolts;

b) Period, in seconds;

c) Time interval, in seconds, between the beginning of the wave and the point of maximum amplitude;

d) The wave width at a tenth and a half of the amplitude.

Also, an analysis was made of the relation between the pulsation wave parameters and the clinical events observed in skin.

RESULTS

Pletismographic recording was performed in the hands of 50 non-exposed people and of 15 supposedly-overexposed people between 1978 and 1983.

Table 1 shows some of the characteristics found in the group of supposedly irradiated workers.

Table 1

Task	Irradiation Source	Estimated dose (Gy)	Duration of irradiation recording
Medical radiology	X-rays (70 KeV)	1.00	15 days
	Co-60	25.00	4 days
		7.00	3 days
		1.08	14 days
		1.15	21 days
Research and Development	X-rays (20 KeV)	12.00	5 days
		0.20	5 days
		0.40	5 days
Industrial Gammagraphy	Ir-192	13.00	1 days
		4.00	1 days
		0.25	7 days
	Cs-137	15.00	10 years
		0.40	7 days
		0.00	4 days
		0.00	4 days

In most cases, pletismographic recording was performed before the dose was known. In all those cases in which non-stochastic effects were observed in skin (erythema, dry or humid desquamation, ulcer, necrosis), the following records were obtained:

a) Changes in the shape of the expected pulsation wave, within the area of the injury;
b) Significant differences between the parameters of the pulsation wave in the overexposed area and homologous areas, particularly in their amplitud and period, which are more easily and directly measured. In every case, the non-exposed areas show reduced variability;
c) Sequential modifications in amplitude and period, from records obtained in the same overexposed area, analyzed at different times. The changes observed correlate with the evolutional severity of the injuries.

During the acute phase, amplitude increases were initially found in every case. Later on in this phase, both increases and decreases were recorded. The modifications observed are generally proportional to the intensity of the harm. When the inflammatory reaction decreases and the injured tissue tends to restore, there is a normalization of period and, later, a normalization of the amplitude. The persistent reduction in the amplitude may be an indicator of residual harm.

Those pletismographic records in which dosimetric reconstruction showed no irradiation in extremities gave no evidence of changes in the pulsation wave parameters. No modifications were either observed in the pletismogram applied on surface injuries, as it is the case of dry radioepidermitis.

The quantitative analysis of the records was distributed into three groups (Table 2): I. Pletismogram of the homologous non-irra-

diated areas, which do not differ from that obtained from non-exposed persons. II. Records of overexposed areas showing acute non-stochastic effects in skin (6 cases). Changes in the pulsation wave were observed in these cases, before the injuries appeared, as from the very beginning of the recording. III. Records of areas with late effects, without acute manifestations (1 case). Evidence of residual harm.

Table 2

Group	Amplitude (mV) $\bar{X} \pm s$	V%	Periodicity (sec) $\bar{X} \pm s$	V%
I Non-irradiated	15 ± 3	20	0.7 ± 0.1	14,28
II Early record	25 ± 13	52	0.3 ± 0.28	93,30
III Late record	10 ± 2.5	25	0.7 ± 0.1	14,28

CONCLUSIONS

Although the number of cases analyzed was small, the clinical and dosimetric observations performed allow for making the following statements:

a) Pletismography may serve to evaluate the evolution of non-stochastic effects in skin, except for those showing surface injuries.
b) Changes in the pulsation wave were observed in all those cases receiving doses of 4 Gy or higher at certain locations in hands.
c) The variations in the pulsation waves indicate which are the overexposed areas before clinical injuries appear on the skin.
d) The sequence of changes in amplitude and periodicity serves to predict how the evolution of the injuries will take place. Less intensive inverse changes in amplitude and periodicity were observed for patients with vascular reactivity. The initial recovery of period during medical follow-up is a signal of a favourable forecast, later confirmed by the normalization of the amplitude. The reduction of the amplitude in the pulsation wave may be an indicator of residual injury after irradiation.

BIBLIOGRAPHY

1.- United Nations Scientific Committee on the Effects of Atomic Radiation. 1982 Report to the General Assembly, Annex H, 392-393 p.
2.- Sagan, L.A. and Fry, S.A. Nucl. Saf. (USA) v.21: 562-569, 1980 No. 5.
3.- Gimenez J.C. et al. X Reunión Científica de la Asociación Argentina de Tecnología Nuclear, Bahía Blanca, Argentina, Nov. 2-6, 1981.
4.- Zanelli G.D. Br. J. Rad. (U.K.) v.50: 68-70, 1977, No. 1.
5.- Strong, P. "Biophysical measurements" Tektronix Inc., 1971.
6.- Challoner, A.V.J. and Ramsay, C.A. Phys. Med. Biol. (U.K.) v.19: 317-328, 1974, No. 3.
7.- Selkurt, E.E. "Fisiología" Buenos Aires, El Ateneo, 1977.
8.- Gyntelberg, F. et al. Scand. J. Clin. Lab. Invest. v.33: 45-51, 1974.

LENS OPACITIES IN MAN AS A POSSIBLE INDEX OF RADIATION EXPOSURE: A CONTRIBUTION FROM FOLLOW-UP OF SOME IRRADIATED PATIENTS

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Slit-lamp microscopy of the crystalline lens in normal individuals often reveals the presence of small punctate opacities which are considered physiological and congenital. Investigations carried on workers professionally exposed to photonic radiations employed for medical purpose as well as on unexposed subjects show a high frequency of punctate opacities. These opacities seem to vary depending on age and not on the duration of radiation exposure.

In order to verify this issue we compared a group of workers professionally exposed to radiation to a group of patients undergoing radiation therapy for different types of tumors. We examined two groups of patients: the first one was composed bu 21 patients irradiated for head tumors (hypophysis, rhinopharynx, brain, paranasal sinuses) and the second one consisting of 22 subjects undergoing radiation therapy for neck and mediastinal tumors (tymoma, Hodgkin's disease, etc.). The therapeutic fields of radiation were classified according to the distance from the lens as "near" for the former group of patients and "distant" for the latter one.

All patients but one who received linear accelerator radiation therapy were irradiated with Co^{60}. For "near" fields the tumor dose ranged from 54 Gy to 70 Gy with field ranging from 5 x 7 (epipharynx) to 13 x 18 (whole brain treatment). For "distant" fields the tumor dose ranged from 45 Gy to 64 Gy with fields ranging from 30 x 30 (Hodgkin's disease) to 12 x 13 (thyroid). In all cases the patients received from 30 to 35 fractions.

A dosimetry of the crystalline lens by means of thermoluminescent dosimeters (T.L.D.) was performed on the Rando-Alderson phantom. The T.L.D. readings outside the radiation fields were compared with calibration measurements obtained by a ionization chamber applied to the standard polystyrene phantom. The simulation of the therapeutic fields required the use of radiation fields suited to the size of the phantom. Further measurements near the lens were obtained "in vivo" to test the difference of the per cent dose to the lens measured on the phantom. This technique revealed a lens dose for "near" fields varying

from 5% to 27% of the maximum tumor dose with Co60 and from 1% to 7,5% of the maximum tumor dose released by Rx 9MV delivered by a linear accelerator (one case). In the 21 patients irradiated for head tumors an increase of punctate opacities was detected in 70% of cases at the end of the treatment and/or in the follow-up after three months.

In the 22 subjects undergoing radiation therapy with "distant" fields the lens dose varied from 2,6% to 2,8% for neck treatment and 5,2% and 6% of the maximum tumor dose for mediastinal Hodgkin's disease and tymoma with linear accelerator (one case) and Co60 respectively. The evident difference between the neck nodes treatment and the mediastinal fields is due to the fact that the dose to the lens depends on several factors given by the energy of the beam and the size of the source, the size and the shape of the field, the distance from the geometric border and the composition of the diffusion mean. Fifty per cent of these patients showed an increase of the lens opacities at the end of the treatment and/or at the follow-up after three months.

In conclusion, we could observe that our patients revealed within 3 months an increase of lens opacities in 70% of the subjects treated with "near" fields and in 50% of those treated with "distant" fields. In all subjects the dose at the level of the lens was not negligible varying from a minimum of 0,45-0,50 Gy for a 4 x 4 cm. field (as that employed for the treatment of a hypophyseal tumor) to a maximum of 13,5-14,5 Gy for a whole brain treatment field.

The regular increase of punctate opacities associated with radiations showed the existence of a possible relationship between high radiation doses and punctate opacities. This relationship cannot be easily explained. In fact, the hypothesis of a direct damage to the lens germinal epithelial cells seems unreliable for the short interval of time occurring before the onset of the punctate opacities (germinal epithelial cells mean turn-over in man extrapolated from radiobiological investigations ranges from one to two years). Thus, indirect mechanisms given by alterations of metabolic and nutritional processes and radio-induced lesions of the ciliary processes could be invocated.

Further studies are required at this purpose; in the meantime the follow-up of the forementioned patients is continuing. However, in our opinion, the research of the punctate opacities in radio-exposed individuals may be a valid parameter or at least a factor deserving further consideration in order to identify an index of radiation exposure of the lens.

DECONTAMINATION OF WOUNDS AND PROBLEMS OF DIFFERENTIAL DIAGNOSIS AT EXTERNAL IRRADIATION

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In praxis, we found very often the associated problems, bound for performation of external decontamination and determination of the characteristic changes, caused by occupationally exposure to ionizing radiation.

The decontamination treatment presents, in any case, the most effective medico-prophylactic procedure, and the basic problem in praxis was an unadequate evaluation of the effect, because the decontamination was reduced on phenomenological observation. The relevant evaluation of decontamination efficiency made by use of different procedures and means, was possible in experimental conditions, and that is what makes the problems of extrapolation on people.

In cases of occupationally exposure to radiation, one from the very important and insufficient analysed and defined problems, was the differential diagnosis in the frame of the postradiation effects and changes caused by other harmful factors.

METHODS

Methodology includs the experimental study and the analysis of epidemiological data taken from the human praxis.

Experimental: The experiments were performed on the white male rats, narcotized with urethane. Each experimental group counted fifteen animals. On the shawed region between scapulae was excised skin with a special circular perforator, but the thermal burn was made by use of Deshery thermocautery at temperature of 600°C. Contamination was carried out by applying on the damaged surface($2cm^2$) of solution $^{85}SrCl_2$ (pH = 6.0) or NaI^{131}(pH = 7.0) with radioactivity off 1,85 MBq. Treatment of radiomixte based on rinsing action by use of vacuum system with 20 ml choosen solution for decontamination.

Human examination: Observations were selected according to occupation, sex, age, character of exposition, film badges and thermoluminescent dosimeters data of the received doses. The investigations at 30 radiation workers (production of radioiodine in the Boris Kidrič Institute and applications in radiation research) included: (a) the clinical checkup towards to the physiological systems (skin, eyes, cardiovascular system, neuro-endocrine system, haematopoitic tissue; (b) biochemical analyses (cholesterol, glucose, SGOT, SGPT, hepatogram and other; (c) radio-immunological analyses of hormones (triiodothyronine, thyroxine and other); (d) haematological examinations (numerical variationes of the peripheral blood cells, incidence of cytomorphological aberration,changes in values of haemoglobine, haematocryte, bleeding and clotting time).

RESULTS

Experimental data concernig the effects of decontamination of radiomixte on contamination with single radionuclide are shown in

TABLE I. EFFECT OF DECONTAMINATION OF THE RADIOMIXTE
(in per cent of radioactivity applied)

Injury	Radio-nuclide	t (min)	Means for decontamination	Body burden	Radioactivity decontaminated area	Decontamination efficiency
A	^{85}Sr	10	Undecontaminated	21,60 ± 6,09	-	-
			Saline	16,67 ± 4,04	24,94 ± 9,06	58,39 ± 9,06
			1% Cetavlon	16,79 ± 6,71	15,96 ± 9,06	57,75 ± 9,06
		30	Undecontaminated	62,18 ± 8,71	-	-
			Saline	58,78 ± 8,19	25,01 ± 6,40	16,21 ± 8,19
			1% Cetavlon	57,88 ± 10,40	15,21 ± 9,55	26,91 ± 10,40
	^{131}I	10	Undecontaminated	28,40 ± 7,42	-	-
			Saline	40,82 ± 11,14	14,13 ± 3,71	45,05 ± 11,14
			0,5% iodine tincture	26,92 ± 6,35	14,92 ± 3,68	58,16 ± 6,35
		30	Undecontaminated	74,92 ± 10,15	-	-
			Saline	82,86 ± 3,87	11,00 ± 1,91	6,14 ± 3,87
			0,5% iodine tincture	74,80 ± 4,82	8,81 ± 1,89	16,39 ± 4,82
B	^{85}Sr	10	Undecontaminated	0,01 ± 0,005	-	-
			Saline	0,03 ± 0,02	51,11 ± 7,56	48,86 ± 7,56
			1°/oo Soap	0,08 ± 0,07	59,69 ± 9,71	40,23 ± 9,71
		30	Undecontaminated	0,18 ± 0,12	-	-
			Saline	0,22 ± 0,07	85,07 ± 7,35	14,71 ± 7,35
			1°/oo Soap	0,24 ± 0,14	80,57 ± 7,96	19,19 ± 7,96
	^{131}I	10	Undecontaminated	1,62 ± 0,68	-	-
			Saline	5,92 ± 2,13	62,27 ± 10,88	31,81 ± 10,88
			1°/oo Soap	4,93 ± 1,65	51,52 ± 5,37	43,55 ± 5,37
		30	Undecontaminated	18,52 ± 6,65	-	-
			Saline	27,97 ± 5,99	55,45 ± 4,65	16,58 ± 5,99
			1°/oo Soap	32,98 ± 5,58	42,50 ± 6,93	24,52 ± 6,93

A - Excised skin wound
B - Thermal burn of the skin

Table I. In conditions of contamination of the excised skin wound and thermal burn of the skin, the penetrated quantities into organism depended on sort of contaminant, duration time of contamination i.e. time of beginning treatment and sorts of the applied means for decontamination (1).

Human data in the frame of the observed changes in the peripheral hlood (deviations and deseases) of the supervised persons in dependence of working exposition is given in Table II.

TABLE II - HAEMATOLOGICAL CHANGES AT SUPERVISED PERSONS

Working exposure (years)	Numerical	Morphological	Anaemia
1 - 10			1
10 - 20	2		2
over 20	5	3	1

The problems of differential diagnosis have been also contemplated according to medical results in relation to all the states and diseases that could give the similar deviations and changes together with importance of effect of the other harmful factors, influence of tabacco, alcohol, medicaments etc.(2).

CONCLUSIONS

At decontamination of the excised skin wound contaminated with different single radionuclide, the effect of decontamination conditioned more by body burden of the penetrated radionuclide, but at thermal burn of the skin preponderant the effect depending of level residual radioactivity in decontaminated region.

In the case of contamination excised skin wound and thermal butn of the skin with radiostrontium the efficiency of decontamination was to 70%, until with radioiodine did not exceed 60% from the initial contamination level.

Human examination showed that the majority of data falls into the normal limits, on the basis of which is possible to make the evaluation of the state of the health of radiation workers.

An early discovery of changes in some radiosensitive systems, observation of the evolution of changes and influence of factors which favourised the progression of some changes and the problems specifity and characteristics of some states and diseases, increasing the importance of differential diagnosis in pathology of irradiation.

*

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REFERENCES

1. Milivojević, K., Stojanović, D., Effects of Decontamination Treatment of Normal Skin and Excoriation Contaminated with Radionuclides, XI.Regional Congress of IRPA, Recent Developments and New Trends in Radiation Protection, Vienna, September 1983.

2. Stojanović, D., Milivojević, K., and col., Evaluation of Pathological Effects at Occupationally Exposed Persons to Radioiodine, IAEA-SM-266/70, Venice, April 1983.

ASPECTS MEDICAUX DES ACCIDENTS RADIOLOGIQUES TRAITES EN FRANCE

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INTRODUCTION

L'expérience française en matière d'accidents radiologiques remonte à 1956 et plus de 200 personnes ont été suivies ; la plupart d'entre elles pour des irradiations localisées, dont 60 s'accompagnant de radiolésions graves ; 14 personnes ont été soignées pour des irradiations globales à doses élevées. Ces deux types d'irradiation diffèrent par leurs caractères diagnostique, clinique, pronostique et thérapeutique. Toutefois, diagnostic et pronostic sont établis sur des données, en particulier dosimétriques, pour lesquelles la démarche est la même.

Dans les conditions idéales, la dosimétrie physique permet de connaître la topographie des lésions, l'importance de la dose délivrée aux tissus, tant en surface qu'en profondeur et donc l'évaluation de la dose absorbée par des organes ou tissus comme la moelle osseuse, le système nerveux, le tube digestif, les gonades, les veines et artères. Trois types de données permettent de l'établir : les caractères physiques de la source, en règle connue ; la géométrie source/tissus exposés et la durée de l'exposition. Ces deux derniers paramètres sont souvent imprécis et conduisent à effectuer une reconstitution de l'accident à l'aide de moulages des zones irradiées ou de fantôme en plastique "équivalent-tissu". Différents scénarii sont proposés, permettant de construire différentes courbes isodoses. La dosimétrie physique ne donne généralement pas de résultats absolus mais ceux-ci, présentés en terme de débit de dose, peuvent être ajustés peu à peu grâce aux données cliniques et biocliniques. D'autre part, la cartographie dosimétrique permet de connaître le gradient d'irradiation et donc de disposer d'une dosimétrie relative et comparative.

IRRADIATIONS LOCALISEES

Il s'agit pour la plupart d'atteintes des mains et des doigts. Le risque de séquelles trophiques et fonctionnelles est donc la préoccupation majeure. Leurs conséquences sont essentiellement fonction du caractère du rayonnement ; toutes les irradiations à fortes doses par ^{60}Co de 1,17-1,33 MeV ont été suivies d'amputations ; à l'inverse l'exposition aux rayonnements de basse énergie (moins de 10 keV), même à fort débit de dose, entraîne des nécroses réparables chirurgicalement ou même spontanément.

La reconstitution de l'accident est donc d'importance capitale. De même, l'apparition des premiers symptômes fonctionnels et physiques permettent de classer l'irradiation suivant sa gravité : la lésion radiologique est la brûlure qui possède des caractères particuliers :

- l'atteinte est fonction de la pénétration du rayonnement ;
- elle est proportionnelle à la dose délivrée et se traduit par : érythème, oedème, phlyctènes, ulcérations, sclérose, nécrose, troubles trophiques ;
- il existe des seuils. Selon la dose reçue par la menbrane basale, il apparaîtra
  . un érythème entre 3 et 8 Gy

. une épidermite sèche au-dessus de 5 Gy
. une épidermite exsudative entre 12 et 20 Gy
. une nécrose au-dessus de 25 Gy ;

- contrairement aux brûlures thermiques ou électriques, il existe un temps de latence et les brûlures radiologiques évoluent selon un schéma relativement stéréotypé : la forme limitée à l'épithélite exsudative comporte un érythème précoce, une phase de latence de trois semaines et un érythème secondaire suivi immédiatement de l'épithélite exsudative, la restauration s'effectue en 3 à 6 mois avec ou sans séquelles trophiques ; la forme avec endothélite vasculaire toujours associée à la précédente est caractérisée par une rechute 6 à 18 mois après l'incident ;
- la douleur est toujours très intense, posant d'importants problèmes thérapeutiques car elle résiste le plus souvent aux antalgiques majeurs ;
- un autre signe fonctionnel important est la sensation de chaleur immédiatement après l'accident. Cette impression est d'autant plus importante que le débit de dose est élevé.

Les explorations complémentaires effectuées le plus précocement après l'accident complètent les évaluations basées sur la dosimétrie et la clinique. Il s'agit essentiellement :

- de la téléthermographie qui fournit des courbes isothermes. Elle permet, avant l'apparition des signes cliniques, de vérifier la véracité de l'exposition : on constate une déformation des courbes isothermes et l'apparition de zones hyperthermiques (2 à 10 degrés de plus qu'une zone comparable non irradiée). Les zones les plus hyperthermiques correspondent aux zones les plus irradiées ; il n'est pas encore possible d'établir une corrélation avec la dose reçue et le rendement en profondeur. Lors de la réparation, il y a habituellement une stabilisation, voire une normalisation des images thermographiques. Toute persistance ou réapparition de point chaud fait craindre une reprise évolutive ;
- la scintigraphie vasculaire au ^{99m}Te visualise les débits vasculaires. L'irradiation s'accompagne dans les heures qui suivent l'accident d'une hypervascularisation des zones irradiées. Cet examen est utilisé également pour la surveillance ultérieure ; dans le cas de séquelles, existe une hypovascularisation des territoires atteints ; au moment des poussées évolutives, une augmentation de débit réapparaît mais qui possède des caractères cinétiques différents de celle de la phase aiguë.

Le principe du traitement est celui des brûlures, avec en particulier un maintien de l'asepsie des zones irradiées. Les lésions superficielles peuvent se réparer spontanément ; la cicatrisation peut être accélérée par l'utilisation d'enzymes comme la Super Oxyde Dismustase. Le risque de séquelles trophiques et fonctionnelles est important et il faut éviter l'immobilisation en position vicieuse et faire pratiquer une kinésithérapie active. Si les lésions sont plus profondes avec des nécroses étendues, ne permettant pas la réparation, le recours à la chirurgie est obligatoire : chirurgie réparatrice si possible ou amputation si nécessaire.

IRRADIATIONS GENERALISEES

Les irradiations mettant en jeu le pronostic vital sont exceptionnelles. La DL 50/60 est évaluée à 3,5 Gy (dose médullaire moyenne). Ces irradiations homogènes à dose élevée sont de deux types : soit irradiation délivrée en un temps très court (8 cas dans notre expérience), soit irradiation prolongée de plusieurs semaines (6 cas).

L'irradiation aiguë globale évolue cliniquement en 4 phases :

- phase de choc initial, avec nausées, vomissements, hyperthermie transitoire ;
- phase de latence, d'autant plus courte que la dose est élevée, de quelques heures à plusieurs semaines ;
- phase critique avec asthénie intense, prostration, obnubilation, céphalées, hyperthermie ;
- phase de rémission si la guérison suit. L'issue fatale peut survenir durant cette phase pour des doses supralétales en cas d'échec thérapeutique.

La symptomatologie sera d'autant plus riche que la dose reçue sera élevée.

Les données biologiques précoces sont essentiellement hématologiques : le pic granulocytaire très précoce et la pente de la chute des lymphocytes dans les 72 premières heures sont directement corrélés avec la dose. La chute ultérieure de toutes les lignées est la règle posant les problèmes de toutes les aplasies : infectieux, hémorragiques, hypoxiques.

Les prélèvements médullaires confirment l'aplasie ; les cultures de progéniteurs en particulier granulo-monocytaires permettent d'apprécier les possibilités de restauration spontanée de la zone ponctionnée.

La fréquence des altérations chromosomiques des lymphocytes circulants (dicentriques et fragments) est directement corrélée avec la dose moyenne absorbée. Elle peut être évaluée avec précision au-delà de 1 Gy.

L'étude électroencéphalographique complète les données hématologiques. Des altérations des ondes de vigilance apparaissent dès que la dose dépasse 0,3-0,4 Gy et sont en rapport avec la dose reçue.

Lors d'irradiations globales prolongées, le diagnostic est porté tardivement et les données dosimétriques physiques et biologiques sont incertaines. L'étude électroencéphalographique est l'examen qui permet sans doute la meilleure évaluation dosimétrique. La phase critique est caractérisée par sa longueur et par la lenteur de la restauration hématologique.

Le traitement est celui des aplasies médullaires quelle que soit leur étiologie. Il faut profiter de la phase de latence pour éradiquer tous les foyers infectieux et maintenir le malade dans le meilleur état d'asepsie possible. Lors de la phase critique la surveillance bactériologique est d'importance majeure. Dans tous les cas, il faut essayer d'obtenir une restauration spontanée, mais des traitements de compensation sont souvent nécessaires. L'importance du nombre de transfusions ainsi que l'éventualité d'une greffe de moelle osseuse impliquent qu'un typage précis du sujet soit effectué pendant la période de latence ainsi que ceux de donneurs éventuels.

Ces problèmes sont particulièrement intenses dans le cas des irradiations prolongées, avec une phase critique qui s'éternise ; plusieurs centaines de transfusions ont été nécessaires pour les 4 patientes traitées en France ; de même les complications infectieuses sont à redouter, en particulier septicémie à point de départ digestif ou mycose généralisée.

LES TECHNIQUES DE LA CAPILLAROSCOPIE DANS LE BUT DE LA PREVENTION DES SUJETS EXPOSES AUX RADIATIONS IONISANTES

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On sait que les radiations ionisantes provoquent des dommages parfois incertains chez les sujets exposés professionnellement. En effet il n'existe pas spécificité certaine des lésions dues aux rayons X et parfois l'évaluation en est aussi contradictoire.

Parmi les autres manifestations dues aux radiations ionisantes, celles-là induites par la microcirculation ne sont pas secondaires. Si on juge que de nombreuses situations de radiopathologie peuvent être la conséquence directe ou indirecte de l'action des rayons X sur les microvaisseaux (vieillessement précoce, radiodermites etc.).

Les lésions dues aux radiations ionisantes peuvent rentrer dans les processus inflammatoires, et la microcirculation des tissus est différemment interessée en les divers composants (libération de substances comme amines sympatique-mimétiques , adhésivité des leucocytes aux parois vasculaires , agrégation des plaquettes et des hématies , intervention des prostaglandines, libération des médiateurs chimiques locaux).

Maintenant on n'entend plus la microcirculation seulement comme l'ensemble des plus petits vaisseaux , mais comme complexe d'unités micro-vasculotissulaires , dans lesquelles, conjointement avec les structures microvasculaires proprement dites, il faut considérer les structures histangéiques du connectif environnant (substance fondamentale , fibroblastes , etc.). On peut donc interpreter de différente façon , l'action des radiations ionisantes sur la microcirculation. Et en effet , sur la peau, comme dans les autres endroits de l'organisme , les rayonnements ionisants engagent non seulement les capillaires, les artérioles, les veinules et les anastomoses artériolo-veineuses , mais aussi le connectif (substance fondamentale, fibroblastes, etc.

Tandis que le coeur et les grandes artères et les veines paraissent radiorésistentes , les capillaires et les petits vaisseaux résultent radiosensibles.

Des doses modérés de radiations déjà peuvent occlure les capillaires à travers l'hypertrophie ou l'hyperplasie des cellules endothéliales ; et l'occlusion des capillaires empêche le ravitaillement sanguin non seulement aux tissus voisins mais aussi aux territoires distaux. Et les effets immédiats dépendent du nombre des cellules détruites ; les effets tardifs, de leur côté, sont influencés par l'ischémie et la fibrose produites dans les tissus irradiés.

Même si négligés par quelques auteurs , les examens de capillaroscopie sont de grande utilité dans l'éclaircissement des altérations précoces induites par les radiations ionisantes sur les microvaisseaux des sujets exposés. Et dans ce domaine , l'étude de la microcirculation peut être d'un remarquable intérêt préventif en utilisant les techniques de la capillaroscopie qui permet l'observation " in vivo " des microvaisseaux, au niveau des divers endroits des téguments et des muqueses.

La technique de la capillaroscopie est simple : un microscope ordinaire avec éclairage oblique peut suffire ; un capillaroscope est cependant préférable. Actuellement on utilise communement des stéréomicroscope qui donnent une vision tridimensionnelle du réseau microvasculaire. A l'état normal , au lit de l'ongle , où la capillaroscopie est communement exécutée , les capillaires forment des boucles en "épingles à cheveux" paralleles ou obliques au plan de vision, disposées sur 3 à 4 couches , au-déssus du réseau veineux sous-papillaire. Il y a environ une vingtaine de boucles par $mm.^2$; la toile de fond est rose et transparente, la branche afférente du capillaire est plus fine de diamètre que l'efférente. Dans l'ensemble les boucles sont isolées, rectilignes ou à peine ondulées , parfois ramifiées en "bois de cerf" ou en "candélabre".Elles ne sont pas visibles en permanence : phénomène normal dû au caractère discontinu de fonctionnement qui comporte une apparente vasomotricité. A la surface de la peau , aux membres et aux pommettes , le réseau des veinules reste bien visible ; les boucles capillaires sont plus nombreuses ici qu'au lit de l'ongle (60 à 70 par $mm.^2$ au lieu d'une vingtaine), et se présentent par leur sommet à l'intérieur de chaque quadrillage dermique. Enfin à la conjonctive oculaire l'examen "in vivo" des microvaisseaux avec la capillaroscopie a un particulier intérêt: ici on peut voir la totalité du réseau terminal à un grossissement relativement faible: artérioles, veinules, capillaires et même anastomoses artério-veineuses. Et au niveau de la conjonctive, à l'état normal , les microvaisseaux sont nets et bien repartis ,il n'y a pas de zone avasculaire , les artérioles sont étroites et rectilignes, les veinules plus larges et ondulées et les capillaires dessinent de fins rameaux.

Avec la capillaroscopie on peut suivre les différents stades de l'érythème provoqué par les radiations ionisantes: d'abord engagement et dilatation du réseau capillaire de réserve et puis augmentation du nombre des capillaires superficiels , distension du réseau sous-papillaire qui devient visible. Et après plusieurs années , en cas de doses très fortes on peut encore reconnaitre les lésions des capillaires, qui restent irrégulièrs, dilatés, allongés, à contours flous et à réaction lente.

Dans un but préventif on a effectué des recherches de capillaroscopie au lit de l'ongle, à la conjonctive bulbaire et aux autres endroits exposés , en utilisant un capillaroscope Leitz fourni de fibres optiques et d'appareillage pour la microphotographie. On a utilisé en outre les techniques de pléthysmographie pour l'évaluation globale du flux sanguin et la thermographie de contact à cristaux liquides microcapsulés de cholestérine qui peut donner des

indications sur le comportement de la circulation superficielle des membres. On a examiné 350 sujets professionnellement exposés aux radiations ionisantes (opérateurs saniteurs , radiologues, techniciens de radiologie , etc.).

On a classifié les résultats de la capillaroscopie en quatre groupes. Dans le premier groupe on a considéré les résultats de normalité ou avec modestes modifications non significatives de nombre, forme, calibre et distribution des microvaisseaux. Dans un second groupe on a classifié les altérations non spécifiques , c'est à dire des altérations provoquées par une pathologie commune comme diabète , artériosclérose etc., ou par des facteurs exogènes comme les microtraumatismes , l'onycophagie , etc. , qui n'ont pas rapport avec les rayonnements ionisants. Dans un troisième groupe on a indiqué les tableaux suspects , parmi lesquels on a relevé les conditions suivantes:

I)Capillaropathie hémorragique
2)Ectasies des capillaires
3)Augmentation du nombre des capillaires
4)Raréfaction des microvaisseaux.

Enfin dans un quatrième groupe on a classifié les cas de capillaroscopie avec des signes très suggestifs de dommage provoqué par les rayons X. En ce dernier groupe on a relevé deux tableaux différents qui pourraient être l'expression d'une diverse condition physiopathologique de la microcirculation envers les radiations ionisantes. Le premier de ces tableaux est représenté par une raréfaction extrême des capillaires qui sont évanescents, à contours flous , dans un contexte de pâleur de la toile de fond et de grande pauvreté de la perfusion sanguine . Le second type capillaroscopique de ce groupe est représenté par des aspects prédominants de néogenèse vasculaire. Dans ce type les boucles capillaires sont fréquemment dilatées, tortueuses , tortillées et avec des considérables altérations de forme,calibre et écoulement. Les capillaires résultent ici, fortement augmentés de nombre , et on peut voir aussi , le plexus veinulaire sous-papillaire et même fréquentes les micro-hémorragiea. Ces rapports peuvent être vérifiés à la conjonctive oculaire ,qui peut aussi témoigner - dans le premier cas - d'une pauvreté de microvaisseaux et d'irroration sanguine ; tandis que ,dans le deuxième cas on peut relever richesse en microvaisseaux,réseau maillé, dilatations capillaro-veinulaires avec des aspects de tortuosité et stase de la microcirculation remarquable.

Dans l'ensemble on peut dire que la plus grande fréquence d'observation de nos cas est relative aux premiers deux groupes dans lesquels on peut exclure le dommage provoqué par les radiations ionisantes. Toutefois on a relevé aussi des cas où la présence d'altérations suspectes ou suggestives de dommage radio-induit , n'était pas associé à radiolésions macroscopiquement visibles. Dans ces cas a été également utile le recours aux autres techniques parmi lesquelles la pléthysmographie , qui nous donne un tracé expression de la réponse globale du système vasculaire , tandis que la thermographie peut être utile pour l'évaluation de la circulation superficielle , spécialement

avec des tests de refroidissement.

La capillaroscopie reste , en définitive, la seule technique qui nous permet la visualisation des capillaires " in vivo " , et chez les sujets exposés aux radiations ionisantes , peut témoigner d'une souffrance assez typique des rayons X sur les structures microvasculaires , parfois initiale et reversible.

Il faut donc rechercher ces éléments de la capillaroscopie dans les sujets professionnellement exposés , pour les interventions de prévention et protection sanitaire , même s'il est nécessaire de différencier ces manifestations de celles qui n'ont pas rapport avec les radiations ionisantes et qui pourraient être préexistantes.

Bibliographie

Curri S.B.,Merlen J.F.-Histangiopathies ou Microangiopathies.Journ.des Sciences Médicales de Lille,93,255,I975.

Duncan W.,Nias A.H.W.Clinic Radiobiology,Churchill,Livingstone Edinburg,London, I977.

Haye C.,Jammet H.,Dollfus M.A.-L'oeil et les radiations ionisantes.Ed.Masson,Paris ,I965.

Lacassagne A.,Gricouroff G.-Action des radiations sur les tissus.Ed. Masson,Paris,I94I.

Merlen J.F.-Acrosyndromes vasculaires et capillaroscopie.Journ.Mal. Vascul.,Masson ,Paris,4,43,I979.

Pennarola R.,Rocco P.,Stanga A.-Alterazioni microcircolatorie in personale esposto alle radiazioni ionizzanti.Riv.Inf. Mal Profess.,66,227,I979.

Piovella C.-La capillaroscopia ed altre indagini strumentali.Min.Angiol.,3,I03, I978.

Rheinhold H.S.,Buisman G.H.-Radiosensitivity of capillary endothelium.Brit.Journ. of Radiol.,46,54,I973.

Strambi E.-Problèmes relatifs à l'évaluation de l'aptitude au travail comportant un risque d'irradiation.Euratom,EUR 5624 f,I976.

LONGTERM EVALUATION OF AN ACCIDENTAL PLUTONIUM INHALATION

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INTRODUCTION

In the course of routine urine analysis of an EIR worker, an alpha-activity of 0,3 pCi/l was found. The isotopes identified by alpha-spectrometry were Pu-239 or Pu-240 and Am-241 or Pu-238. Further analysis confirmed an unobserved incorporation of plutonium. The only acceptable explanation was an inhalation of aerosols 2 months earlier during welding on a plutonium contaminated glove box.

MEASUREMENTS AND METHODS

To assess the activity inhaled, low energy X-ray measurements of the lung were made with phoswich detectors and excreta were collected. The first lung measurement, made in April 1980, was not clearly interpretable. An additional incorporation of 50 nCi I-125 was detected, which disturbed severely the measurement in the plutonium X-ray region. The EIR-plutonium contains always known amounts of Am-241 which emits an easily detectable γ-radiation of 60 keV. After correction for the I-125 contribution in the 60 keV energy region there remained no significant higher counting rate compared with similar persons.
With a calculated detection limit of 1 nCi Am-241 in a measuring time of 40 minutes and a Pu-239/Am-241 activity ratio of 5 - 7 an upper limit for the lung activity of around 6 nCi Pu-239 at that time can be given. The activity ratio was estimated from the fuel handled in the box. Further lung measurements gave no other results.

URINE AND FECES ANALYSIS AND MEASUREMENTS

In the first year, one day every week, later on on three consecutive days every month and finally on five concecutive days every three months all excretions were collected and analysed. This programm was stopped in October 1983 because the detection limit for plutonium was reached.
From every urine sample plutonium and americium were extracted with cupferron. The samples for proportional counting and alpha-spectrometry were prepared by evaporation.
The fecal samples were ashed at 600°C. Aliquots of 0,5 g or 1 g were analysed by the cupferron-method.
Additional uranium analyses from feces were made using the aliquat-method. The activity was constant (6±3) pCi/24h.

DISCUSSION OF THE EXCRETION PATTERN

The results from the excretion measurements show large variations in the amount of alpha-activity. Statistical evaluations of the data in Fig. 1, Fig. 2 and Fig. 3 were made using linear regression methods.

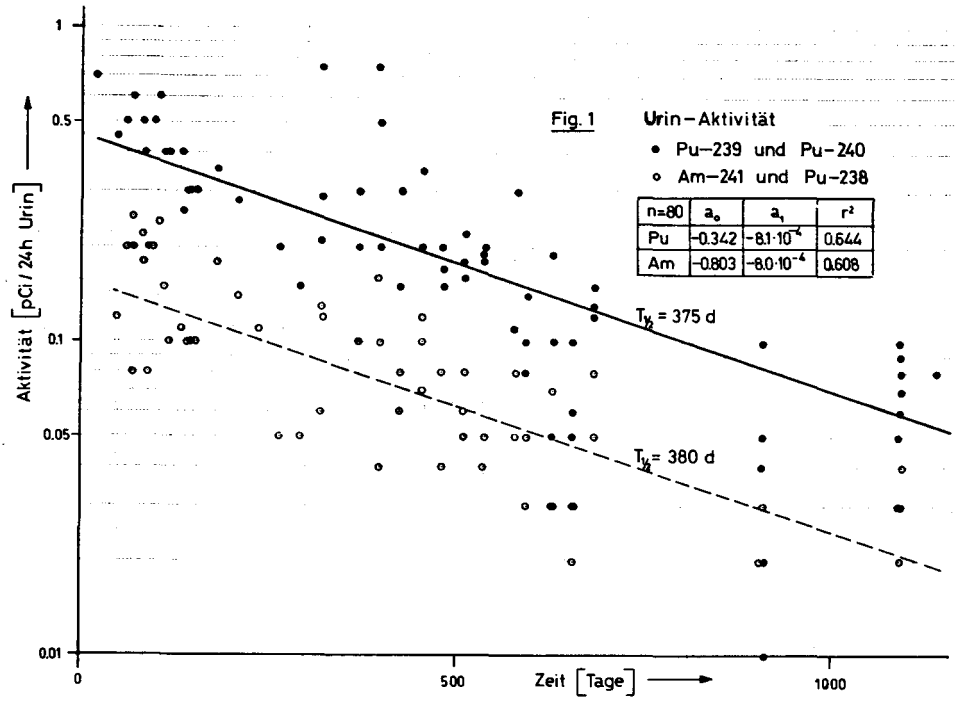

Fig. 1: Daily urinary excretion

The decrease of plutonium and americium activity in urine and feces has a half live of 370 days and 310 days, respectively. There are constant but different Pu/Am activity ratios in urine and feces. In urine the ratio is 3 and in feces it is around 5. The higher solubility of americium may explain this effect.

As compared with other publications the wide variation of the daily activity excretion is normal. One reason for this effect may be in our case a strongly changing Po-210 activity up to 0,3 pCi/day in urine and up to 3 pCi/day in feces. For low level plutonium analysis the Po-210 must be removed.

Nevertheless, we had expected a better homogeneity of the urine values, because the person under investigation had a steady drinking pattern. The daily drinking volume is 1 l mineral water, 1 l tea and some cups of coffee.

DOSE CALCULATIONS FROM EXCRETION DATA

The urine and feces results were interpreted using the ICRP 30 lung model for class Y plutonium, but with the experimentally determined biological halflife, a particle size of 1 µm AMAD, a urine volume of 1,8 l/d and a fecal ash weight of 9 g/d.
Using these parameters, we found a dose difference about a factor of

10 in interpreting the urine and feces results. This discrepancy led to the engagement of specialists from other institutions to calculate the dose from the excretion data [1][2][3].
In Tab. 1 all results are compared. In general, our findings were confirmed.

Tab. 1 Intake and committed effektive dose equivalent calculated from different institutions

INSTITUTION	eff. $t_{1/2}$ (d)	INTAKE (nCi) based on urine	INTAKE (nCi) based on feces	H_{50} (rem) from urine	H_{50} (rem) from feces
EIR-Switzerland	370	170	19	55	6
Chalk-River Canada	374	175	21	55	7
BGA-Germany	365	86	19	25	6
KFA-Germany	273	250	20	75	8
ICRP 30	500				

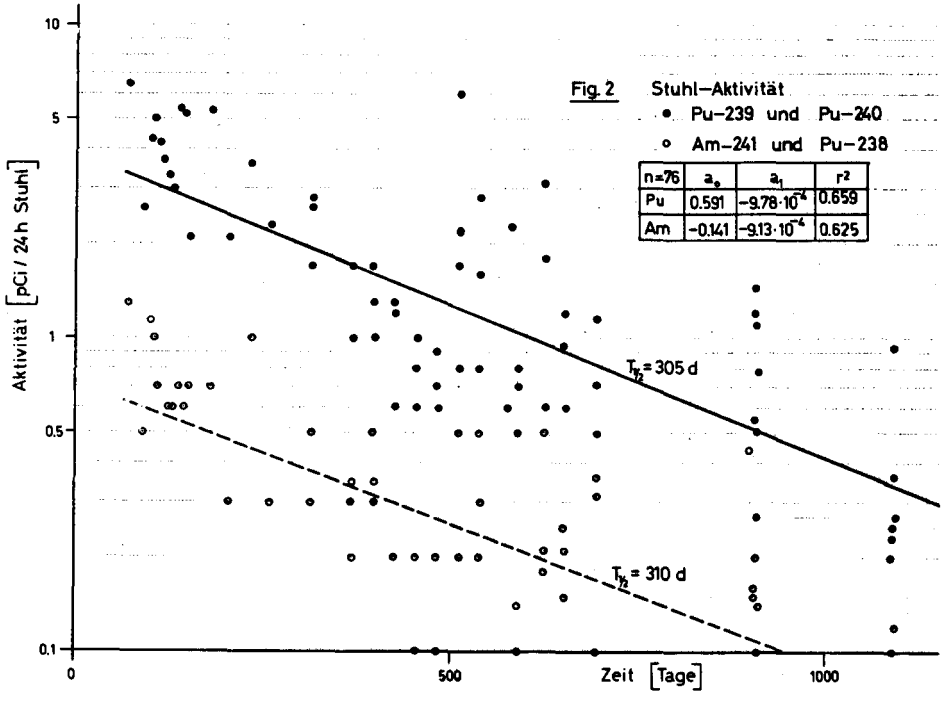

Fig. 2: Daily fecal excretion

Attempts made by J. Johnson [1] and K. Hendrichs [2] to find a dose agreement by varying single ICRP-parameters failed.

Urine samples of the worker were distributed for an international intercomparison for the determination of metabolized plutonium. It proved the absence of systematic measurement errors. The comment of all persons involved in the dose calculation was, that in this and other cases [4] the ICRP parameters are not directly applicable.

If we compare the excretion data and the lung measurements, there is no disagreement between fecal excretion and lung counting, but there is one between urinary excretion and lung counting.

Obviously the urine analysis overestimates the dose by a factor of nearly 10. This may produce serious problems with the control authorities. It is suggested that the ICRP 30 parameters of the lung model should be reviewed.

[1] J. Johnson, Chalk River Nuclear Laboratories, Canada
Private Communication
[2] K. Hendrichs, Bundesgesundheitsamt, Neuherberg, Germany
Private Communication
[3] D. Beyer, Kernforschungsanstalt Jülich, Germany, Private Com.
[4] G. L. Voelz, Health Phys., Vol. 29(4), 551-561, Oct. 75

ENVIRONMENTAL SURVEILLANCE : DO LESS AND LEARN MORE

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ABSTRACT

In this paper, we consider the principles that govern the design of environmental surveillance programmes for discharges from various installations under normal conditions of operation. Two opposing objectives bear on the situation: to estimate the doses to the public with acceptable uncertainty and to avoid expenditure on unnecessary or unstructured monitoring. We suggest that it is possible to classify discharges according to their potential for delivering doses to individuals and the community and then to decide on the scale of monitoring. This approach leads to an environmental surveillance programme shorn of surplus elements and to which additions on grounds other than radiological protection are made with awareness. Present surveillance practices not complying with such principles may need to be reformed.

INTRODUCTION

There has been a steady increase, over recent times, in the number of radiation sources that cause exposure of the public and a pronounced increase in public awareness of such exposures with their imputed risks. This enhanced perception finds expression in demands for reduced releases of radionuclides to the environment and generates pressures for increasingly comprehensive and sensitive environmental surveillance. The pressures have been intensified by the emphasis that the International Commission for Radiological Protection places on the ALARA approach(1) and the general assent given to that recommendation. We have noted elsewhere the apprehension(2) among environmental analysts that they will be required to achieve ever lower levels of detection.

These pressures, whether from the public or within the profession of radiation protection, should be resisted. To yield to them would be to misallocate human and material resources with too much effort being expended on unimportant pathways that lead to trivial doses. The pitfall could be avoided if environmental monitoring programmes were designed with clear objectives and on logical lines. Reviews of such programmes in the USA(3,4) have indeed revealed that the elements requiring most attention are rationale and design. In this paper, therefore, we shall concentrate on objectives and design and offer some broad suggestions. We are not speaking for the Board, but exposing some personal proposals.

FUNCTION OF MONITORING

When installations discharge radioactive waste to the environment, it is generally found that the dose to the public is dominated by a few radionuclides in a few media. In general, levels of radioactive contamination are difficult to measure outside the immediate surroundings of installations. Environmental monitoring tends therefore to be concentrated on proximal environments and on critical groups. Measurements may be made further afield, but only to determine so-called background levels. The collective dose from discharges of radionuclides is usually estimated from measurements or calculations of the discharges and mathematical models of environmental transport and transfer.

If we restrict our reasoning to protection, we are forced to the conclusion that the primary purposes of normal environmental monitoring are to

assess doses in the community and to demonstrate compliance with authorised limits. The order of these objectives here deliberately reflects the shift of emphasis from limitation to optimisation. A monitoring programme would be justified only if it could fulfil these objectives and naturally it would be expected to fulfil them. Monitoring programmes may have many secondary objectives, and international organisations have described them(5,6). We shall touch on some of these later, but it is well to dispose of one at this juncture. The need to assure the public of the continuing safety of the installation is seen by some of our colleagues as a prime objective in its own right, and they imply that monitoring programmes should be designed with this in mind. We disagree. Assurance should flow from a programme designed for protection purposes and not vice versa.

CLASSIFICATION OF INSTALLATIONS

There are about twenty nuclear installations and over a thousand other installations in the United Kingdom that routinely produce radioactive waste. Many discharge radioactive material, but few undertake routine environmental monitoring, nor indeed should they. The government departments that authorise waste disposal consider each case on its merits and prescribe or effect the monitoring. As in other countries, the resulting programmes are inspired by wisdom, understanding, counsel and other professional gifts. Previous practice is also an important influence. There is, we know, no substitute for quality of judgment in these matters, but we should like to quantify part, at least, of the process.

We begin with an attempt to classify installations according to the individual and collective doses that they cause in the community. The classification is solely related to routine environmental monitoring: the annual dose refers to the mean effective dose equivalent in the critical group, and the collective dose refers to the collective effective dose equivalent commitment from one year of operation. The doses will have been inferred with varying degress of uncertainty from previous measurements and modelling.

Our colleagues are proposing at this congress(7) that there must be some levels of dose and risk that should be of no concern to regulating authorities. As lawyers manqué, they call them de minimis levels, but we know that they mean negligible. The doses they suggest are 1 μSv a year to individuals and 1 manSv collectively, the collective dose referring to the impact of the extended operation. We adapt their numbers to our purpose, and assert that normal environmental monitoring is never required for installations that do not cause doses in excess of 1 μSv a year to individuals and 0.1 manSv collectively from a year of operation.

We then proceed to delimit a minor class in which installations cause doses above these negligible values but below 10 μSv a year to individuals and below 1 manSv a year collectively. Many non-nuclear installations are in this class - hospitals, universities, factories. At the other extreme, we define major installations that cause annual doses above 100 μSv a year or collective doses above 10 manSv: many of the installations associated with the nuclear fuel cycle would be in this class. We have, by exclusion, identified a medium class, which will have intermediate requirements for normal environmental monitoring.

At this point, the reader may argue that the classification is deficient because it takes no account of the potential need for environmental monitoring in an emergency. We recognise that there may have to be a plan for emergency

monitoring at an installation and that the complexity of a plan should be commensurate with the potential magnitude of the emergency, but we would counter that the objectives of emergency monitoring are quite different from the objectives of normal monitoring and consequently that their designs are quite independent: in blunt terms, normal and emergency schemes should not be confused. But if it so happens that the designs overlap and that some normal monitoring may be regarded as preparation for emergency monitoring, the opportunity should be seized to reduce the cost of emergency preparedness. Although we are not dealing with emergency monitoring on this occasion, it seems to us that a classification related to potential dose to individuals and potential collective dose might also be feasible.

SCALE OF MONITORING

The next step is to devise a monitoring scheme that matches the classification. In doing so, we shall refer to source monitoring, by which we mean an appropriate procedure at an installation to determine the discharge of radioactive material to the environment or the emission of radiation in any other manner. We shall also refer to a scale of frequency with three bands: continual, which implies measurements in fairly close succession; periodic, which means regular but not frequent; occasional, which suggests annual measurements at most. These frequencies will inevitably be affected by the practicability of collection and measurement, the half-life of the radionuclides of interest, and temporal variability such as seasonal changes.

Starting with individual minor installations, we suggest that they are unlikely to require routine environmental monitoring to satisfy the objectives we have laid down: doses can be assessed with adequate certainty from an inventory of the radioactive sources, from a knowledge of the disposal procedures, and from a simple model of exposure in the environment. They are most unlikely to have source monitoring. For the class as a whole, however, it would seem worthwhile for the authorising agencies to monitor the overall impact in the environment occasionally. This environmental monitoring would, in effect, be an economical national scheme.

For major installations, we suggest that continual monitoring is required in the environment for any radionuclide that contributes more than 25% to the critical or collective doses and that periodic monitoring should be considered for contributions exceeding 10% to either. It is likely that installations in this class will carry out source monitoring, a most important function of which is to warn of abnormal releases.

In the intermediate class, we propose that periodic measurements be made in the environment for radionuclides contributing more than 10 μSv to the critical group dose or 1 manSv to the collective dose, and it would be logical to make occasional measurements for other contributors in aggregate. Source monitoring may be encountered in this class.

If there is appreciable uncertainty in an environmental model and the estimated doses are near the boundary of a class, it would be appropriate to verify the estimates experimentally: this advice applies to all classes of installation.

SHIFT OF EMPHASIS

Our prescription is a deliberate attempt to cure our friends of an obsession with the monitoring of critical or near-critical groups. We know that the possibility of legal sanctions, no matter how remote, is a powerful incentive in this matter, but too much attention to limits is a sure sign of

reaction. On a more serious note, one could point to the fact that the doses that delineate our categories imply far greater impact through collective exposure than through critical exposure. To lend extra force to the argument, it is only necessary to remind them that any decision to reduce discharges from an installation would be heavily influenced by the expected saving on collective dose.

What we are advocating therefore is more monitoring for collective dose at the expense of individual dose. Were we pressed to say what the proportion might be, we would have to admit that it is virtually impossible to generalise, but that one should expend effort on each type according to the coordinates of the installation within our classification scheme. Should routine environmental monitoring for collective dose prove impracticable in any circumstance, we would not be in favour of restoring the original lopsided position. The probable consequence would be a reduction in the amount of normal environmental monitoring practised in the United Kingdom. The same would occur, we believe, in other advanced countries.

There are inadequacies in present practices that we would seek to eliminate with the resources thus made available. We should be inclined, first of all, to improve quality assurance to the degree that it is at least acceptable: at present, we pay mere lip-service to it. It would also be desirable to devote some of the savings to a systematic review of the monitoring programme each year. For these tasks, we suggest allocating an amount equal to one tenth each of the restructured budget for routine environmental monitoring. There may even be resources left over for informative special investigations.

We realise that we have not been able to deal, in this short space, with some important but subsidiary problems relating to the design of monitoring programmes - the disposition of sampling locations around an installation, for instance. We intend to turn to these matters in another place and in the light of the reception given to these proposals.

REFERENCES

1. International Commission on Radiological Protection, Publication 26, Annals of the ICRP, 1, 2, (1977).

2. Fry, F.A. and O'Riordan, M.C., Low-level environmental measurements in radiological protection. Presented at ICRM seminar on Alpha Particle Spectrometry and Low-level Measurement, May 1983, to be published in Nucl. Inst. Meth.

3. Waite, D.A. and Denham, D.H, Identification of potential improvements in environmental surveillance techniques, Nucl. Saf. 17, 344, (1976).

4. Denham, D.H., Environmental radiological surveillance in perspective: the relative importance of environmental media as a function of effluent pathway and radionuclides, Hlth. Phys. 26, 273, (1979).

5. International Commission on Radiological Protection, Publication 7, Principles of Environmental Monitoring related to the Handling of Radioactive Materials, (1965).

6. International Atomic Energy Agency, Safety Series No. 41, Objectives and Design of Environmental Monitoring Programmes for Radioactive Contaminants, (1975).

7. Fleishman, A. and Clarke, R.H., these proceedings.

MEASUREMENT OF SPECIFIC RADIONUCLIDES IN GASEOUS EFFLUENTS FROM NUCLEAR POWER PLANTS AND THEIR CONTRIBUTION TO RADIATION EXPOSURE

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INTRODUCTION

Requirements for the surveillance of radionuclides in gaseous and liquid effluents from nuclear power plants are outlined in regulatory guidelines. In addition to these nuclides which are measured by the operators there exist in the effluents from nuclear power plants other radionuclides which require special measuring techniques due to their mode of decay.

This paper describes methods for sampling and measuring P-32, S-35, Fe-55, Ni-63 and Tc-99 which are not measured on a routine basis. Results of measurements of samples taken at the stack of nuclear power plants in the Federal Republic of Germany are given. Also presented are measurement results of C-14 in the gaseous effluents from nuclear power plants and research reactors. An assessment is made of the radiation exposure to the public resulting from the emissions of these radionuclides.

SAMPLING AND MEASURING OF SPECIFIC RADIONUCLIDES

C-14 may be contained in the effluent air from nuclear power plants in the form of carbon monoxide, carbon dioxide, methane and other hydrocarbon compounds. Since in most cases, direct measurement of the beta emitter C-14 in the gaseous effluents of a nuclear power plant will hardly be possible because of the high activity of radioactive noble gases, continuous sampling and discrete measurements in the laboratory have to be relied on. Molecular sieves can be used for this purpose. This procedure also permits the continuous sampling of gaseous compounds of phosphorus and sulphur for determining the radionuclides P-32 and S-35. After sampling, these three radionuclides are eluted and precipitated in order to prepare measurement samples. The activity is determined using a liquid scintillation spectrometer.

Since 1982 the Federal Health Office has been conducting measurements of technetium emissions in effluent air from nuclear power plants. For the purpose of selecting a suitable measurement technique for the detection of this nuclide in liquid and gaseous forms and as a particulate, a series of separation methods was first examined.

Sampling of Fe-55 and Ni-63 in the effluent air from nuclear power plants is done by continuous separation of particulates on filters in a bypass of the stack. After ashing the filter samples, iron and nickel are precipitated as carbonates. The iron is extracted as a chlorine complex and nickel is precipitated with dimethylglyoxim from the remaining solution. The samples are measured by liquid scintillation spectrometry.

RESULTS AND DISCUSSION

From 1978 to 1982 carbon dioxide-bound C-14 emissions from nuclear power plants have been measured by the Federal Health Office. Additional measurements of C-14 as carbon monoxide or organically bound C-14 were performed. Table 1 shows the results of C-14 measurements for the years 1978 to 1982. Where as in boiling water reactors nearly the entire C-14 activity is emitted as carbon dioxide, pressurized water reactors emit only approx. 5 - 50 % as carbon dioxide. With this data collected from 1978 to 1982 a yearly mean total C-14 emission of 200 GBq/GWe can be calculated for a nuclear power plant with a pressurized water reactor and 500 GBq/GWe for a plant with a boiling water reactor. In addition, during the years 1980 to 1982 C-14 samples were taken from research reactors for several weeks, respectively. Assuming a constant power, the yearly emissions from research reactors were estimated and compiled in table 2.

Since the beginning of 1983 the emissions of the pure beta-emitters P-32 and S-35 in the gaseous effluents have been determined. Table 3 contains the results of P-32 and S-35 measurements up to date. Results of technetium measurements can be tentatively expected by 1984. From 1979 to 1982 the emissions of Fe-55 and Ni-63 were determined in the gaseous effluents from nuclear power plants. The results from these measurements are compiled in table 4 and 5.

RADIATION EXPOSURE

The radiation exposure was calculated on the basis of averaged annual releases of C-14, Fe-55 and Ni-63 with gaseous effluents. A similar calculation was performed for P-32 and S-32 using the data from table 3. The results are given in table 6 and are below 1 µSv/a.

Table 1: Annual releases of C-14 with gaseous effluents from nuclear power plants

Nuclear power plant	Annual releases in GBq/a* 1978	1979	1980	1981	1982
Brunsbüttel	170	7.4	30	240	59
Gundremmingen	3.7	3.7	0.74	0.26	-
Isar	-	-	-	180	4.8
Würgassen	230	160	350	200	130
Philippsburg	-	-	-	6.3	93
Biblis A	22 (56)	-	-	-	52(110)
Biblis B	15 (15)	-	-	-	22
Neckarwestheim	7.4(140)	22(160)	11(96)	22(110)	44
Obrigheim	19 (15)	74(33)	22	37	26
Stade	44	56(70)	41(78)	210	78
Unterweser	-	19(230)	56	33	41

* The results from measurements of carbon monoxide- and hydrocarbon-bound C-14 are listed in brackets

Table 2: Extrapolated annual releases of C-14 from research reactors

Research reactor	Type	Power in MW	Annual releases of C-14 in GBq/a*
Neuherberg (FRN)	LWR	0.14	0.081
Karlsruhe (MZFR)	HWR	200	1400(1400)
Karlsruhe (FR2)	HWR	44	78(48)
Jülich (AVR)	GCR	46	41(22)
München (FRM)	LWR	4.15	- (1,6)

* Values of C-14 as carbondioxide are listed in brackets

Table 3: Releases of S-35 and P-32 with gaseous effluents during the first half of 1983

Nuclear power plant	Activity releases (MBq) S-35		P-32	
	I. Quarter	II. Quarter	I. Quarter	II. Quarter
Brunsbüttel	<100	<100	<100	<100
Stade	1000	750	<100	<100
Unterweser	230	320	<100	<100
Würgassen	<100	<100	<100	<100

Table 4: Annual releases of Ni-63 with gaseous effluents from nuclear power plants

Nuclear power plant	Annual releases (MBq/a) 1979	1980	1981	1982
Brunsbüttel	22 (3)	1.2 (3)	0.06 (1)	0.15
Gundremmingen	4.2	0.11 (3)	0.053 (2)	0.15
Isar	1.0 (3)	0.37 (2)	0,38 (2)	6.68
Kahl	0.014 (3)	0.083 (3)	0.001 (2)	0.01
Lingen	0.08	0.31 (2)	0.02 (3)	0.04
Philippsburg	0.05 (1)	0.21 (3)	0.39 (2)	0.71
Biblis A	0.04	0.066 (2)	0.01 (2)	0.03 (3)
Biblis B	0.04	0.02 (3)	0.008 (1)	0.04
Neckarwestheim	0.28 (3)	1.1 (2)	0.45 (3)	0.58
Obrigheim	11	11 (3)	-	13.6
Stade	1.1 (3)	1.3	0.066 (2)	0.09
Unterweser	0.46 (1)	0.46 (1)	0.32 (1)	0.04
Grafenrheinfeld	-	-	-	0.1

(1) Value of one quarter
(2) Value of two quarters
(3) Value of three quarters

Table 5: Annual releases of Fe-55 with gaseous effluents from nuclear power plants

Nuclear power plant	Annual releases (MBq/a) 1979	1980	1981	1982
Brunsbüttel	1200 (3)	13 (3)	28 (3)	44
Gundremmingen	34	2.5	3.7	
Isar	452	250	570	600
Kahl	0.13	0.27(2)	0.15(3)	0.01(2)
Lingen	1.6	8.0 (3)	8.6	2.0
Philippsburg	18	23	42	6.5 (3)
Biblis A	1.0	4.9	9.1	13
Biblis B	1.1	8.7 (3)	2.9 (3)	3.2
Neckarwestheim	8.8	12 (3)	81	15.8
Obrigheim	25	20	2.5 (1)	1.6 (1)
Stade	12	7.3	1.2 (2)	3.0
Unterweser	2.8	4.8 (3)	3.4 (3)	22
Grafenrheinfeld	-	-	-	9

(1) Value of one quarter
(2) Value of two quarters
(3) Value of three quarters

Table 6: Radiation exposure by inhalation and ingestion from the emission of 100 GBq/a C-14, 100 MBq/a P-32, 100 MBq/a S-35, 10 MBq/a Fe-55 and 1 MBq/a Ni-63 in gaseous effluents from nuclear power plants.
(a longterm dispersion factor of 1,7 10^{-7} is assumed)

Organ	Exposure (nSv/a) C-14	P-32	S-35	Fe-55	Ni-63
Total body	54	0,8	0,5	5E-3	3E-3
Bone	270	27	2,2	2E-2	0,2
Liver	54	1,4	-	3E-2	1E-2
Thyroid	54	-	-	-	-
Kedneys	54	-	-	-	-
Lungs	54	-	-	1E-2	-
G-I-Tract	54	14	1,9	2E-2	3E-3

UMGEBUNGSDOSIMETRIE DER SCHWEIZER KERNKRAFTWERKE WAS WOLLEN WIR MESSEN UND WIE INTERPRETIEREN WIR DIE MESSERGEBNISSE?

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1. EINFUEHRUNG

Eine kritische Bewertung der TLD-Messdaten der letzten 10 Jahre ergab eine Interpretationsmethode, die die Aussagekraft der Umgebungsdosimetrie wesentlich verbessert. Der Aufwand für das Routine-Umgebungsüberwachungsprogramm bleibt dabei in verhältnismässig kleinem Rahmen. Die damit erreichbare NWG genügt, um das Einhalten der behördlichen Limiten (Schutzziele) - insbesondere auch während des Normalbetriebes - zu dokumentieren.

2. AUFGABENSTELLUNG FUER DIE UMGEBUNGSDOSIMETRIE IM NORMALBETRIEB

Wenn wir über Umgebungsdosimetrie sprechen, handelt es sich hauptsächlich um die Messung der Strahlenbelastung durch die Edelgaswolke. Aerosolemissionen aus KKW sind erfahrungsgemäss vernachlässigbar klein. Auch die Wahrscheinlichkeit einer grösseren Aerosolemission ist im Normalbetrieb sehr gering. Dagegen kann eine kurzzeitige Edelgasemission im Bereich von ca. 1/10 der Jahreslimite, die im Durchschnitt im Bereich von ca. $3 \cdot 10^4$ bis 10^5 Ci/a liegen, als viel wahrscheinlicher betrachtet werden. In Abhängigkeit von den Ausbreitungsverhältnissen kann dabei eine Dosisbelastung im Bereich von 0.02 bis 0.1 mSv erwartet werden. Eine ähnliche Dosisbelastung ist bei einer gleichmässigen Ausschöpfung der Abgabelimiten über das ganze Jahr zu erwarten. Trotz der Tatsache, dass eine derartige Dosis von einigen Prozenten der natürlichen Jahresstrahlenbelastung, ein geringes Risiko für den Menschen darstellt, würden solche Aktivitätsemissionen durch die Oeffentlichkeit als beunruhigende Ereignisse bewertet werden. Ueber die dabei verursachten Strahlenbelastungen würden exakte Angaben gefordert werden. Die Aufgabe der Umgebungsdosimetrie ist somit, eine objektive, gut interpretierbare Aussage über die Auswirkung der Kernanlage während des Normalbetriebes auf die in unmittelbarer Nähe wohnenden Personen zu liefern. Dazu ist es erforderlich, den durch die Kernanlage bedingten Anteil der Dosisbelastung in der Umgebung messtechnisch von der Untergrunddosis zu trennen. Bei Unfallsituationen ist die Trennung der gemessenen Dosis in Untergrund- und Anlagedosis für die Aussage bezüglich Belastung in der Umgebung weniger relevant und wird hier nicht betrachtet.

3. INTERPRETATION DER MESSRESULTATE

3.1 Ortsdosis als Interpretationsobjekt

Im Rahmen der Strahlenschutzmesstechnik stellt die Trennung der Messergebnisse in "nützliche Signale" und "Untergrund" immer wieder eine der Grundaufgaben dar. Leider verfolgen die beiden für die Umgebungsdosimetrie massgebenden Dokumente ("Richtlinie zur Emissions- und Immissionsüberwachung kerntechnischer Anlagen" (1); "Technische Empfehlungen für Festkörperdosimetrie zur Umgebungsüberwachung" (2)) nicht diese Grundaufgabe. Vielmehr zielen sie auf die Bestimmung von Ortsdosen der Photonenstrahlen an einer Vielzahl von Orten hin. Die in (1) postulierte NWG von 0,4 bis 0,5 mSv pro Jahr bzw. die in (2) postulierte Gesamtunsicherheit im Bereich von $\pm$ 30 % sind eine Folge dieser Zielsetzung.

Die "Technischen Empfehlungen" haben sich im wesentlichen als Ziel gesetzt
- als Entscheidungshilfe bei der Auswahl bei Neuanschaffung von Dosimetersystemen und
- als praktische Hilfe bei der Durchführung von Umgebungsüberwachungsprogrammen zu dienen.

Was das Umgebungsüberwachungsprogramm betrifft, führen sie aber zu einer Verwissenschaftlichung der Routine-TLD-Messtechnik. Der eigentlichen Interpretation der Messdaten wird dagegen zu wenig Platz eingeräumt.

3.2 Nettodosis als Interpretationsobjekt

Im Gegensatz zu diesen beiden Dokumenten (1) und (2), in denen über Messungen der Ortsdosis als Ziel der Umgebungsdosimetrie die Rede ist, wird hier gezeigt, dass die Bestimmung der anlagebedingten Dosis (Nettodosis) die primäre Aufgabe ist (3, 4).

Der Ausgangspunkt unserer Betrachtungen beruht auf einer Verfolgung der strengen Korrelation aller Messwerte einer beliebigen Messreihe mit allen anderen, an denselben Orten durchgeführten, Messungen.

Wir räumen daher in unserer Interpretationsmethode der Bestimmung der ortsspezifischen Eigenschaften des Strahlenfeldes der einzelnen Messpunkte eine grosse Bedeutung ein. Dieses sehen wir auch als Hauptaufgabe der Umgebungsdosimetrie in der Beweissicherungsphase.

Wir interpretieren den Ortsdosismesswert als Bruttowert, der aus drei Komponenten zusammengesetzt ist. Er beinhaltet die eventuell durch die Anlage verursachte Nettodosis, eine orts- und eine umgebungsspezifische Komponente.

Der ortsspezifische Teil der gemessenen Ortsdosis kann für jeden Messort als nahezu konstant betrachtet werden.

Der umgebungsspezifische Teil umfasst die zeitlich variablen, jedoch für die ganze Messerie einheitlichen Einflüsse (z.B. Einflüsse des Wetters, Transport, Lagerung, Kalibrierung).

Die Summe der orts- und umgebungsspezifischen Komponenten kann somit als effektive Ortsuntergrunddosis betrachtet werden.
Die über mehrere Messreihen bestimmte mittlere Differenz zum Umgebungsmittelwert ergibt nach unserer Interpretation die sogenannten ortsspezifischen Parameter (OSP) für die einzelnen Messorte. Sie werden in der Beweissicherungsphase bestimmt und bei Beibehaltung derselben Messorte, nach der Inbetriebsetzung der Anlage, als konstante Grössen betrachtet. Sie sind nach einer 2-jährigen Beweissicherungsphase statistisch gut gesicherte Grössen mit einer Unsicherheit im Bereich von ca. 0.01 mSv.

Nach der Inbetriebsetzung der Kernanlage ergibt die Summe der effektiven mittleren Umgebungsdosis und des OSP für die einzelnen Messorte den effektiven oder den erwarteten Untergrunddosiswert für den Fall, wo keine zusätzliche Strahlenbelastung stattgefunden hat. Nach der Subtraktion, so bestimmter Erwartungswerte für jeden Messpunkt, von den gemessenen Bruttodosiswerten, ergeben sich die eventuell durch die Kernanlage verursachten Nettodosen. Somit haben die oben erwähnten Einflüsse, die eine signifikante Unsicherheit bei der Bestimmung der Ortsdosis bringen können, für die Bestimmung der Nettodosis praktisch keine Bedeutung. Die erreichbare Dosisunsicherheit der Nettodosiswerte entspricht den Forderungen der Schutzziele der Umgebungsüberwachung.

4. BEISPIEL EINER UMGEBUNGSDOSIMETRIE IN DER SCHWEIZ

Der Umfang der Messprogramme in der Umgebungsdosimetrie ist schon heute recht gross und nimmt noch immer zu. Aus Kapazitätsgründen müssen diese Messungen in einem verhältnismässig einfachen Rahmen als Routineprogramm durchführbar sein. Wenn die beschriebene Interpretationsmethode angewendet wird, erfüllt beispielsweise das nachstehend kurz beschriebene Dosimetriesystem die an das Umgebungsüberwachungsprogramm gestellten Anforderungen. Als Detektoren werden TLD-200 verwendet. Kalibriert wird in Chargen, wobei periodisch die TLD, deren Empfindlichkeit um mehr als 10 % vom Chargenmittelwert abweichen, ausgesondert werden. Bei jeder Dosimeterausgabe wird mit einer Teilcharge der Nulleffekt kontrolliert. In der Mitte der 3-monatigen Expositionsperiode werden Kalibrierdosimeter bestrahlt, mit denen unmittelbar vor der Auswertung der Felddosimeter das Auswertegerät kalibriert wird. Zur Erfassung eventueller Transportdosen werden bei der Ausgabe und beim Rücktransport der TLD Kontrolldosimeter mitgegeben, obwohl eine einheitliche Transportdosis bei der verwendeten Interpretationsmethode keinen unmittelbaren Einfluss auf die Aussagekraft der Messwerte hat.

5. ZUSAMMENFASSUNG

Bei der Beurteilung der TLD-Umgebungsmesswerte ist eine orts- und umgebungsspezifische Interpretation der Daten von grosser Bedeutung. Aufgrund von Daten früherer Messreihen ergeben sich statistisch gut gesicherte Erwartungswerte, auf deren Grundlage die aktuellen Messdaten beurteilt werden können. Dies führt zu aussagekräftigen Resultaten, die den behördlichen Anforderungen an die Umgebungsüberwachung entsprechen.

6. LITERATUR

(1) Richtlinie zur Emissions- und Immissionsüberwachung kerntechnischer Anlagen, BMI, GMBl.1979

(2) E. Piesch, J. Bohm, B. Burghardt, M. Heinzelmann, D. Türk.
FS-81-AKD Dezember 1981

(3) J. Czarnecki
An Interpretation Method of TLD-Einvironmental Data
Health Physics, Vol. 45, No 1 (July) pp. 173-1979, 1983

(3) J. Czarnecki, M. Baggenstos, J. Schuler, H. Völkle
Eine Methode zur Auswertung von Messresultaten der Umgebungsüberwachung von Kernkraftwerken mit Thermolumineszenzdosimetern.
Jahrestagung des Fachverbandes für Strahlenschutz, Lausanne, 30.9.-2.10.1981

SPECIAL ENVIRONMENTAL SURVEILLANCE PROGRAM FOR UF_6 CONVERSION AND URANIUM FUEL FABRICATION PLANTS IN THE UNITED STATES

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INTRODUCTION

In 1977, the U.S. Environmental Protection Agency (EPA) issued regulations setting forth environmental protection standards for uranium fuel cycle facilities. The regulations require that radioactivity in normal effluent releases, radon and its daughters excepted, from uranium fuel cycle plants be limited so that no member of the public will receive an annual exposure resulting in a dose equivalent of more than 25 millirems to the whole-body, 75 millirems to the thyroid or 25 millirems to any other organ. The U.S. Nuclear Regulatory Commission (NRC) is responsible for assuring that uranium fuel cycle plants licensed by the NRC meet these radiation protection standards. The enforcement of the standards has revised the conventional method of environmental monitoring requirements of these facilities. The following sections describe the general procedure and special environmental program required to demonstrate compliance with the EPA standards for two types of nuclear facilities, uranium hexafluoride conversion and uranium fuel fabrication plants. The current status of compliance for these plants is also discussed.

GENERAL PROCEDURE FOR DEMONSTRATION OF COMPLIANCE

The EPA standards require assessment of realistic doses received by individuals living near the plant. The general procedure for demonstration of compliance is to perform a dose calculation using measured effluent release data and applying environmental pathway models for dose calculations. For the uranium hexafluoride conversion and uranium fuel fabrication plants, the critical pathway of dose to man is from direct inhalation. The liquid pathway is not a realistic pathway, since there is no nearby resident using surface water in streams where the plants' liquid effluents discharge. Most of the dose received is expected from the air pathway. Table 1 summarizes the current release rate of radionuclides in air effluents from these facilities in the United States. At present there are two uranium hexafluoride conversion facilities and seven fuel fabrication plants in the United States. The release rates in Table 1 vary substantially because of differences in the types of processes, operational rates and effluent treatment systems. Also included in Table 1 are the distances from the plant to the nearest resident who is expected to be highest exposed from the plant's release. For dose calculation purposes, the critical organs from the intake of uranium are either the lung or the bone depending on the solubility classification of uranium. Table 2 summarizes the air concentrations of uranium of different solubility classification and particle size which, if breathed for one year, would result in a bone or lung dose of 25 mrem. The permissible concentrations were derived using 50-year committed dose conversion factors generated from the ICRP

Task Group Lung Model, and 80% occupancy time and inhalation rate of 8,000 m^3/yr for an adult. As shown in Table 2, the permissible uranium concentrations could vary by order of magnitude depending on the solubility classification and the particle sizes which are highly important in dose calculation.

SPECIAL MONITORING PROGRAM REQUIRED

To demonstrate compliance with the EPA standards, measurement has to be made of uranium solubility and particle size to provide sufficient information for dose assessment unless the plant meets the standard by conservative calculation. The measurements of solubility and particle size of uranium in stack effluents from plant releases are generally difficult and expensive, since plants typically have multiple stacks. The preferred method of measurement is to take samples for analysis from the location of the nearby residence. The cascade impactor is normally used to determine the activity median aerodynamic diameter (AMAD) of the particle size. High volume air samplers are used to collect air particulates for the solubility test and the dissolution rates in simulated lung fluids are determined in terms of the ICRP Task Group Lung Model. In addition to these measurements, continuous air sampling is conducted at the nearby residence to measure annual average concentration of uranium rather than relying on measurements of all plant effluents and applying a meteorologial model such as the Gaussian plume model to calculate the uranium concentration at the residence. This is necessary at plants which have multiple stacks and when the nearest resident is so close to the plant that the normal Gaussian plume model cannot be applied.

For analyzing air particulates, the conventional method of gross alpha analysis of the air filter cannot be used since the method is not capable of measuring air concentrations as low as 10^{-15} to 10^{-16} microcuries per milliliter. The collected air samples must be composited and analyzed either by the fluorimetric method or by gamma spectroscopy. At one facility, residences are located directly across a street within 50 meters from the plant and there are more than 30 small discharge stacks. The points of maximum impact from the plant's discharge must be determined by measuring the isopleths of uranium deposition around the plant under typical weather conditions. The above special monitoring program was not required before the more restrictive standards became effective.

CURRENT COMPLIANCE STATUS

Table 3 summarizes the current compliance status of the EPA standards for these plants. At present, all of the facilities are in compliance.

Table 1
Annual Routine Releases of Uranium from UF_6 and Fuel Fabrication Facilities

Facility	Release Rate (Ci/yr) from Stacks	Major Uranium Compounds	Nearby Resident from Plant (m)
A. UF_6 Conversion Facilities			
1. Allied Chemical Corp.	2.8×10^{-1}	U_3O_8, UO_3, UO_2F_2, UF_4	430 meter
2. Kerr-McGee Corp.	1.4×10^{-1}	U_3O_8, UO_3, UO_2F_2, UF_4	800 meter
B. Fuel Fabrication Facilities			
1. Exxon Nuclear Co. (UF_6 to UO_2 & pelletizing)	3.9×10^{-5}	UO_2F_2, UO_2, U_3O_8, $(NH_4)_2U_2O_7$	3,350 meter
2. Westinghouse Electric Corp. (UF_6 to UO_2 & pelletizing)	1.7×10^{-3}	UO_2F_2, UO_2, U_3O_8, $(NH_4)_2U_2O_7$	1,100 meter
3. General Electric Co. (UF_6 to UO_2 & pelletizing)	8.5×10^{-4}	UO_2F_2, UO_2, U_3O_8, $(NH_4)_2U_2O_7$	600 meter
4. B & W Co. (Apollo) (UF_6 to UO_2)	9.4×10^{-4}	UO_2F_2, UO_2, U_3O_8, $(NH_4)_2U_2O_7$	50 meter
5. B & W Co. (Lynchburg) (pelletizing)	8.3×10^{6}	UO_2	800 meter
6. Combustion Engineering, Inc. (UF_6 to UO_2)	5.6×10^{-5}	UO_2F_2, UO_2	630 meter
7. Combustion Engineering, Inc. (pelletizing)	4.3×10^{-5}	UO_2	800 meter

Table 2
Permissible Air Concentrations of Uranium in Compliance with EPA Standards Under Various Solubility Classification and Particle Sizes

Uranium Solubility Classification	Permissible Concentration of Uranium (μC/ml)	
	Bone (25 mrem)[b]	Lung (25 mrem)
1. Y Compounds[a]		
A. AMAD[c] = 0.3 micron	7.1E-13[d]	6.0E-15
B. AMAD = 1.0 micron	1.0E-12	9.3E-15
C. AMAD = 5.0 micron	1.7E-12	2.0E-14
2. W Compounds		
A. AMAD = 0.3 micron	4.2E-13	5.9E-14
B. AMAD = 1.0 micron	4.2E-13	9.3E-14
C. AMAD = 5.0 micron	3.4E-13	2.0E-13
3. D Compounds		
A. AMAD = 0.3 micron	1.0E-13	3.5E-12
B. AMAD = 1.0 micron	1.0E-13	5.5E-12
C. AMAD = 5.0 micron	8.1E-14	1.3E-11

[a] Solubility classification based on ICRP Task Group Lung Model.
[b] Dose calculation based on 80% occupancy time and inhalation rate of 8,000 m^3/yr for an adult.
[c] AMAD = Activity median aerodynamic diameter.
[d] E-13 = 10^{-13}

Table 3
Current Compliance Status of EPA Standards for the UF_6 Conversion and Fuel Fabrication Facilities in the United States

Types of Facility	Nearby Resident from Plant	Maximum Critical Organ Dose to the Nearby Resident	
		Lung (mrem)	Bone (mrem)
A. UF_6 Conversion Facilities			
1. Allied Chemical Corporation	430 meter	24.0	9.0
2. Kerr-McGee Corporation	800 meter	6.0	4.0
B. Fuel Fabrication Facilities			
1. Exxon Nuclear Co.	3,350 meter	<1	<1
2. Westinghouse Electric Corp	1,100 meter	3	<1
3. General Electric Co.	600 meter	1	<1
4. B&W Co. (Apollo)	50 meter	6	1
5. B&W Co. (Lynchburg)	800 meter	<1	<1
6. Combustion Engineering, Inc. (Hematite)	630 meter	<1	<1
7. Combustion Engineering, Inc. (Windsor)	800 meter	<1	<1

DAS VERHALTEN VON AKTINIDEN IN DER UMGEBUNG DES KERNFORSCHUNGSZENTRUMS KARLSRUHE

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Zusammenfassung

Geringe Mengen Pu, Am und Cm werden aus der Wiederaufarbeitungsanlage Karlsruhe (WAK) und dem Kernforschungszentrum Karlsruhe (KfK) mit der Abluft und dem Abwasser in die Umgebung emittiert. Aus den Emissionswerten und der gemessenen Verteilung in der Umwelt werden Rückschlüsse auf die Transportmechanismen gezogen, die zu der gefundenen Verteilung führten. Die Dosisexposition der Bevölkerung in der Umgebung des Kernforschungszentrums als Folge der geringen Aktinidenemissionen ist vernachlässigbar klein.

1. Aktinidenemissionen aus dem KfK mit Abluft und Abwasser

Die wichtigsten Aktinidenemittenten des KfK mit der Abluft sind die WAK und die zur Verbrennung brennbarer radioaktiver Abfälle betriebene Verbrennungsanlage. Die Emissionen von ^{238}Pu und $^{239+240}Pu$ mit der Abluft werden in der Verbrennungsanlage seit 1976, in der WAK seit 1977 überwacht. Seit 1980 wird zusätzlich ^{241}Pu gemessen, seit 1982 werden auch ^{241}Am, ^{242}Cm und ^{244}Cm erfaßt. In Tab. 1 sind die kleinsten und größten Emissionswerte pro Monat für die einzelnen Aktiniden aufgeführt sowohl für die gasförmigen Ableitungen als auch für die Emissionen mit den Abwässern des KfK, die zusammen mit dem Abwasser der WAK gereinigt und dann in den als Vorfluter dienenden Altrhein abgegeben werden. Seit Beginn der Plutoniumüberwachung wurden bis Ende 1982 insgesamt 85 MBq ^{238}Pu und 97 MBq $^{239+240}Pu$ mit dem Abwasser in die Umgebung des KfK emittiert. Die jährlichen Emissionsraten liegen zwischen 1,3 und 46 MBq für ^{238}Pu und $^{239+240}Pu$. Die Emissionen von ^{241}Am, ^{242}Cm und ^{244}Cm liegen wesentlich niedriger (^{241}Am: 4 MBq/a, ^{242}Cm: 0,2 MBq/a, ^{244}Cm: 0,9 MBq/a von Januar bis Dezember 1982).

Tab. 1: Aktinidenemissionen aus dem KfK von 1980 - 1982 in MBq/Monat

Nuklid	Abluft		Abwasser
	WAK	Verbrennungsanlage	
^{238}Pu	0,0004 - 5,26	0,0022 - 2,74	0,093 - 5,55
$^{239+240}Pu$	0,0004 - 14,3	0,0015 - 5,20	0,17 - 3,66
^{241}Pu	$\leq$0,037 - 285	0,48 - 186	30 - 244
^{241}Am	0,0016 - 1,85	0,0009 - 1,57	-
^{242}Cm	$\leq$0,0001 - 0,019	$\leq$0,00002 - $\leq$0,0009	-
^{244}Cm	$\leq$0,0002 - 0,087	$\leq$0,00004 - 0,01	-

2. Messung der Aktinidenkonzentrationen in Proben aus der terrestrischen und aquatischen Umgebung des KfK

2.1 Terrestrische Umgebung

Boden: Der Boden in der Umgebung des KfK ist im wesentlichen Sandboden. Die höchsten Pu-Konzentrationen findet man in der 0 - 5 cm Schicht im Bereich der 1. Hauptwindrichtung im Umkreis der WAK: 0,2 - 1,5 mBq ^{238}Pu/g Asche und 0,7 - 1,9 mBq $^{239+240}$Pu/g Asche. Im Bereich der 2. Hauptwindrichtung liegen diese Pu-Konzentrationen bis zum Faktor 10 niedriger. Die ^{241}Pu-Konzentrationen in diesen Proben betrug 4 - 17 mBq/g Asche. Eine Unterscheidung zwischen dem in der gleichen Größenordnung liegenden Fallout-Plutonium und dem Plutonium aus der WAK ist über die Aktivitätsverhältnisse ^{238}Pu/$^{239+240}$Pu und ^{241}Pu/$^{239+240}$Pu möglich.

Luft: In der 1. Hauptwindrichtung von der WAK aus wird seit 1977 die Pu-Konzentration in der bodennahen Luft monatlich gemessen. Für ^{238}Pu wurde im Mittel 0,5 µBq/m³ gemessen (Bereich: 0,1 - 12,5 µBq/m³), für $^{239+240}$Pu wurden 0,9 µBq/m³ nachgewiesen (Bereich: 0,1 - 14,8 µBq/m³). Die Luftkonzentrationen für ^{238}Pu liegen meßbar über den zu erwartenden Fallout-Werten von 0,02 - 0,09 µBq/m³, wohingegen die $^{239+240}$Pu-Werte im Bereich des Fallouts von 0,2 - 1 µBq/m³ liegen.

Pflanzen: In Grasproben wurden 0,04 - 0,2 mBq ^{238}Pu/g trocken und 0,07 - 1,5 mBq $^{239+240}$Pu/g trocken gemessen. Die Pu-Konzentrationen in Kiefernnadeln lagen im gleichen Bereich. Proben von 20 verschiedenen Nutzpflanzen wurden auf ^{238}Pu, $^{239+240}$Pu, ^{241}Am, ^{242}Cm und ^{244}Cm untersucht. Die Aktinidenkonzentrationen lagen praktisch alle unter der erreichbaren Nachweisgrenze. In der 2. Hauptwindrichtung wurden in abgefallenem Laub und abgestorbenen Pflanzenproben von der Bodenoberfläche die höchsten Pu-Konzentrationen aller untersuchten Pflanzenproben gefunden: 5 mBq ^{238}Pu/g trocken, 6 mBq $^{239+240}$Pu/g trocken und 300 mBq ^{241}Pu/g trocken.

Tiere: Der Pu-Gehalt in verschiedenen Organen von Damwild und Kaninchen aus der Umgebung des KfK wurde gemessen. Während die ^{238}Pu-Konzentration durchweg unter der Nachweisgrenze lag, wurden für $^{239+240}$Pu 0,006 - 0,6 mBq/g frisch ermittelt, wobei Knochen, Leber und Nieren die höchsten Plutoniumkonzentrationen aufwiesen.

2.2 Aquatische Umgebung

Sedimente: Im 23 km langen Altrhein, der als Vorfluter für die geklärten Abwässer des KfK dient, ist eine Abweichung von der Fallout-Kontamination in den Sedimenten nur in den ersten 3,5 km nach der Einleitung meßbar. Die maximalen Konzentrationen betragen 6 mBq ^{238}Pu/g trocken und 9 mBq $^{239+240}$Pu/g trocken.

Wasser: Im filtrierten Altrheinwasser wurden 20 m nach der Einleitungsstelle im Monatsmittel mit 0,7 Bq ^{238}Pu/m³ und 1,1 Bq $^{239+240}$Pu/m³ die höchsten Werte gemessen. 3,5 km flußabwärts war bereits der Fallout-Bereich von 0,007 Bq ^{238}Pu/m³ und 0,1 Bq $^{239+240}$Pu/m³ erreicht. Der Pu-Gehalt in den abfiltrierten Schwebstoffen lag bis zum Faktor 50 höher, abhängig von der Kontamination der Sedimente an der Probenahmestelle und der Fließgeschwindigkeit. In ausgewählten Proben wurde ^{241}Pu gemessen. Die ^{241}Pu-Konzentrationen liegen zwischen 4 Bq/m³ und 6 kBq/m³, wobei die Verteilung im Altrhein den anderen Pu-Isotopen entspricht.

Pflanzen: In Wasserpflanzen (Schilf, Schwertlilie, Wasserpest) wurde in den unter der Wasseroberfläche wachsenden Pflanzenteilen 4 - 200 mBq/kg frisch sowohl für ^{238}Pu als auch für $^{239+240}$Pu gemessen. Die Wurzeln wiesen bis zum 20fachen höhere Konzentrationen auf, dagegen war in Pflanzenteile über der Wasseroberfläche der Pu-Gehalt um den Faktor 5 kleiner.

Tiere: In Fried- und Raubfischen wurden $^{239+240}$Pu-Konzentrationen von 0,7 - 160 mBq/kg frisch gemessen, wobei die höchsten Werte in den Gräten und die niedrigsten im Fleisch gefunden wurden. In Bisamratten wurden um den Faktor 10 niedrigere Pu-Gehalte gemessen. In Muschelfleisch wurden 9 mBq $^{239+240}$Pu/kg frisch gefunden, in den Muschelschalen ein 100fach höherer Wert.

3. Rückschlüsse auf Transportmechanismen, die zu der gefundenen Verteilung von Plutonium in der Umgebung des KfK führen

3.1 Terrestrische Umgebung

Ausbreitungsfaktor: Unter der Annahme, daß der ^{238}Pu-Untergrund durch Fall-out in der Umgebungsluft der WAK ebenso vernachlässigbar gering ist wie im Südschwarzwald, wurde mit den gemessenen Luftkonzentrationen und den Emissionsraten für ^{238}Pu der Langzeitausbreitungskoeffizient $\chi = 4{,}8 \cdot 10^{-6}$ s/m^3 berechnet.

Depositionsgeschwindigkeit: Aus den Luftkonzentrationen und den deponierten Pu-Aktivitäten wurden Depositionsgeschwindigkeiten berechnet. Für die Ablagerung mit dem Niederschlag wurde 0,1 - 4,2 cm/s berechnet, die trockene Ablagerung auf Gras ergab 0,4 - 5,0 cm/s und die Gesamtdeposition von Pu-Aerosolen auf den Boden 0,2 - 4,6 cm/s.

Migrationsverhalten: In der Umgebung des KfK nimmt die Pu-Konzentration im Boden wie erwartet mit der Tiefe ab, allerdings wurde in jedem Fall in den Schichten unter 20 cm Tiefe noch ein Teil des abgelagerten Plutoniums gefunden. Dies kann auf den leichten Sandboden zurückgeführt werden, der einen schnelleren Transport in die Tiefe ermöglicht.

Transfer Boden → Pflanze: Die Berechnung des Transferfaktors Boden → Pflanze aus den in Boden- und Pflanzenproben gemessenen Pu-Konzentrationen kann nicht vorgenommen werden, da der Anteil des Plutoniums, der durch Deposition von Pu-Aerosolen auf der Pflanzenoberfläche in die Pflanze gelangt, nicht abgeschätzt werden kann.

3.2 Aquatische Umgebung

Sedimentationsverhalten: Die in den Sedimenten des Altrheins gefundene Pu-Verteilung verläuft parallel zur korngrößenabhängigen Sedimentation der Schwebstoffe. Deren Sedimentation wiederum ist abhängig von gewässerspezifischen Parametern, wie z. B. Abstand von der Einleitungsstelle, Breite, Fließgeschwindigkeit und Wasserdurchsatz.

Transfer Wasser → Lebewesen: Die in den Wasserpflanzen, Fischen, Muscheln und Bisamratten gemessenen Pu-Konzentrationen wurden ins Verhältnis gesetzt zu dem Plutoniumgehalt im Altrheinwasser und den nichtsedimentierten Schwebstoffen. Die erhaltenen Transferfaktoren sind in Tab. 2 zusammengestellt. Die im allgemeinen niedrigen Transferfaktoren zu tierischen Lebewesen werden auch im Altrhein bestätigt. Erhöhte Transferfaktoren bei Pflanzen sind auf abgelagerte Schwebstoffe zurückzuführen.

Tab. 2: Transferfaktoren Wasser → Lebewesen im Altrhein beim KfK

Lebewesen		Transferfaktor Bq/kg frisch : Bq/kg Wasser
Pflanze:	Schilf	0,4 - 100
	Schwertlilie, Wasserpest	4 - 194
Fisch:	Knochen	0,6 - 60
	Eingeweide	0,6 - 5,2
	Fleisch	0,3 - 0,9
Muscheln:	Schale	14 000 - 26 000
	Fleisch	260
Bisamratte:	Knochen	0,3 - 2,2
	Fleisch	0,3 - 1,8
	Leber	0,2 - 1,5
	Lunge	≤ 0,1 - 2,6
	Nieren	2,7 - 7,0

4. Abschätzung der Dosisexposition

Die Dosisexposition der Umgebungsbevölkerung durch die emittierten Aktiniden kann durch Inhalation von aktivitätshaltigen Aerosolen und durch den Verzehr kontaminierter Lebensmittel erfolgen. In Tab. 3 ist die nach der Strahlenschutzverordnung pro Jahr zulässige Aufnahme der Pu-Isotope gegenübergestellt der maximal möglichen Aufnahme in der Umgebung des KfK. Für die Jahresaufnahme durch Trinkwasser und Nahrungsmittel wird dabei angenommen, daß das Altrheinwasser mit jeweils der höchsten gemessenen Konzentration unfiltriert als Trinkwasser benutzt wird und daß ca. 1 000 l/a konsumiert werden (^{238}Pu: 74 mBq/l, $^{239+240}$Pu: 85 mBq/l, ^{241}Pu: 6 Bq/l). Da in allen anderen Nahrungsmittel sehr viel niedrigere Pu-Konzentrationen als im Altrheinwasser gemessen wurden, ist deren Verzehr für die Pu-Ingestion bedeutungslos. Für die Pu-Aufnahme durch Inhalation wurde eine Atemrate von $2{,}32 \cdot 10^{-4}$ m³/s angesetzt und mit der höchsten in der Umgebungsluft gemessenen Pu-Kozentration die Jahresaufnahme abgeschätzt. Die Abschätzung macht deutlich, daß die zur Zeit in der Umgebung des KfK gemessenen Pu-Konzentrationen für die Dosisexposition der Umgebungsbevölkerung auch mit extrem unwahrscheinlichen Annahmen bedeutungslos ist. Dies gilt in noch höherem Maße für Am und Cm, deren Emissionsraten wesentlich unter denen von Pu liegen.

Tab. 3: Maximal mögliche Pu-Aufnahme durch Inhalation und Ingestion in der Umgebung des KfK im Vergleich mit der zulässigen Aufnahme in Bq/a

Nuklid	Jahresaufnahme in Bq/a			
	zulässig nach Strahlenschutzverordnung		in der Umgebung des KfK maximal möglich	
	Inhalation	Ingestion	Inhalation	Ingestion
^{238}Pu	1,1	8 900	0,09	74
$^{239+240}$Pu	0,95	8 000	0,11	85
^{241}Pu	51	$4 \cdot 10^5$	-	$6 \cdot 10^3$

PREVISIONAL METHOD FOR THE EVALUATION OF WATER AND SEDIMENT RADIOACTIVE CONCENTRATIONS IN A RIVER BASIN BASED ON ASSOCIATION OF EXPERIMENTAL RESULTS AND HYDROLOGIC DATA

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For a given radioactive release in a river, the proposed method allows an estimation of radioactive level in water and sediment below a nuclear plant and this for every liquide (Ql) and solide flows occuring an hydrologic cycle.

Non disturbed turbid surface water collected in the river during periods of low-median and high flows are contamined in laboratory with a mixture of gamma emiting radionuclides usually present in the liquid wastes of the tested nuclear plant.

Parallely a statement of the most important locations for use of the river is attempted. For every station transit time (Tt) from the release point is calculated by a mathematical model and this for every liquid flow of the river.

A graph called hydrologic radioactive graph (HRC) presents a very synthetic view of the radioactive situation in the river. For a given release rate expressed in Bq.s-1, we can directly read the corresponding values of activity in water, suspended or bed sediment in Bq Kg-1 (ordinate) versus distance dwonstream the release point (absciss).

This synthesis taken in account main parameters. (Tt, Ql, Qs) allows a very easy and quick evaluation of radioactive situation and is practical use to organize environmental surveillance, to calculate doses to man and to compare previsionnal and actual values.

The behaviours of cesium-quickly and moderately sorbed by sediment - cobalt, progressively and highly associated with suspended matter and iron which precipitates independantly of sediment load are presented, as examples.

A METHOD FOR FIELD DETERMINATIONS OF THE CHEMICAL AND PHYSICAL CHARACTERISTICS OF RADIONUCLIDES AFTER RELEASE INTO THE RIVER WATER

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The paper describes the field methods for the assessment of concentration levels and physical and chemical forms of radionuclides released by nuclear facilities, fall-out and medical uses into the river water. From the extended studies performed on the Po river, results of an experimental check carried out near E. Fermi, ENEL nuclear power plant (PWR) are presented and briefly discussed: samples simultaneously collected in a low rad waste tank, in the first mixing step and in river water at 10 Km downstream were characterized in order to investigate the behaviour of radionuclides into the river water.

EXPERIMENTAL

The system employed[1,2] carries out the separation of suspended matter by a sequential filtration equipment; the last step is a 0.3 um membrane filter. The recovery of ionic and non-ionic forms of radionuclides is obtained by ion exchange beds, alumina and NCFC, selective adsorbers for caesium. The system is able to treat large volumes of fresh water during a few hours; filters, resins and adsorbers are analysed by low background γ spectrometry with detection limit of a few pCi per sample.

RESULTS AND DISCUSSION

A typical situation without radioactive discharges is reported in table I from which the high sensitivities to fall-out and other radionuclide sources is pointed out. In table II, III, IV the distributions of the physical and chemical forms of radionuclides in the samples simultaneously collected during the radioactive waste release are presented. In table II one can see that ^{134}Cs and ^{137}Cs are almost completely in the cationic form; ^{54}Mn and mostly ^{60}Co are bound on solid material, whose composition turned out to be a mixture of amorphous oxides (CRUD). Suitable studies on the CRUD behaviour in the river water showed a full independence of water chemical-physical parameters. Table III reports concentration values and distributions at the first mixing with the river water after 30 min. of contact time and dilution factor 200. From table III and II one can see that caesium isotopes shift from the cationic species toward the solid one and ^{54}Mn, ^{60}Co almost maintain the same distribution. The distribution coefficient (Kd) of table III were calculated without CRUD activity that is assumed not to participate in the exchange mechanisms between water and suspended material. Table IV shows concentration values and distributions of samples collected 10 Km downstream (dilution factor = 140,000; 3 hours of contact time). A progressive shift of cationic forms toward the particulate turns out of table IV; ^{60}Co behaviour is complex owing to significant amounts of particulate, cationic, anionic and non-ionic forms. ^{103}Ru and ^{106}Ru come out in anionic and particulate forms; ^{95}Zr and ^{95}Nb are mostly bound on suspended matter (see table I too).

The adsorption mechanisms of radionuclides on suspended matter were carefully studied (see also paper No. 236 of this Congress) and the relation between specific activity and particle size was investigated: as the adsorption is roughly proportional to the specific area of particles a linear relation between log of radionuclide concentration and log of particle size should be found[3]. The speci-

fic activity of 5 granulometric classes of suspended matter evidenced this relation; for ^{137}Cs the coefficient of linear regression turns out to be -0.8. Then most ^{137}Cs appears to be transported by 5-10 μm particles in suspended sediment of different granulometric distributions (see fig. 1).

The system for field measurements of radionuclides in river water showed high sensitivity for natural and artificial radionuclides in order to follow the shift of physical and ionic characteristics. The fate of most radionuclides and of ^{137}Cs particularly is connected with the suspended sediment transport in the Po river. The granulometric class more significant for caesium transport has 5-10 μm particle size.

REFERENCES

1) D.E. Robertson et al. "Transport and depletion of radionuclides in the Columbia River"in Radioactive Contamination of the Marine Environment IAEA, Vienna 1973, p. 141-155.
2) A. Battaglia et al. "Sistema per il prelevamento e la determinazione contemporanea di un ampio spettro di radionuclidi in acque naturali". Simposio sulle Metodologie Radiometriche e Radiochimiche nella Radioprotezione. Pavia 1980.
3) W.W. Sayre et al. ('1963) with reference to J.L. Glenn in "Relations among Radionuclide Content and Physical Chemical and Mineral Characteristics of Columbia River Sediments". Report TID-25786 (1971).

TABLE I - Concentration (pCi/m^3 ± 2 σ) and physical chemical distributions of radionuclides (June 12, 1981). Suspended matter = 125 mg/l. Flowrate = = 605 m^3/sec.

	^{7}Be	^{95}Nb	^{95}Zr	^{103}Ru	^{106}Ru	^{125}Sb	^{131}I	^{177}Cs
Particulate	136±18	127.6±4.8	82.3±4.4	23.0±2.4	58±15	5.1±4.5	S.S.(*)	64.5±3.7
dissolved	S.S.(*)	15.4±2.1	12.2±2.0	36.1±2.6	76±13	2.2±2.0	14.4±1.9	7.6±1.4
% Particulate	100	89	87	39	43	70	-	89
% Cationic	-	2	2	1	5	-	-	9
% Anionic	-	6	7	48	42	30	100	2
% Non-Ionic	-	3	4	12	10	-	-	-
Kd (cm^3/g)	-	66,000	54,000	5,100	6,100	∿18,000	-	68,000

(*) S.S. = below the detection limit.

In the particulate form were moreover measured (pCi/m^3): ^{59}Fe 54.3±5.0, ^{60}Co 3.6±2.0, ^{203}Hg 66.3±4.0.

TABLE II - Concentration (pCi/l = 2σ) and physical chemical distributions of radionuclides in a low rad waste tank (June 10, 1981).

	^{54}Mn	^{60}Co	^{125}Sb	^{134}Cs	^{137}Cs
Particulate	399±35	16,110±110	309±43	283±38	2,620±75
dissolved	1,696±151	12,390±150	87±69	9,110±210	69,610±390
% Particulate	19	56.0	78	4	4
% Cationic	81	34.0	-	96	96
% Anionic	-	8.4	15	-	-
% Non-Ionic	-	0.7	7	-	-

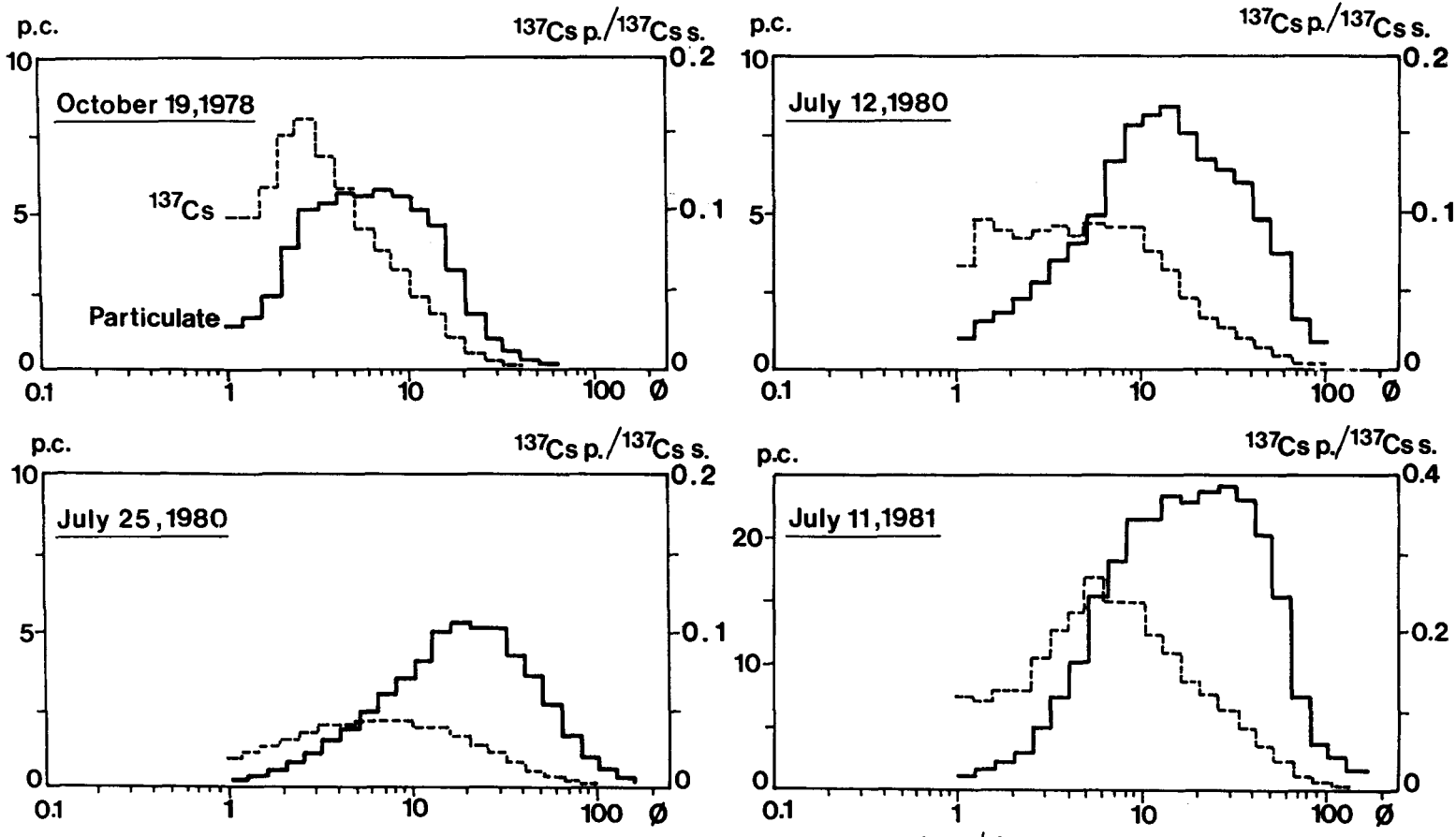
TABLE III - Concentration (pCi/l ± 2σ) and physical chemical distributions of radionuclides at the first mixing with river water (June 10, 1981), Suspended matter = 111 mg/l.

	^{54}Mn	^{60}Co	^{125}Sb	^{134}Cs	^{137}Cs
Particulate	2.42±0.68	87.5±1.1	1.23±0.83	16.1±0.8	140.3±1.6
dissolved	8.17±0.54	65.6±2.3	0.75±0.17	27.1±1.7	210.7±11.6
% Particulate	23	57.1	62	37	40
% Cationic	77	37.1	-	63	60
% Anionic	-	4.3	38	-	-
% Non-Ionic	-	1.5	-	-	-
Kd (cm^3/g)	3,000	1,800	11,000	5,700	6,200

TABLE IV - Concentration (pCi/m^3 ± 2σ) and physical chemical distributions of radionuclides at 10 Km downstream (June 10, 1981). Suspended matter = = 111 mg/l. Flowrate = 650 m^3/sec.

	^{54}Mn	^{60}Co	^{95}Nb	^{95}Zr	^{103}Ru	^{106}Ru	^{134}Cs	^{137}Cs
Particulate	5.1±2.4	145.8±5.8	134.9±8.9	84.8±8.9	19.8±4.9	49±9	41.5±5.8	335.3±9.7
dissolved	9.5±2.2	68.4±3.4	16.3±2.7	8.5±1.9	36.0±3.2	68±15	27.0±1.5	221.0±3.3
% Particulate	35	68.1	89	91	35	42	61	60
% Cationic	65	21.1	2	-	5	-	39	40
% Anionic	-	8.1	6	6	46	44	-	-
% Non-Ionic	-	2.7	3	3	14	14	-	-
Kd (cm^3/g)	4,800	5,200	74,000	90,000	5,000	6,500	13,200	10,700

Were moreover measured (pCi/m^3): ^{7}Be 120±43; ^{131}I 15.3±3.2 (88% anionic and 12% non-ionic).


p.c.
^{137}Cs p./^{137}Cs s.
October 19,1978
^{137}Cs
Particulate
July 12,1980
July 25,1980
July 11,1981
Ø
p.c. = particulate concentration (mg/l)
^{137}Cs p./^{137}Cs s. = ^{137}Cs particulate/^{137}Cs soluble
Ø = diameter (μm)

DISTRIBUTION IN THE PO RIVER OF RADIONUCLIDES RELEASED BY NUCLEAR POWER PLANTS SOME RESULTS RELEVANT TO THE YEARS 1978-1982

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Wide investigations on the environmental effects of nuclear power plants, at Trino Vercellese (PWR, in operation since 1963) and at Caorso (BWR, started in 1981) were carried out in the period 1978-1982. The study was concerned with over 200 Km of the Po river (upstream Trino to downstream Caorso), a few nearby low marshy lands, the related main irrigation canals and the involved agricultural lands (Fig. 1a).

Several environmental matrices such as sediments, fishes, water, aquatic vegetables, agricultural soils and products were analysed by γ-spectrometry with low background Ge(Li) detector facilities. A special care was taken of the radioactivity space and time variation analyses and of the assessment of deposition and self-purification phenomena. The radionuclide concentrations in samples were then correlated to the activity released by power plants and to the river flowrate.

SEDIMENTS

Sediment samples (over 650 in five years) were collected across 29 transects along the Po river. Other 96 samples were collected in the Lanza irrigation canal (near Casale Monferrato; $\sim$15 Km downstream Trino V.) in 9 central points along the water body ($\sim$ 20 Km). Some artificial radionuclides coming from nuclear plants, fall-out and medical uses were detected in the sediments; the detection frequencies ($\frac{n}{N}$ %) for the Po river and Lanza canal were, respectively:

	^{137}Cs	^{60}Co	^{95}Zr	^{7}Be	^{134}Cs	^{125}Sb	^{144}Ce	^{106}Rh	^{141}Ce	^{110m}Ag	^{54}Mn
Po river	100	59	43	42	31	18	16	16	14	10	9
Lanza canal	100	99	40	54	71	23	11	7	5	2	2

The data examination was performed by several statistical methods following three main ways:-search for relations among characters of samples (organic content, density, natural radioactivity, etc.) and conc. of artificial radionuclides; -radioactivity space and time variation analyses;-correlations among sediment activities, flowrate and discharges of power plants.

About the relations among sample parameters high correlations between U and Th series, and among ^{137}Cs, ^{60}Co and the organic content was found: a higher accumulation capability turned out in the sediments with higher organic content.

The radioactivity space and time distribution analyses pointed out that:- as to concentrations of U and Th the Po river appears to be divided into two lengths from the entry of Sesia and Ticino tributaries;-owing to a dam, Casale M. site seems to be critical, especially for ^{137}Cs and ^{60}Co accumulations;-during 1979 a progressive accumulation of ^{137}Cs and ^{60}Co along the whole river was recorded followed by a fast self-purification process. An accumulation in the terminal zones of Lanza canal was also noticed; the time distribution follows that of the Po. These phenomena were recognized to be mainly related to hydrological trends of the river and to the hydraulic local peculiarities of the two water bodies.

To study the correlation among flowrate, liquid radioactive discharges and concs, in river sediments the trends of the three variables were examined for the period 1977-1982. Figs. 1 b,c,d report these correlations referred to Casale M. site; it can be seen that the sediment accumulates the ^{137}Cs and ^{60}Co discharges until flowrates of 800-900 m^3/s (3-4 times the average flowrate) give rise to purification of the site.

FISHES

The samples of ubiquitarian and prevailing species in the Po river (Barbus barbus, Leuciscus cephalus, Chondrostoma genei), were systematically caught, in 11 sites, using the traditional methods of commercial fishing. The classical biometric parameters were measured over the whole caught population (3,000 fishes). The main statistical indexes and tests were performed over the data divided into species, period and site fishing, in the aim of characterizing the stocks used for human alimentation.

The γ-spectrometric analyses of the 277 samples of the three species showed the non-significance of the concentration values from the radioprotection viewpoint. The most frequently detected nuclides were ^{137}Cs (74%) from double source (nuclear plants and fall-out) and the mean half-life nuclides ^{75}Se (52%) and ^{203}Hg from medical uses.

Space and time distributions of ^{137}Cs in fishes follow the same trends as in the sediments; some delay (although scarcely proved by statistical tests) seems, however, to occur in the time.

REFERENCES

- E. Bazzano: "Indagine Radioecologica in una zona agricola irrigata con le acque del fiume Po", paper presented at 1° Convegno Nazionale di Radioecologia Agraria, Piacenza 1980.
- R. Artioli, A.L. Traversi: "La Radioattività nei sedimenti del tratto di fiume Po da monte Trino a valle Caorso: alcuni risultati relativi al periodo 1978-82", paper presented at 3° Convegno Nazionale di Radioecologia, Bologna 1983.

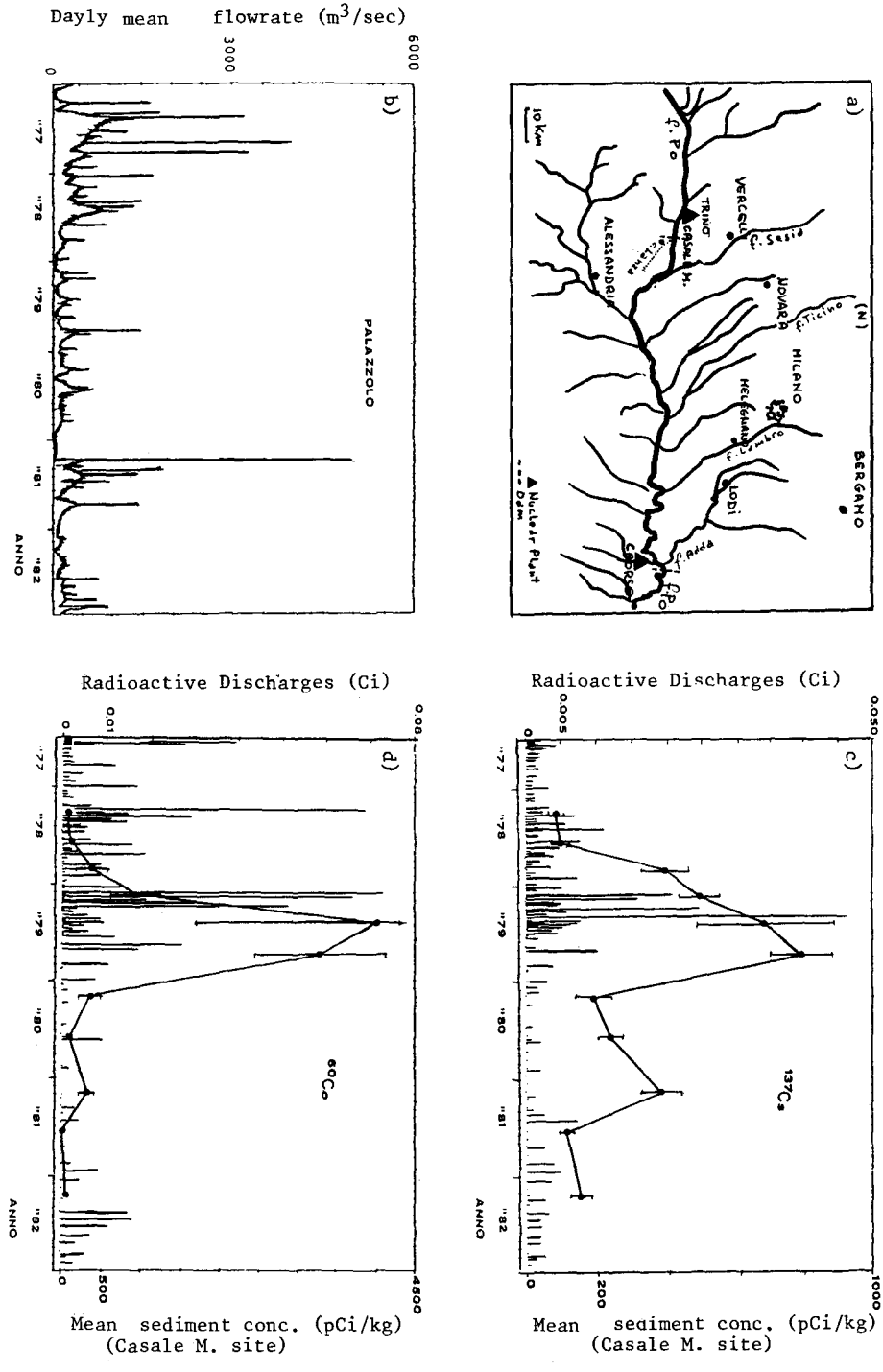
a)
10 Km
f. Po
VERCELLI
f. Sesia
TRINO
CASALE M.
ALESSANDRIA
NOVARA
f. Ticino
(N)
MILANO
f. Lambro
LODI
BERGAMO
Nuclear Plant
Dam
f. Adda
b)
Dayly mean flowrate (m^3/sec)
0
3000
6000
PALAZZOLO
77
78
79
80
81
82
ANNO
c)
Radioactive Discharges (Ci)
0.005
0.050
^{137}Cs
200
1000
Mean sediment conc. (pCi/kg)
(Casale M. site)
d)
Radioactive Discharges (Ci)
0.01
0.08
^{60}Co
500
4500
Mean sediment conc. (pCi/kg)
(Casale M. site)

Fig. 1

GESCHICHTE DER ABLAGERUNG VON RADIONUKLIDEN IN DEN SEDIMENTEN EINER FLUSS-STAUSTUFE

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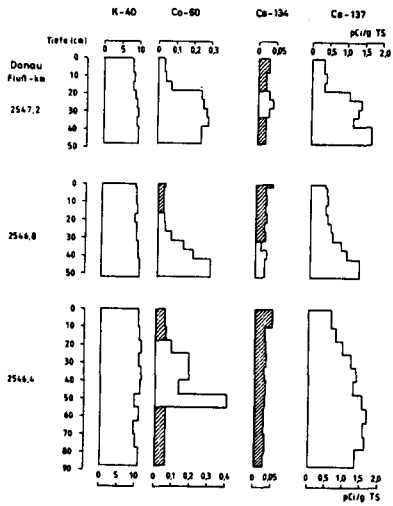
Die Staustufe Faimingen an der Donau wurde im Jahre 1965 errichtet. Sie beginnt beim Wasserkraftwerk Gundelfingen bei Fluß-km 2545,5. Im Herbst 1966 wurde das Demonstrations-Kernkraftwerk REW-Bayernwerk in Gundremmingen (KRB) mit einer elektrischen Bruttoleistung von 237 MW in Betrieb genommen. Seine Kühl- und Abwässer werden bei Fluß-km 2549,0 rechts in die Staustufe Faimingen eingeleitet. KRB wurde nach einem Störfall im Jahre 1977 nicht mehr in Betrieb genommen, die Einleitung von radioaktiven Abwässern ist jedoch noch nicht beendet.

Seit 1967 werden im Rahmen eines Forschungsvorhabens Radioaktivitätsmessung im Wasser und hydrobiologischen Material in der Staustufe Faimingen durchgeführt. Diese Staustufe eignet sich aus diesem Grund gut zur Untersuchung des Verbleibs der radioaktiven Emissionen und zur Bestimmung der Sedimentationsraten in staugeregelten Flüssen.

Im Juli 1982 und März 1983 wurden in der Staustufe Sedimentbohrkerne bei Fluß-km 2547,2, 2546,8 und 2546,4 jeweils links und rechts vom Flußschlauch genommen, der Länge nach aufgeteilt und γ-spektrometrisch analysiert. Diese Querprofile wurden aufgrund von Tiefenprofilmessungen des Betreibers der Wasserkraftwerke als Stellen erhöhter Sedimentation ausgesucht.

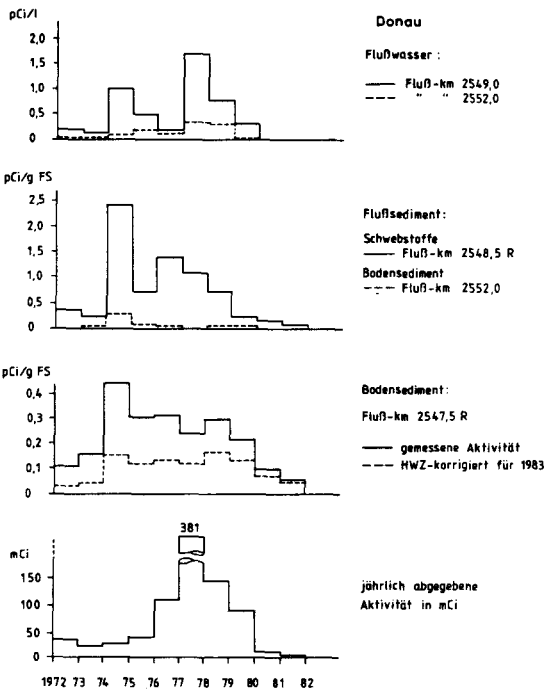
Die spezifischen Aktivitäten entlag der Sedimentbohrkerne aus dem rechten Vorland zeigt Abbildung 1. Ein deutliches interpretierbares Maximum wurde im Bohrkern bei Fluß-km 2546,4 rechts in Tiefe von 50 cm bei Co-60 gefunden. In den zur KRB- Einleitung näher gelegenen Bohrkernen ist mit einer Steigung der spezifischen Aktivität in tieferen Schichten zu rechnen. Im linken Vorland wurden dagegen keine bedeutenden Unterschiede in der Aktivität der einzelnen Schichten festgestellt, was auf einen ausgeprägten Fahneneffekt schließen läßt.

Abb. 1 Spezifische Aktivität in den Sedimentbohrkernen in pCi/g TS



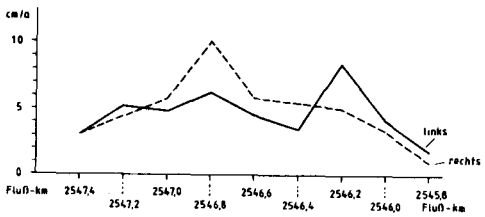
Der zeitliche Verlauf der in der Staustufe gemessenen Co-60 Aktivitäten ist Abbildung 2 zu entnehmen. Hier fällt das Jahr der höchsten Aktivitätsabgaben (1977) nicht mit dem Jahr der höchsten gemessenen Aktivitätskonzentrationen (1974) zusammen. 68% der 1977 abgegebenen Co-60-Aktivität wurde in den Monaten November und Dezember in die Staustufe eingeleitet. Sie wurde bei einer mittleren Wasserführung (157 m^3/s) und niedrigem Schwebstoffgehalt (0,8 mg/l) in der gelösten Form wahrscheinlich weiter transportiert.

Abb. 2 Spezifische Aktivität von Co-60 in Wasser, Schwebstoffen und Sedimenten, Staustufe Faimingen, 1972-81.



Ordnet man das Maximum im Bohrkern dem Jahr der höchsten gemessenen Aktivitätskonzentrationen in der Staustufe zu, so beträgt die maximale Sedimentationsrate an der Probennahmestelle 6,3-7,5 cm/a. Die Tiefenprofilmessungen der Jahre 1973-82 ergeben für Fluß-km 2546,4 rechts eine mittlere Sedimentationsrate von 5,4 cm/a, Abbildung 3.

Abb. 3 Mittlere Sedimentationsraten in der Staustufe Faimingen 1973-82.

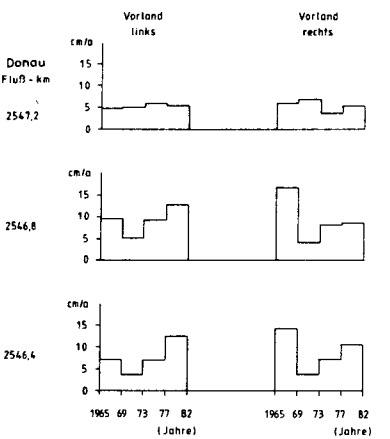


Der relativ steile Abfall der Aktivität in den Bohrkernen bei Fluß-km 2547,2 und 2546,8 läßt sich den Jahren 1978-79 zuordnen (Abb. 1 und 2). Daraus ergeben sich Sedimentationsraten von ca. 5,3 cm/a bei Fluß-km 2547,2 rechts und ca. 8 cm/a bei Fluß-km 2546,8 rechts. (Vergleiche die mittlere Sedimentationsrate in Abb. 3)

Die Tiefenprofilmessungen zeigten, daß bis 1969 die größten Veränderungen der Sohle erfolgten. Die Sedimentation im überschwemmten Vorland war beträchtlich. In den darauf folgenden Jahren blieb der Flußschlauch praktisch unverändert, während die überschwemmten Flächen durch Sedimentablagerungen verlanden; näher zum Flußschlauch stärker als bei den Ufern. Eine deutlich niedrigere Sedimentationsrate wurde in den Jahren 1969-73 an den zum Stauwehr Faimingen näher gelegenen Querprofilen festgestellet (Abb. 4).

Nach Auskunft des Betreibers wird der Abfluß in der Staustufe bis zu Wasserführung Q von 400 m^3/s durch Absenken der Wehrklappen, bei höheren Q-Werten durch Anheben des Wehres geregelt. Bei Q >600 m^3/s ist mit Erosion der Sohle zu rechnen. Ein Blick in das Gewässerkundliche Jahrbuch zeigt, daß das Jahr 1970 ein Jahr der Hochwässer war; an 33 Tagen im Jahr hat die Wasserführung in der Staustufe 400 m^3/s überschritten, während sie im Schnitt nur an 6 Tagen im Jahr überschritten wird. Es ist anzunehmen, daß 1970 die Staustufe Faimingen teilweise ausgeräumt wurde.

Abb. 4 Sedimentationsraten in der Staustufe Faimingen



137-Cs IN ALGAE FROM THE SWEDISH WEST COAST, 1967-1983

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Introduction

In order to follow the long-term trends of radionuclide concentration in the marine environment off the Swedish west coast, measurements on the brown algae Fucus serratus and Fucus vesiculosus collected at a well defined sampling site at Särdal (56.76 deg. N, 12.63 deg. E) were started as early as 1967 and are still in progress. The observed increase in concentration of radioactive activation products in algae from this sampling site has initiated radioecological studies related to the controlled releases from the Barsebäck nuclear power plant (1,2) as well as studies of radionuclides released from nuclear fuel reprocessing plants (3) and those present in fall-out from nuclear weapons tests (4). Even though the concentration factor for 137-Cs from water to Fucus is much lower than for most other radionuclides of interest it is posible to use Fucus as an indicator for the 137-Cs present in the marine environment. The purpose of this paper is to summarize the long-term variation of the 137-Cs concentration and to discuss possible sources of this 137-Cs.

Materials and Methods

Samples of the brown algae Fucus serratus containing minor amounts of Fucus vesiculosus were collected at the shore at Särdal (56.76 deg. N, 12.63 deg. E) at water depths of between 0.5 and 1 m. All the material collected was firmly rooted to rocks under the water surface. After drying in air for 2-3 days in the laboratory and grinding, the samples were packed in 180 ml cylindrical plastic containers and analysed using a Ge(Li) detector. Counting times varied between 24 and 100 hours. After these measurements the samples were dried at 105 deg. C for 24 hours to obtain the reference weights. In July 1977, June 1981 and June 1982 samples of Fucus were also collected at other places along the Swedish west coast. The samples were collected, treated and analysed in the same way as described above. For further details see Mattsson et al. (1) and Nilsson et al. (2).

Results and Discussion

The variation in the 137-Cs/K ratio in Fucus during the period 1967-1983 (16 years) is shown in figure 1. As the potassium concentration is essentially constant throughout the year the time variation is almost entirely due to the variation in the 137-Cs concentration. In periods of more frequent sampling (1978-1983) a pronounced seasonal variation in the 137-Cs/K ratio is seen with maximum values in summer and minimum values in winter. The results indicate considerable differences between the uptake and retention of caesium and potassium in Fucus. In this context it may be

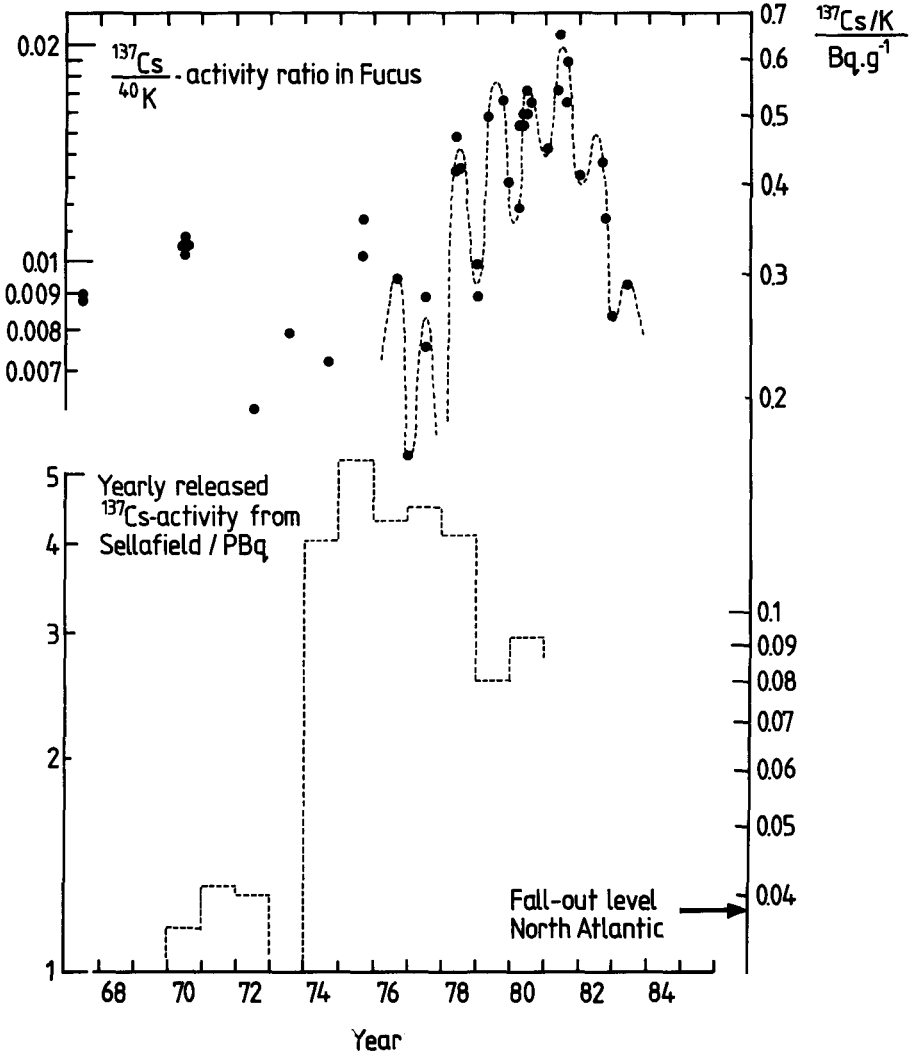

Fig.1. Ratio between 137-Cs and 40-K activity (left scale) and the 137-Cs activity per unit mass potassium (right scale) in samples of Fucus serratus and Fucus vesiculosus collected at Särdal (56.75 deg. N, 12.63 deg. E) on the west coast of Sweden during the period 1967-1983. The average potassium concentration was (23.1 ± 1.6) g per kg dry algae, which is equivalent to (760 ± 49) Bq 40-K per kg dry material. The reported releases of 137-Cs into the Irish Sea from the reprocessing plant for spent nuclear fuel at Sellafield (formerly Windscale) are shown in the bottom half of the figure.

interesting to note that for other artificially produced radionuclides the seasonal variation may be quite different from that of 137-Cs. For example, data for 99-Tc analysed in the same samples show maximum values in winter and minimum values in summer (3). It is extremely improtant to take such seasonal variations into consideration when Fucus is used to monitor the temporal as well as spatial distribution of a specific radionuclide or of activity ratios between two radionuclides, e.g. 99-Tc/137-Cs.

Even though the seasonal variation and the biological variability between samples, to a certain extent, masks the long-term trends, our results show a significant increase in the 137-Cs/K ratio starting in 1978 and reaching a maximum in 1981. Between 1981 and the end of 1983 the ratio has decreased with a half-life of approximately 2 years.

The measurements of the 137-Cs/K ratio in Fucus from other parts of the Swedish west coast collected on three occasions, once before the increase (July, 1977), and twice during the period of the maximum (June, 1981 and June, 1982), do not indicate any significant contribution from the small releases of 137-Cs from the nuclear power plants at Barsebäck and Ringhals. The most likely explanation is therefore that the increase in the 137-Cs content is mainly due to releases from the European reprocessing plants for spent reactor fuel; the facility at Sellafield (formerly called Windscale) dominating the releases (5). The yearly releases from Sellafield (6,7) are also shown in figure 1. The figure clearly shows that the time delay between the large releases of 137-Cs during the period 1974-1978 and their appearance at the Swedish west coast is about 4 years. This result is in good agreement with that of Aarkrog et al., (8) who followed the 137-Cs concentration in the inner Danish waters. They concluded that in 1979 nearly all the 137-Cs found in water samples originating in the North Sea (salinity > 3%) was of Sellafield origin, and that the transport time from Sellafield to the Danish waters was about 4 years. It is thus reasonable to assume that from 1978 a dominating fraction of the 137-Cs found off the Swedish west coast was of Sellafield origin.

In the period 1967-1977 the 137-Cs/K ratio appears to be rather too high to be explained by the release of 137-Cs from Sellafield. Let us consider the contribution to the total 137-Cs concentration from fall-out. From measurements on Fucus from the south coast of Greenland by Aarkrog et al. in June, 1979 (9), one can estimate the North Atlantic fall-out contribution to be 0.04 Bq/g K. Using an average value of the global fall-out concentration for 137-Cs of 7 mBq/l in North Atlantic water, and a concentration factor of 100 for water to dry Fucus (2) gives a fall-out contribution of about 1 Bq 137-Cs/kg dry Fucus or 0.04 Bq per gram K. As this contribution is insignificant for the period 1967-1977 we could expect a similar time variation pattern for the 137-Cs/K ratio in Fucus to that of the release rate of 137-Cs from Sellafield 4 years earlier. This was, however, found not to be the case. The time variation for 137-Cs in Fucus is not so pronounced as the variation in 137-Cs release rate, indicating another significant 137-Cs source. The explanation may be that the fall-out component in Fucus from the Swedish west coast is about 5 times higher than estimated for the North Atlantic (due to run-off from the land), or that the releases from La Hague, even if small, are more easilly transported to the

Cattegat area (the sound between Denmark and southern Sweden) than the Sellafield releases.

Conclusions

In the period 1979-1981, the contriburtion to the 137-Cs level present in the marine environment off the west coast of Sweden, due to releases from the European processing plants for spent nuclear fuel, was in the region 70-90%. In 1983 the corresponding contribution was 55-85%.

References

1) Mattsson, S., Fink, R. and Nilsson, M. Environ. Pollut. (Ser. B) 1, 105-115 (1980)

2) Nilsson, M., Dahlgaard, H., Edgren, M., Holm, E., Mattsson, S. and Notter, M. In: Impacts of Radionuclide Releases into the Marine Environment, IAEA, Vienna. pp 501-513 (1981)

3) Rioseco, J., Holm, E. and Mattsson, S. To be published

4) Holm, E., Persson, B.R. and Mattsson, S. In: Radiation Protection. Proc. 5th Congr. IRPA, Pergamon Press, Oxford, pp 1103-1106 (1980)

5) Jefferies, D.F., Steele, A.K. and Preston, A. Deep-Sea Research, 29, 713-738 (1982)

6) Department of the Environment Scottish Office, Welsh Office, London, UK. Annual Survey of Radioactive Discharges in Great Britain 1980 (1981)

7) Atherton, R.S. Private communication, British Nuclear Fuels Service Limited, Warrington, UK (1983)

8) Aarkrog, A., Bøtter-Jensen, L., Dahlgaard, H., Hansen, H., Lippert, J., Nielsen, S.P. and Nilsson, K. Risø Report 421 (1980)

9) Aarkrog, A., Dahlgaard, H., Holm, E., Lippert, J. and Nilsson, K. Risø Report 423 (1980)

ON-LINE LIQUID-EFFLUENT MONITORING OF SEWAGE AT LAWRENCE LIVERMORE NATIONAL LABORATORY

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INTRODUCTION

Since its beginning in 1952, the Lawrence Livermore National Laboratory (LLNL), has supported an environmental surveillance program to determine its impact, if any, on the local environment. LLNL, located in the city of Livermore which is 65 km east of San Francisco, is involved in widely varied research and development programs for the U.S. Department of Energy (DOE). The main efforts are in nuclear and nonnuclear weapons, magnetic and laser fusion energy, biomedical research, and nonnuclear energy technologies, such as geothermal power and fossil-fuel utilization. These programs and their support groups generate waste products that could have a negative impact on the environment if improperly managed.

LLNL's sanitary sewer system is a possible route for the escape of toxic materials. Liquid effluents are released to Livermore's sanitary sewer system and the effluent is treated at the Livermore Water Reclamation Plant (LWRP). The plant is a secondary-treatment operation that returns most of the water to the San Francisco Bay via a transport pipeline. The remaining portion is used for irrigating vegetation along the roadways and a local golf course.

Small samples of the effluent stream are taken at regular intervals to give us a composite sample of all the liquid effluent that leaves the LLNL. These results provide a record of the effluent by which LLNL is assured of compliance with local regulations.

The sewer-monitoring system was designed and constructed to detect toxic-material releases and to facilitate immediate response to such releases. This is necessary in preventing damage to the LWRP and regions receiving discharged effluent. If toxic materials are detected at levels that exceed predetermined LLNL alarm limits, signals are sent to a central alarm station that is manned 24 hours a day and a sample of the suspect toxic effluent is automatically collected. Since at least four hours pass before LLNL effluent reaches the treatment plant, sufficient time is available to alert emergency personnel, evaluate the situation, and, if necessary, arrange for diversion of the material to emergency holding basins at the treatment plant. The toxic waste can be treated in the basins or removed without destroying or reducing the efficiency of the treatment plant.

An automatic on-line, sewage-effluent-monitoring system has been developed that diverts a representative fraction of the total waste stream leaving the site. This portion is monitored for pH, radiation, and heavy metals as it passes through a detection assembly. The assembly consists of an industrial pH probe, two NaI radiation detectors, and an x-ray fluorescence metal detector. A microprocessor collects, reduces, and analyzes the data to determine if the levels are acceptable by established environmental limits.

MONITORING SYSTEM COMPONENTS

The on-line monitoring system consists of several components. The flow route to these components starts at a point in a manhole where all LLNL sewage-discharge lines converge. As the sewage flows through a Parshall flow-measuring flume, approximately 40 liters/minute of the sewage is pumped to an aboveground building where the detection instrumentation is located. Inside this building, the sample enters a tank housing the high- and low-energy radiation detectors, pH probe, and a sample line leading to the metal analyzer unit. After it is scanned, the sample is returned to the sewer via an outlet pipe.

RADIATION DETECTORS

Radioisotopes being used at LLNL that could be released in quantities exceeding regulatory guide levels are ^{90}Sr, ^{235}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{241}Am, ^{244}Cm, and ^{3}H. All of these, with the exception of ^{90}Sr and ^{3}H, emit heavy element x-rays and low-energy gamma rays during decay. For ^{90}Sr, low-energy bremsstrahlung photons give an indication of specific activity. Cate and Hoeger (1) designed a radiation detection system with the primary detector, a 3 x 127 mm NaI crystal separated from the sample by a thin polycarbonate plastic sheet. This allows recognition of the low-energy x rays, gamma rays, and bremsstrahlung photons in the range of 10 to 100 keV. In addition, a 50 x 50 mm NaI crystal detector is mounted immediately adjacent to the sample tank to provide detection of high-energy (100-1000 keV) events. The low-energy beta particles emitted from tritium will not be detected by this system. A separate tritium detection unit is being developed.

This system gives us a minimum detectable activity in a 10-minute count time (95% confidence level) of 1.6 x 10^{-6} µCi/ml or 0.016 of the concentration guide for ^{239}Pu, and 3 x 10^{-6} µCi/ml or 0.3 of the concentration guide for Sr^{90} (17-minute count at 95% confidence level). The concentration guides are limits of continuous release for a 40-hour work week. These are found in DOE Order 5480.1 (2).

The electronics associated with the detectors includes single-channel analyzers and a Digital Equipment Corporation LSI-11 microprocessor which compares the data to preset alarm levels. The count rates are checked every minute. If the current rate is greater than the alarm levels, the count rate over the past 60 minutes is averaged. If this average exceeds the preset levels, an alarm is sent. This is a way of monitoring for low-level continuous releases.

pH MONITOR

There are many locations in LLNL where acid and basic solutions are used routinely. An accidental release of extreme pH levels could damage the LWRP. Therefore, the pH is monitored continuously by a commercial industrial pH probe housed in the tank holding the radiation detectors. Output from the probe is recorded on a seven-day circular chart and is sent to the DEC LSI-11 microprocessor to be evaluated. The average pH level is checked every minute. If the high or low level is exceeded, the pH from the previous 30 minutes is averaged. The low and high pH limits have been set taking into account pH and length of time of release.

METAL DETECTOR

The deleterious effects of excess concentrations of heavy metals on the necessary bacteria population in the sewage treatment plant prompted a search for a continuous metal-detection system that would prevent the recurrence of plant down-time caused by a metal release to the sewer. Available commercial units were not suitable for sewage analysis. Cate, Matthews and Rueppel then designed a monitoring unit capable of detecting hazardous concentrations of ions found to be most harmful to the bacteria in the treatment plant process, specifically copper, nickel, chrome and zinc (3). The system is fairly reasonable in cost, works on-line and in real time, does not require extensive pretreatment of the sample, and requires minimal maintenance.

The metal detection unit is an x-ray fluorescence analyzer (XRFA). Its design is based on the principle that elements emit characteristic x-ray lines when excited by a radiation source. These x-ray lines can be measured by energy-dispersion techniques to determine their energy, which permits species identification, while the intensity of the lines is proportional to concentration.

A portion of the sample stream that flows through the tank is routed through a grinder to reduce any solids to particles 100 microns or less in diameter. The flow is then introduced through a nozzle into a flume inclined at a 45^{0} angle. The flow spreads to a sheet 1 mm in depth before it reaches the source/detector region of the unit. A thin plastic window backed by air is positioned under the stream in this region to prevent detection of the chrome and nickel in the stainless steel flume. A ^{109}Cd source is used to excite the elements of interest if they are present in the stream, and a commercial xenon-CO_2 mixture x-ray proportional counter is the detector used. The output from the detector goes to amplifiers, then through an analog-to-digital converter interfaced to a DEC LSI-11 microprocessor.

A dual-register system for the data output allows both rapid response to a high metal concentration (500-3600 sec count) and more sensitive response to low concentrations released over a longer time frame (5000-36,000 sec accumulated count). The longer count length is used as the activity of the ^{109}Cd source decays. The data counting system also provides for advisories at concentrations that are elevated, but not at alarm levels.

Maximum permissible discharge concentrations for copper, chrome, nickel, and zinc have been established at LLNL based on studies of the effects of toxic metals on sewage treatment and the dilution factors resulting from the intermixing of LLNL's effluent with the domestic sewage from Livermore. These concentrations, as well as the short and long count time alarm limits for the elements of concern, are shown in Table 1.

GRAB SAMPLER

The identification and concentration of the toxic material is needed to evaluate an alarm situation. An automatic grab sampler extracts a sample from the waste stream as soon as the alarm is tripped. This mechanism can also be activated manually.

TABLE 1. LLNL Maximum Permissible Metal Discharge Limits and XRFA Alarm Limits

Single Metal	LLNL Limit (ppm)	XRFA alarm limit ppm)(a) Short Count	Long Count
Cr	100	107	50
Cu	10	17	10
Zn	50	50	50
Ni	10	23	10
Hg	--	16	10
As	--	10	10
Pb	--	12	4
Combined metal	100		

(a) Count times vary from 500 - 3600 second (short) and 5,000 - 36,000 second (long), depending on the source strength.

SYSTEM MONITOR

Since the operation of the whole monitoring system depends on the pumping of a continuous flow through the instrument tank, we installed a sensitive differential pressure switch on the tank. Changes in pressure drop across the system indicate changes in flow rate and will trigger an alarm. Other problems such as a count rate lower than background and power failures will also send an alarm signal to the central alarm station.

CONCLUSIONS

The on-line sewer monitoring system cannot in itself prevent the accidental discharge of toxic materials into sewers. However, by rapidly detecting releases and sounding alarms to alert emergency personnel to respond, actions can be taken to prevent damage to the environment. Because it is a real-time system, LLNL personnel can more easily locate the source of a toxic discharge and take corrective action to prevent its recurrence.

REFERENCES

(1) Cate, Jr., J. L., and T. O. Hoeger. 1972. "A Radioisotope Monitoring System for Sewage Effluent." Amer. Indust. Hygiene October 1972: 693-699.

(2) U.S. Department of Energy Order 5480.1 Standards for Radiation Protection.

(3) Cate, Jr., J. L., M. A. Matthews and D. R. Rueppel "A Prototype On-line X-Ray Fluoresence Analyzer for Detection of Metals in Sewage", Proceedings of the 34th Industrial Waste Conf., Purdue Univ. 1979. Ann Arbor Science.

ETUDE DE PLUSIEURS ANNEES SUR LES EFFLUENTS RADIOACTIFS LIQUIDES REJETES PAR UNE CENTRALE NUCLEAIRE PWR

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OBJECTIF

La connaissance du terme source est nécessaire pour l'établissement d'un programme de surveillance cohérent d'un site d'implantation d'une installation nucléaire. C'est pourquoi, depuis 1975, des échantillons mensuels représentatifs d'effluents liquides rejetés par la centrale nucléaire PWR de Tihange 1 (890 MWe) dans l'eau de la Meuse ont été analysés au point de vue radioactif, chimique et biologique.

METHODES

Analyses Radiochimiques

Les teneurs en radionucléides des effluents liquides ainsi que des algues ont été déterminées par l'emploi de la spectrométrie gamma (NaI et GeLi) et de méthodes radiochimiques spécifiques (^{3}H, ^{14}C).

Analyses Chimiques

La composition chimique des effluents liquides mensuels (éléments majeurs et éléments traces) a été déterminée par l'emploi des méthodes conventionnelles d'analyse : spectrophotométrie de flamme, absorption atomique, etc...

Analyses Biologiques

Des cultures en conditions contrôlées de l'algue unicellulaire Scenedesmus obliquus ont été réalisées dans chaque échantillon mensuel d'effluents et ce à différentes dilutions. Pour les cultures témoins, on utilise d'une part l'eau distillée et d'autre part le milieu "eau de Meuse" du mois correspondant. La productivité algale est suivie au cours du temps de culture. L'algue Scenedesmus obliquus a été choisie comme matériel biologique parce qu'il s'agit d'une algue commune d'eau douce, représentative d'un des premiers maillons de la chaîne alimentaire. De plus, la multiplication de ce "végétal-test" est rapide et les conditions de culture aisément contrôlables. L'effet des effluents sur la morphologie de ces algues a été évalué en fin de culture par la mesure des tailles des cellules et du degré de groupement des cellules. La fixation des radionucléides et l'incorporation du tritium par les algues ont été également mesurées en fin de culture.

RESULTATS

Caractéristiques Radiochimiques

Le tableau ci-dessous établit le bilan des rejets radioactifs effectués par la Centrale durant la période 1976-1980. On constate que neuf radionucléides sont toujours présents dans les effluents rejetés et que six autres le sont occasionnellement. La gamme des concentrations mesurées varie du $pCi.l^{-1}$ au $nCi.l^{-1}$ excepté dans le cas du tritium dont la concentration est de l'ordre du $\mu Ci.l^{-1}$.

Bilan des rejets liquides de la centrale de Tihange-1 (Ci/an)

Radio-éléments	1976	1977	1978	1979	1980
^{3}H	162	320	356	282	334
^{51}Cr		$4,2x10^{-2}$	$7,7x10^{-2}$	$3,7x10^{-1}$	$9,0x10^{-2}$
^{54}Mn	$3,1x10^{-2}$	$1,8x10^{-1}$	$9,3x10^{-2}$	$3,7x10^{-2}$	$3,0x10^{-2}$
^{57}Co		$2,5x10^{-2}$	$3,0x10^{-3}$	$1,0x10^{-3}$	$7,1x10^{-4}$
^{58}Co	$3,8x10^{-1}$	$1,6x10^{0}$	$1,0x10^{0}$	$3,5x10^{-1}$	$4,0x10^{-1}$
^{59}Fe	$4,2x10^{-4}$	$8,0x10^{-3}$	$2,0x10^{-2}$		
^{60}Co	$2,2x10^{-2}$	$8,3x10^{-1}$	$2,7x10^{-1}$	$3,9x10^{-1}$	$4,8x10^{-1}$
^{65}Zn		$4,0x10^{-3}$		$5,0x10^{-3}$	$1,0x10^{-2}$
^{95}Zr	$1,9x10^{-4}$	$5,0x10^{-3}$		$4,0x10^{-3}$	$1,3x10^{-2}$
^{95}Nb	$3,2x10^{-4}$	$9,0x10^{-3}$	$5,0x10^{-3}$	$2,8x10^{-2}$	$3,8x10^{-2}$
^{110m}Ag				$1,0x10^{-3}$	
^{131}I	$3,0x10^{-1}$	$2,8x10^{-1}$	$2,0x10^{-1}$	$4,5x10^{-2}$	$1,4x10^{-1}$
^{133}I					$1,0x10^{-1}$
^{134}Cs	$3,5x10^{-2}$	$2,0x10^{-1}$	$8,4x10^{-2}$	$2,2x10^{-2}$	$1,9x10^{-2}$
^{137}Cs	$6,4x10^{-2}$	$3,5x10^{-1}$	$1,7x10^{-1}$	$4,3x10^{-2}$	$3,7x10^{-2}$

Caractéristiques Chimiques

D'une manière générale, la centrale produit très peu d'effluents de caractéristiques chimiques particulières. Les effluents liquides sont des eaux déminéralisées contenant quelques milligrammes d'acide borique et polluées par les eaux de buanderies et de lavages de sol. Exceptionnellement, ces eaux peuvent contenir un peu de chrome (provenant de la réduction des chromates utilisés comme inhibiteurs de corrosion) et des produits de décontamination (réduction du permanganate de potassium et des acides faibles). Les détergents utilisés pour le linge ou les sols sont des produits biodégradables. En marche normale, la centrale rejette de manière continue 15 à 20 m3/h d'eau déminéralisée (purges de déconcentration des générateurs de vapeur) dans lesquels elle injecte environ 20 m3/jour d'effluents "dits" radioactifs (eaux provenant des installations nucléaires) qui sont tous contrôlés avant rejets. Il n'est pas possible de détailler ici les caractéristiques chimiques de chacun des réservoirs analysés. On peut dire cependant que ces effluents respectent les normes suivantes :

pH	compris entre 6,5 et 8,5 dans le canal de rejet
Δ pH	inférieur à 0,5 unité dans la zone d'influence du rejet (entre S.E.M.O. et barrage Ampsin)
Cr^{6++}	inférieur à 50 ppb dans le rejet
Mn^{++}	inférieur à 100 ppb dans le rejet
Détergents	< 3 mg/l au rejet.

Caractéristiques Biologiques

Les résultats montrent que la croissance des algues n'est pas affectée par l'effluent radioactif, quelle que soit sa concentration. Par ailleurs, les algues cultivées dans l'eau distillée ont une taille plus grande que celles cultivées dans l'eau de Meuse. Plus on dilue l'effluent avec de l'eau de Meuse, plus la taille des algues diminue et se rapproche de celle des algues cultivées dans l'eau de Meuse.

On a aussi constaté qu'une série de radionucléides sont fixés dans les algues. Le nombre de ceux-ci varie d'un mois à l'autre suivant la qualité de l'effluent. Tous les radionucléides présents dans l'effluent ne se retrouvent pas dans les algues. Par contre certains radioéléments sont observés dans les

algues après culture, alors qu'ils n'ont pas été détectés en raison de leur faible concentration, dans l'effluent incorporé dans le milieu de culture. C'est le cas du Carbone-14 qui s'incorpore dans la matière sèche de l'algue, les teneurs observées étant du même ordre de grandeur que celles du tritium. En ce qui concerne ce dernier radioélément, des expériences répétées ont montré que si la source de tritium est de l'eau tritiée, le rapport de concentration entre l'activité de l'eau de combustion des algues et l'activité du milieu de culture est égal ou inférieur à 1. Au contraire, lorsque la source de tritium se présente sous forme organique tritiée assimilable, ce rapport est nettement plus élevé que 1. Les expériences effectuées avec les effluents de Tihange-1 ont montré que ceux-ci ne contiennent pas, ou très peu,de tritium organique. Par contre, on a décelé dans un échantillon d'eau de Meuse témoin la présence de molécules organiques tritiées provenant de sources situées en amont de la centrale nucléaire de Tihange.

Il est par ailleurs intéressant de noter que les paires de radioisotopes Co-58 et Co-60 d'une part et Cs-134 et Cs-137 d'autre part sont présentes avec les mêmes rapports de concentration dans les algues et les effluents.

RESUME ET CONCLUSIONS

L'analyse des échantillons mensuels moyens, résultant eux-mêmes d'un mélange de quatre ou cinq échantillons hebdomadaires effectués depuis la mise en service de la Centrale de Tihange-1 (1975) a montré que le Co-60, Cs-134 et Cs-137 sont les radionucléides critiques à surveiller, la voie d'atteinte critique d'exposition des individus de la population étant les sédiments du cours d'eau.

L'emploi de cultures d'algues Scenedesmus obliquus a permis de démontrer que les paires Co-58 / Co-60 et Cs-134 / Cs-137 sont respectivement biologiquement disponibles au même degré. La méthode de culture permet aussi la détection de composés tritium et carbone-14 dans les effluents.

La comparaison de la composition chimique des effluents de l'eau de la Meuse, et de l'eau de la nappe phréatique a montré que la modification résultant des corps chimiques présents dans les rejets d'effluents est très faible ($< 10^{-6}$), c'est à dire que l'impact de ces substances est non-significatif.

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PREOPERATIONAL STUDIES BY THE ENVIRONMENTAL MONITORING SYSTEM OF THE PAKS NUCLEAR POWER STATION

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Details are given of the preoperational investigations carried out by the continuous on line environmental monitoring system and mobile laboratory developed and installed for the Paks Nuclear Power Station.

Introduction

The telemetric stations of the continuous monitoring system have been developed for measuring gamma dose rates and the time integrals of ^{131}I concentration. In addition to the telemetric units, high volume aerosol and ^{131}I samplers are operated at the telemetric stations. The mobile laboratory is equipped with gamma-spectrometers and GM counters similar to those at the telemetric stations [1-5]

The minimum detectable dose rates and time integrals of the iodine activity concentration in air were calculated by different methods and compared with the authorized dose limit.

Measurements

Continuous background gamma dose rate measurements were carried out by the GM counters of the telemetric stations.

The temporal variations in background count rates at the different telemetric stations are in very good cross-correlation [6]. Both types of temporal variations /hourly and daily/ can be related to changes in weather parameters: i.e. to hourly changes in the temperature gradient and to daily changes in air pressure and precipitation.

The early morning positive temperature gradient increases the near-surface concentration of radon and its daughters.

The inversion effect, generally preceding precipitation, causes the same type of increase in the daily mean count rates on rainy days. Additionally, the decrease in air pressure enhances the intensity of the other component of background radiation, i.e. secondary cosmic rays.

The good cross-correlation in the temporal changes observed at the different telemetric stations enables the use of a correction method [6] for a more accurate determination of the actual background count rate at the downwind station. On applying this method the coefficient of variation for the actual background count rates can be reduced to 1.2-1.3 %, compared with that of 2.8-3.0 % for the measured values.

Results obtained by evaluating data of the continuous telemetric ^{131}I monitors can be summarized as follows:

- the temporal variations of count rates show only a weak correlation with changes in the level of radon and those of its daughters';
- the standard deviation of count rates is 3.5 % which hardly exceeds the statistical error.

The minimum detectable values derived from the data of one--year's operation are:

GM counter	3	nGy/h
GM counter /with Pb+Sn filter/	3.4	nGy/h
^{131}I monitor	1	Bqh/m^3
High volume ^{131}I sampler	0.03	mBq/m^3

As a part of the preoperational survey programme, in situ measurements were performed by a mobile laboratory equipped with a NaI/Tl/ spectrometer and a pair of GM counters. Since the first unit of the nuclear power station was put into operation the measurements have been continued, and the spectrometric method has been improved by the application of a Ge/Li/ detector [7].

We determined the dose rate contributions and the radioactive concentrations of the natural radioisotopes in the soil. The presence of ^{137}Cs originating from global fallout was detected in a few cases.

The mean, minimum and maximum values of the total terrestrial gamma dose rates are: 49.6 nGy/h, 31.8 nGy/h and 82.0 nGy/h, respectively. The overall uncertainty of the dose rates is about 15 %.

We found good agreement between the total average dose rate values measured by different methods. The differences do not exceed 25 % and arise from the uncertainties of the calculational model and the calibration used. A good correlation was also found between the dose rates obtained by the GM counter of the mobile laboratory and those of the telemetric stations.

The minimum detectable dose rates /nGy/h/ achievable by in situ measurement for several characteristic source distributions are as follows:

Detector	Natural emitters in the soil			^{137}Cs in the soil	^{131}I on the ground surface	^{133}Xe in plume
	^{40}K	^{232}Th series	^{238}U series			
GM counter	4.0	4.1	3.7	10.2	11.0	6.5
GM counter with filter/	5.1	6.1	5.7	14.0	17.5	715
NaI/Tl/	0.4	0.9	2.2	1.6	0.6	2
Ge/Li/	0.2	0.5	1.1	0.07	0.05	-

Conclusions

Hungarian regulations [8] require that the effective dose equivalent to members of the public living in the vicinity of the nuclear power station should not exceed 1/3 of the 5 mSv recommended by the ICRP. The radiation exposure due to activity released to the atmosphere per 1000 MW/e/ must not exceed 1/10 of the value above, namely 170 µSv/year [9].

The detection limit for the different units of the measuring stations are as follows:

	short term release / 1 h/	continuous release
GM counters	3 nGy_{air}/h	26 μGy_{air}/year
^{131}I monitors	3 $\mu Gy^{inhal.}_{thyr.}$/case	104 $\mu Gy^{inhal.}_{thyr.}$/year
High volume ^{131}I samplers	0.02 $\mu Gy^{inhal.}_{thyr.}$/case	1 $\mu Gy^{inhal.}_{thyr.}$/year

For continuous releases a detection limit of about 18 µSv/year was obtained when the effective dose equivalent was calculated by assuming a shielding factor of 0.7 for external radiation, and the weighting factor of 0.03 is applied for ^{131}I thyroid exposure /inhalation/. If the ^{131}I dose contribution by milk consumption is also taken into account a resultant /external + inhalation + ingestion/ effective dose equivalent detection limit of 26 µSv/year is obtained. /Here a 500 times higher effective dose equivalent is assumed due to ingestion than from inhalation for the same airborne ^{131}I concentration/

Dose calculations were carried out by the AIREM diffusion program [10]. Data actually measured at the meteorological tower during the first year of operation were used as input to the program. The results of the calculations led to the conclusion that the maximum dose occurring anywhere in the vicinity of the nuclear power station does not exceed by more than 5 % the highest dose measured at the telemetric stations. Approximately 70 % of the maximum value is expectable at the boundary of the 3 km radius safety zone whereas only 55 % of the maximum is expectable for the mostly exposed settlement.

Measurements made by the mobile laboratory complement those made by the telemetric system, especially in the case of the sensitive and specific determination of the fallout.

It is evident from the preoperational investigations that the environmental monitoring system of the Paks Nuclear Power Station is a suitable means of establishing that the exposure of the population does not exceed 1/5 of the value prescribed for the atmospheric release of a 1000 MW/e/ power unit.

References

[1] I. Fehér, S. Deme, A. Andrási: Environmental monitoring system at the Paks Nuclear Power Station. Acta Phys. Hung. 52, 3-4, p 373, 1982.

[2] S. Deme, I. Fehér, M. Rövid: Telemetric and data acquisition system for environmental monitoring system at the Paks Nuclear Power Station. Acta Phys. Hung. 52, 3-4, p 381, 1982.

[3] A. Andrási, P. Zombori: Gamma spectrometric measuring system for environmental sample analysis. Acta Phys. Hung. 52, 3-4, p 389, 1982.

[4] L. Koblinger, L. Németh, P. Zombori, A. Andrási: Comparison of field dose rate measurements and Monte Carlo calculations. Acta Phys. Hung. 52, 3-4, p 397, 1982.

[5] I. Fehér, S. Deme, A. Andrási: Application of the new limitation system in the radiation protection monitoring program of the nuclear power plant in Hungary. IAEA-SM-258/59, 1982.

[6] S. Deme, I. Fehér, E. Láng, J. Rónaky: GM-counters for continuous environmental monitoring at the Paks Nuclear Power Station. Proc. XIth. Regional Congress of IRPA, Vienna, 20-24 Sept. 1983.

[7] I. Németh, P. Zombori, L. Koblinger, A. Andrási, E. Germán, L. Kemenes: Investigation on the preoperational level of radiation field in the vicinity of the Paks Nuclear Power Station. Proc. XIth Regional Congress of IRPA, Vienna, 20-24 Sept. 1983.

[8] Regulation on health aspects of nuclear power plants No. 4. /1979. V. 29./ EüM, Magyar Közlöny 32 /1979/ 490 /in Hungarian/.

[9] Regulation on radionuclide atmospheric release limits for nuclear power plants No.1. /1980. II. 6./ OKTH, Magyar Közlöny 8 /1980/ 114 /in Hungarian/.

[10] Martin J.A. et al.: A computer code for calculating doses, population doses and ground depositions due to atmospheric emissions of radionuclides. Rep. EPA-520 /1974/.

ENVIRONMENTAL SURVEILLANCE FOR LOW-LEVEL WASTE DISPOSAL SITES

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INTRODUCTION

All nuclear facilities, including waste disposal sites, are expected to evaluate their effect on the environment. This paper describes a multidisciplinary approach to the design of environmental surveillance programs for near-surface disposal sites. Surveillance is broadly considered to include burial trench and other on-site measurements as well as offsite monitoring, and to include both radiological and non-radiological measurements.

The objectives of a surveillance program are: to determine the extent of any pollutant migration early enough for corrective action to be taken to minimize adverse impacts, to provide data to assess the impact of site operations on the environment and public, to predict long-term waste behavior, to satisfy regulatory agency requirements, and to obtain data on site and pollutant behavior for the design of future disposal sites and surveillance programs.

To meet these objectives, the surveillance program should include measurements of radioactive and chemically-toxic substances and of leachate indicators, measurements of direct penetrating radiation, determination of specific site and area properties, collection of specific information on the area, and interpretation of the results of the program in terms of health, environmental impact, standards, and expected performance.

These activities can be divided into three groups depending on their purpose: to measure the impact of the site on the public; to monitor site and burial trench performance; and measurements and data collections that support the first two activities. The first group of measurements should be made close to man; the second, close to the site and trenches. The third group includes observations and measurements needed for deciding on sampling locations and frequencies for the first two purposes, or are themselves indicators of site and trench performance.

DESIGN OF SURVEILLANCE PROGRAMS

An important starting point in program design is a pathway analysis that describes the routes of waste transport from burial site to man. Pathway analysis is a modeling exercise that analyzes compartmental transfer kinetics for a substance in the pathway. The major uncertainty in this process is that the compartmental transfer coefficients are highly site and substance specific and are not well known. However, even a qualitative analysis is very useful.

The waste composition and its chemical and physical properties are of particular importance, since it is the source term for the monitoring program and determines to a large degree the needed environmental measurements and the important pathways for pollutant transport. The number of radionuclides in the waste is limited by the definition of low-level waste and radioactive decay rates. The variety of chemical compounds is probably greater than the number of radionuclides. The pathways by which buried waste reach man are qualitatively the same whether the hazard of the migrating material is due to its radioactivity or chemical toxicity. Because of the variety of chemical and physical forms in which many radionuclides can exist, it is possible that the same radionuclide can migrate at different rates from the same trench. Also, the source term in a trench varies with time.

Information on a variety of properties and characteristics of the disposal site will be needed to design an adequate monitoring program and to evaluate the results of that program. Most of this information (e.g., meteorology, surface- and ground-water flow, geochemistry, land and water usage, and population distribution) will be obtained during the preoperational phase of a disposal site and when the site and surrounding area are being evaluated for suitability for waste disposal.

With the use of the information described above, a rationale for determining sampling and measurement locations and frequencies can be developed. The environmental media to be considered below are air, surface water, soil moisture in the unsaturated zone, water in trenches, ground water, surface soil, direct radiation, and plants and animals. Although the primary means of dispersal of buried waste are through air and water, biointrusion can be an important pathway in some locations.

For air and water, in which the direction and velocity of migrating waste can be estimated, the rationale is based on these basic principles: if the direction of a pollutant is from A toward B, a sampling station is placed along a line between A and B; the position of a station depends on the desired time for pollution detection and the speed of migration; and the time between sampling periods should be less than the travel time from A to the monitoring station. A different rationale is used for other cases, and statistical criteria for determining the number, as contrasted with the frequency, of samplings and measurements are also needed. Monitoring and sampling must also be conducted for comparison purposes at locations where the presence of migrating waste from the site is highly unlikely.

From measurements of wind velocity as a function of atmospheric stability, samplers can be placed where the average probability of detection is known. For a high total probability of detection, samplers are best placed in the buffer zone between the buried waste and site boundary. To detect releases before they leave the site boundary, an on-line device needs to be operated continuously downwind from an open trench. Such early detection is not possible for all contaminants. Sampling and measurement should be made for particulate contaminants, hydrogen-3, carbon-14, and radioiodines in several volatile chemical forms, and for some noble gases.

Surface streams and lakes may receive waste from runoff of surface water that has been in contact with a trench and from discharge of contaminated ground water. Visual observation and topographic maps will indicate the appropriate locations for sampling. It is necessary to sample streams during dry seasons when the base flow is not being diluted by direct precipitation and runoff.

If the unsaturated zone is large, 15 meters or greater, it is useful to measure changes in soil moisture content beneath or between trenches and to sample soil moisture with pressure-vacuum lysimeters. Flow rate and solute transport in the unsaturated zone should be estimated by a combination of field measurements and mathematical modeling. Water monitoring inside a burial trench is a special case of unsaturated zone monitoring, and the same principles apply. Sumps should be installed in each trench, and from the concentration of a pollutant in sump water, the rate at which it leaves the trench can be estimated and used as the source term. The ground-water system beneath and in the vicinity of a disposal site must be evaluated by the methods described for the unsaturated zone - measurements and modeling. From this, monitoring lines from a group of trenches in the directions of flow can be established and monitoring wells placed along them.

Surface soil collected on and near the site, with locations concentrated in downwind directions, is needed to detect any pollutants deposited from airborne contamination, accidental spillage, evaporation of waterborne contamination, and resuspension and subsequent deposition. Two types of vegetation should be sampled: plants growing on completed trenches and plants used as food for man and grazing animals in areas that could become contaminated. Sampling of two types of animals should be conducted: animals that can burrow into a trench; and those that graze on land or consume water that can become contaminated or live in rivers and lakes that can become contaminated.

Direct radiation from buried waste should be measured near trenches, at the site perimeter, and at offsite locations for a complete evaluation of this source of radiation exposure.

STATISTICAL CONSIDERATIONS

Statistical criteria should be used to assist in deciding on the numbers of samples and measurements, and in deciding when small, but real, changes from background or previous levels have been detected. The application of statistics should begin in the preoperational phase. Sufficient sample collections and measurements should be made for all important media and pre-existing potential contaminants to obtain an observed mean and standard deviation of sufficient reliability to allow the use of established statistical tests. From the central limit theorem, this number is about 30. Subsequently, the number can be reduced, based on experience. Since releases from a burial site can be directional, individual results should be compared with preoperational and offsite means to judge the statistical significance of small changes.

MONITORING PROGRAMS

Programs for the three phases (preoperational, operational, post-closure) of the life cycle of a waste disposal facility can be developed from the above discussions. Suitable sampling and analysis programs will be given in the poster session; space does not allow giving such details here. Reliance should be placed on high-sensitivity gamma-ray spectrometry with large germanium detectors for gamma-ray emitting radionuclides. However, some of the most mobile long-lived radionuclides are pure low-energy beta emitters, for which radiochemical analyses are needed, and if the limits of transuranic elements in low-level waste are increased, their measurement should be included. The most important radionuclides that cannot readily be detected at low concentrations by gamma-ray spectrometry are hydrogen-3 (as water vapor and simple organic compounds), carbon-14 (as carbonate and simple organic compounds), nickel-63, krypton-85, strontium-90, technetium-99, iodine-129, radon-222, radium-226, uranium, and transuranic nuclides.

Of equal importance to radionuclide analyses, particularly in ground-water monitoring, are measurements of chemical indicators of waste migration to identify leachate movement. These measurements include pH, electrical conductivity, chloride and nitrate ions, chemical oxygen demand, organic carbon, and complexing agents for metallic ions. Chemically-toxic compounds and pathogens that may be associated with the buried waste should also be measured. Sample collection, preservation, and measurement procedures have not been addressed due to lack of space, but adequate consideration of these aspects are essential to the success of a monitoring program.

The details of postoperational monitoring will be determined principally by the operational monitoring results. The important migration and exposure pathways should be well known, and the number of radionuclides and chemical pollutants of concern should be fewer because of radioactive decay and non-detection during operational monitoring.

SUMMARY AND CONCLUSIONS

It should be recognized that shallow-land burial techniques will not retain the waste completely in place over time. Some migration is expected, but it must be acceptable in amount and rate. Surveillance programs will provide the information to determine if this is the case. Proper application of monitoring techniques and principles will result in a program that will meet its objectives and supply the data needed by operators and regulators to maintain a properly operating site.

REFERENCE

Environmental Monitoring for Low-Level Waste-Disposal Sites, Low-Level Radioactive Waste Management Handbook Series, Report DOE/LLW-13Tg, U. S. Department of Energy, EG&G Idaho (January, 1983).

TRITIUM MONITORING AT THE TRITIUM SYSTEMS TEST ASSEMBLY*

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The Tritium Systems Test Assembly (TSTA) is a computer-controlled facility recently constructed at the Los Alamos National Laboratory in Los Alamos, New Mexico, to mock up full scale the fuel cycle of future fusion reactors. The TSTA fuel loop consists of a mock torus vacuum vessel, a vacuum cryopumping system, fuel cleanup and hydrogen isotope separation systems, and associated storage, delivery and transfer pumping systems. The maximum flow rate in the fuel loop is 360 moles of DT per day, comparable to the flow rates expected at fusion power reactors. To achieve this rate requires an inventory of approximately 1.5×10^6 Ci of tritium gas.

The major goals at TSTA are to demonstrate the effectiveness and reliability of its major subsystems in continuous operation and to show that this can be done with a high degree of safety. This safe operation is attained by a combination of containment and monitoring in depth, and the use of two tritium removal systems: an on-line Tritium Waste Treatment system (TWT) and an Emergency Tritium Cleanup system (ETC) for processing room air. All components that contain tritium are secondarily enclosed as are interconnecting piping between enclosures. Moreover, all of the tritium at TSTA is contained within the 3000-m^3 main experimental room, which provides a form of tertiary containment. In case of a significant release of tritium into the room, valves in the room ventilation ducts immediately isolate the room atmosphere, which is then processed by the ETC. Safety is further assured by the use of two main control computers, two back-up safety computers, an emergency generator, and an uninterruptable battery power supply which supplies power directly to the computers and the tritium monitors.

THE TSTA TRITIUM MONITORING SYSTEM

Critical to the control of any tritium released from the primary containment of the fuel loop is the tritium monitoring system. This system consists of stack and duct monitors, room monitors, and glovebox monitors. For monitoring the performance of the TSTA subsystems, there are also process monitors located in the fuel loop, in the tritium removal systems, and in the on-line gas chromatographs. The emphasis in this discussion is on instruments used to monitor for tritium released from the primary containment.

Table 1 lists the types of monitors being installed at TSTA and the numbers of each type, broken down according to their principal use or function. With two exceptions--the stack bubbler and the ETC water monitor--all are ionization-chamber instruments using the Kanne design with the outer chamber at ground potential and the inner chamber at high voltage. Sampled gas enters the space between

*Work performed under the auspices of the United States Department of Energy.

TABLE 1

TRITIUM MONITORS USED TO MONITOR RELEASES WITHIN THE TSTA EXPERIMENTAL ROOM

Number	Function	Location or Type	Range
3	Stack	Bubbler (1)	1 mCi-10^6 $\mu Ci/m^3$
		Low-Range (1)	1-10^4 $\mu Ci/m^3$
		High-Range (1)	10^{-3}-10^3 Ci/m^3
2	Exhaust Duct	Main Duct (1)	1-10^4 $\mu Ci/m^3$
		Auxiliary Duct (1)	1-10^4 $\mu Ci/m^3$
10	Room	Low-Range (8)	1-10^7 $\mu Ci/m^3$
		High-Range (2)	10^{-2}-10^3 Ci/m^3
14	Secondary Enclosures	Gloveboxes (13) Vacuum System Enclosure (1)	1-2x10^6 $\mu Ci/m^3$

the two chambers, which serves as a built-in ion trap, before entering the inner, measuring chamber.

Glovebox Monitoring

Including the process monitors, there are 50 instruments at TSTA, many of which incorporate innovations developed at Los Alamos. Figure 1 is a photograph of open-walled ionization chambers designed primarily for use in the gloveboxes. The response of these chambers is for all practical purposes identical to that of conventional flow-through chambers. Their advantages are simplicity of design and the elimination of a pumping system to introduce air into the chamber, usually the primary source of instrument failure. In use, the chamber is suspended from the roof of the glovebox and the small interface chassis is located outside of the glovebox itself. Mention should be made of the gamma check-source unit shown in the photograph. All of the low-range ion-chamber instruments incorporate a Ba-133 gamma check source normally stored in a small lead shield next to the chamber. Once a day, on command from the computer, a solenoid raises the source to a window in the shield and exposes the chamber. If the resultant chamber signal is not within prescribed limits, the operator is notified.

Electronics for the glovebox monitors consist of commercial electrometers and Los Alamos-built units housing the chamber power supply and two adjustable alarm circuits. The low alarm level ($\sim$20 $\mu Ci/m^3$), set comfortably above background, alerts the TSTA operator of an increase in glovebox concentration. The higher alarm ($\sim 10^3$ $\mu Ci/m^3$) automatically begins purging the glovebox atmosphere to the TWT.

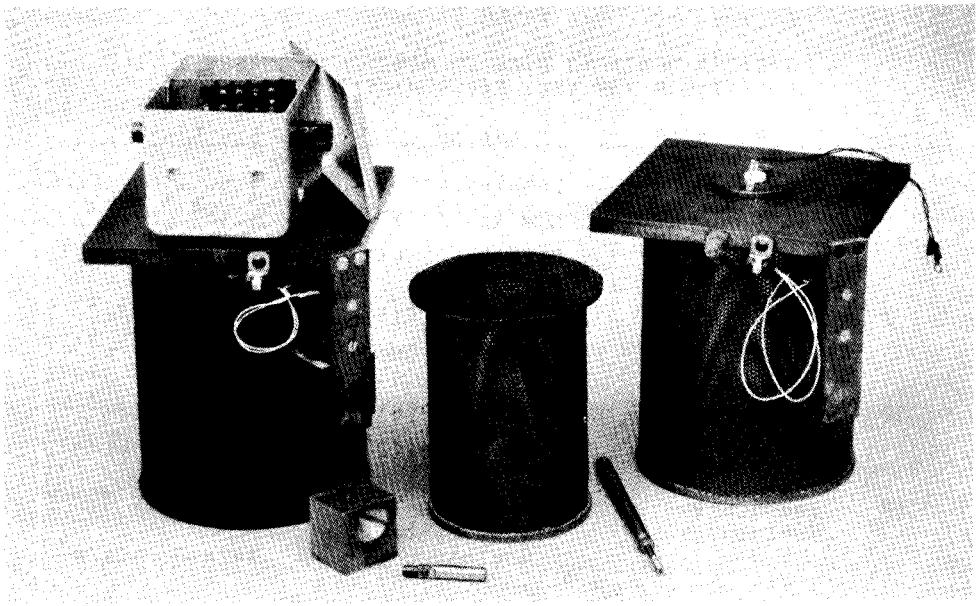

Fig. 1. TSTA-designed, open-walled ionization chambers used for monitoring the gloveboxes. The assembled chamber on the left is covered with a cloth dust cover. The appendages are the gamma check-source units for automatically checking the performance of the instruments (see text).

Besides local alarms on the tritium monitoring instruments, visual/audible alarms are also provided to the TSTA operator, who can then quickly determine the cause of the alarm before taking appropriate action.

Room Monitoring

Room monitoring is accomplished mainly by instruments commercially-built to TSTA specifications. The range of the eight principal room monitors covers 7 decades and incorporate three alarm circuits (tripping at $\sim$10 $\mu Ci/m^3$, 10^2 $\mu Ci/m^3$, and 10^4 $\mu Ci/m^3$). These instruments employ flow-through chambers to enable each instrument to monitor several points simultaneously. By means of pressure switches in the sampling lines, the computer monitors their flow rates as it does for the three stack monitors, which also employ pumps.

Two high-range room monitors employing commercial picoammeters and 0.1-ℓ open-walled ionization chambers cover the concentrations that would be realized should gram quantities of tritium be released into the experimental room.

Stack and Duct Monitoring

The stack is monitored by three instruments (Table 1). The glycol bubbler operates at about 80 mℓ/min and incorporates a flow controller and a catalyst for differentially monitoring the oxidized and elemental forms of tritium. The main stack monitor uses a 10-ℓ flow-through chamber and Los Alamos-built electronics. Two adjustable alarm levels, presently set at ∿10 and 10^4 μCi/m^3, are provided. This instrument has integrating capability with its own alarm, set at 10 Ci. The computer rezeroes the output each day. With the range of the instrument only 10^4 μCi/m^3, a high range tritium monitor using a flow-through 0.1ℓ chamber and a commercial 6-decade logarithmic picoammeter is also used to monitor the stack. Duct monitoring is accomplished with open-walled chambers placed directly into the (two) exhaust ducts being monitored. The electronics are identical to those of the main stack monitor.

Besides simply monitoring room releases, the main stack and both duct monitors along with the main room monitors are critical to the prevention of major releases to the environment. Should any one of these monitors alarm at its high trip level (∿10^4 μCi/m^3), the building evacuation alarm is sounded and the room air is isolated by ventilation duct valves, independent of the operation of the computers. In addition, if any two such monitors alarm at this level, the ETC is automatically turned on. The duct and stack monitors, having integrating capability, will also isolate the cell if the tritium released to the environment within a 24-hour period exceeds the preset value.

Process Monitoring

Process monitoring is accomplished by commercial electrometers and Los Alamos-built interfaces coupled to stainless-steel chambers whose size varies from 20 mℓ for the gas chromatographs to 1.5ℓ for the tritium removal systems.

In the case of some of the process or glovebox monitors, there may be significant memory effects for chambers exposed to a wide range of concentrations. Should this become a chronic problem, the chambers will be gold plated or redesigned with grid walls that minimize the surface area of the measuring volume.

All of the instruments mentioned above are monitored by the main computers, which also provide redundant alarms to back up the instruments' own alarm circuits. Another safety feature incorporated in most of the tritium monitors is the continuous monitoring of the instruments' high and low voltage power supplies by the computer. For the instruments important to safety, there are also provided hard-wired readouts in the control room and in an auxiliary Support Center.

The prevention of significant tritium releases to the environment is an important goal in itself. It is all the more important for an energy program that wishes to demonstrate that the fuel it must use can be safely handled and contained. At TSTA we have attempted to make the tritium monitoring system, which is critical to this control, as reliable and effective as we could reasonably make it.

NADAM, DAS SCHWEIZERISCHE NETZ ZUR AUTOMATISCHEN UEBERWACHUNG DER UMGEBUNGSSTRAHLUNG: VERSUCHSBETRIEB UND ERSTE ERGEBNISSE

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Eidg. Kommission zur Ueberwachung der Radioaktivität, Fribourg

ABSTRACT

We started the operation of the NADAM-network in summer 1982 with eight pilot-stations. The final network consists of 51 dose-rate sensors connected to the automatic stations of the Swiss meteorological network ANETZ. The local dose-rate values are transmitted every 10 minutes, together with several meteorological parameters, to a central computer through a fixed telecommunication network (leased voice-grade telephone lines). The central gathering and handling of the data in real time are described. We explain further the quality check and correction of the data, the issue of messages, as well as the emergency processing in case of warning induced by dose-rate increase. More than one year of continuous NADAM data collection allows to assess the environmental dose for different time-scales and various meteorological parameters, e.g. precipitation, soil state, etc.

DAS KONZEPT

Für den Fall einer Gefährdung infolge erhöhter Radioaktivität muss der Alarmausschuss Radioaktivität (AA) das Ansteigen der Ortsdosisleistung kontinuierlich und grossräumig verfolgen können /1/. Mit dem Netz für Automatische Dosis-Alarmierung und -Messung, NADAM /2/, befindet sich ein dauernd messendes Ueberwachungssystem im Aufbau, das

- die Ortsdosisleistung vom natürlichen Pegel bis zu ca 5 R/h auf dem Gebiet der Schweiz überwacht,
- bei Ueberschreiten einer wählbaren Alarmschwelle die Ueberwachungszentrale (UWZ) unverzüglich warnt und ihr erlaubt, im Fall einer

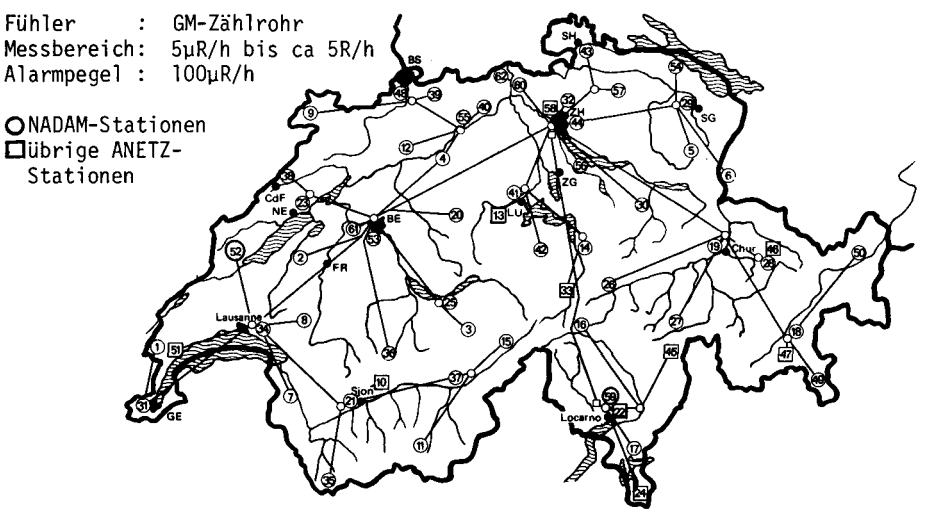

Fig. 1: Das Netz für Automatische Dosis-Alarmierung und -Messung, NADAM

radioaktiven Kontamination diese automatisch zu verfolgen,

- im Normalfall eine räumliche und zeitliche statistische Erfassung der natürlichen Strahlung sowie deren Korrelation mit meteorologischen Einflussgrössen ermöglicht.

NADAM basiert weitgehend auf der Infrastruktur des automatischen Messnetzes ANETZ /3/ der Schweizerischen Meteorlogischen Anstalt (SMA). Dieses besteht aus 64 automatischen Stationen (ASTA), die über ein Netz gemieteter Telefonleitungen mit dem Rechner der ANETZ-Zentrale (AZEN) verbunden sind. An 51 ASTA's wird eine NADAM-Sonde angeschlossen (Fig. 1), welche die Ortsdosis mit einem GM-Zählrohr erfasst.

In einer ersten Phase wurden Mitte 1982 acht Stationen für einen Pilotversuch ausgerüstet, der nach einem Jahr erfolgreich abgeschlossen werden konnte. In der Folge wurde mit dem sukzessiven Vollausbau des NADAM-Netzes begonnen, dessen Abschluss auf Ende 1985 geplant ist.

ZENTRALE DATENERFASSUNG UND AUSWERTUNG

Von der AZEN werden im 10-Minutentakt alle ASTA's abgefragt. Diese melden einmal stündlich die Daten des ganzen Messsatzes, zu den Zwischenzeiten nur die Werte von rasch ändernden Messgrössen, wie z. Bsp. Wind, Lufttemperatur, Niederschlag, Sonnenschein, Ortsdosisleistung (in Imp/10 Min.) usw. Die erfassten Daten werden laufend auf einem Datensicherungsband abgespeichert.

Die NADAM-Messwerte werden in der AZEN mit einem Polynom vierten Grades linearisiert und von Imp/10-Min. in Dosisleistung umgerechnet. Die Darstellung aller Messwerte in der AZEN erfolgt mit vier Stellen, was für die meteorologischen Parameter auch vollauf genügt. Für den Messbereich der NADAM-Sonde (5µR/h bis 5R/h) musste demzufolge eine geeignete Darstellungsform gefunden werden: Wir wählten die in diesem Bereich wohl unübliche logarithmische Form von millibel bezüglich lnR/h:

$$\text{millibel} \triangleq 1000 \log_{10} D$$

Bei jedem Abfragezyklus werden die NADAM-Messwerte auch auf Ueberschreiten des festgesetzten Alarmpegels von 0.1mR/h überprüft; gegebenenfalls wird eine Alarmmeldung im Telexformat (siehe Fig. 2a) erzeugt. Stündlich werden meteorologische Angaben zusammen mit der Ortsdosisleistung in ein Fernschreib-Bulletin (Fig. 2b) umgesetzt. Alarmmeldungen und Bulletins werden dem Rechensystem METEOR der SMA übermittelt, von dem sie den entsprechenden Adressaten, wie z. Bsp. der UWZ weitergeleitet werden.

Einmal täglich wird das Datensicherungsband auf METEOR überspielt. Die Messwerte von jedem 10-Minutenintervall werden in der "Zehnminutendatei" abgelegt; der Messwert der vollen Stunde und die aus den Zehnminutenwerten abgeleiteten Werte (Stundenmittel, -minimum und -maximum) werden in der "Stundendatei" abgelegt. Der Inhalt der Stundendatei wird täglich einer automatischen Qualitätskontrolle unterzogen und auf die Kriterien

- Vollständigkeit der Daten
- Ueberschreiten eines stationsspezifischen Grenzwertes
- Variabilität der Extremwerte des Stundenintervalls
- Variabilität des Mittelwertes von zwei aufeinanderfolgenden Intervallen

überprüft. Verdächtige Werte werden an einem Bildschirm-Arbeitsplatz mit Hilfe von meteorologischen Parametern auf ihre Plausibilität überprüft und gegebenenfalls korrigiert; ebenfalls werden Lücken in der Datei unter Berücksichtigung der jeweils herrschenden meteorologischen Situation durch Interpolation geschlossen. Am Ende eines Monats wird aus der Stundendatei der Monatsmittelwert berechnet und zusammen mit dem datierten Minimum und Maximum in der "Monatsdatei" abgelegt.

a) Alarmmeldung:

```
zczc ks 241210          Meldungskopf
pay 5890 5'279 2210 /026  Monatstag und GMT-Zeit (ddhhmm)
                          Niederschlagssumme (0.1mm), letzte 10 Min.
                          Wind (Dekagrad, m/s)
                          Dosisleistung (Exponent, 0.1µR/h)
                          Stationsbezeichnung
```

b) Stündliches Bulletin:

```
 station   ttt ttn ttx uuuu  ppppp wwww wwww vvnm  oogp de//  Lrrr
ber 5520 +088+085+092 0895 09603 //// //// //// //// //// 4110=
fre 6185 +124+118+133 0443 ///// //// //// //// //// //// 4211=
dol 8280 +091+087+096 0579 08407 //// //// //// //// //// 4136=
neu 6340 +153+146+157 0502 09700 //// //// //// //// //// 4075=
pay 5890 +094+085+100 0855 09689 //// //// 7500 0200 //// 4030=
puy 8100 +132+127+139 0686 09717 //// //// //// //// //// 4110=
aig 7970 +116+112+121 0929 ///// //// //// //// //// //// 4130=
mls 5620 +130+124+135 0398 08127 //// //// //// //// //// 3989=
```

meterologische Grössen — Dosisleistung in millibel

Fig. 2: Die Fernschreibmeldungen der AZEN

EINFLUSS DER METEOROLOGISCHEN BEDINGUNGEN AUF DIE DOSIS

Die während mehr als einem Jahr im Rahmen des Pilotversuches gesammelten Daten lassen den Einfluss verschiedener meteorologischer Parameter auf die Ortsdosis erkennen. Die bedeutendsten kurzzeitigen Schwankungen der Ortsdosisleistung werden durch Niederschläge erzeugt. Die grösste bisher registrierte Erhöhung des 10-Minutenwertes von +132% wurde während eines heftigen Gewitters am 19. August 1983 für die Station Payerne beobachtet. Innerhalb einer Stunde fielen mehr als 42 mm Regen (10-Min.-Maximum: 12.5mm).

Es ist bekannt, dass durch Niederschläge Aerosole aus der Luft ausgewaschen und auf der Erdoberfläche angesammelt werden. Nun suchten wir nach einer Korrelation zwischen Niederschlagssumme und der integralen Dosiserhöhung. Um Auswascheffekte

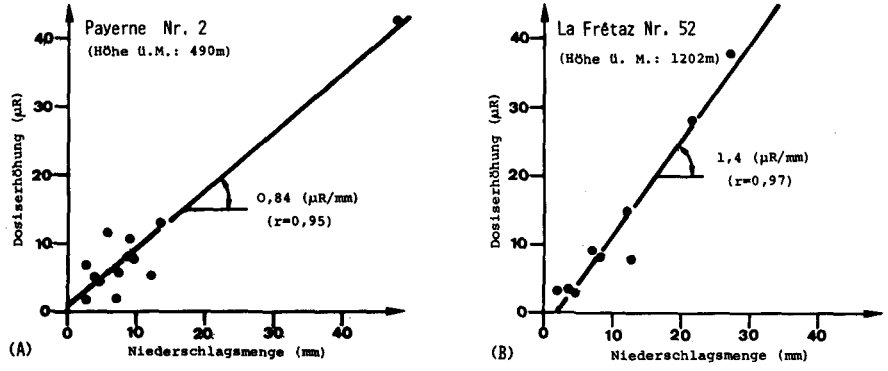

Fig. 3: Dosiserhöhung, ausgelöst durch Regenschauer (Dauer weniger als 4 Std., nach einer Trockenperiode von wenigstens 24 Std.), in Abhängigkeit von der Niederschlagsmenge pro Regenschauer für die Stationen Payerne (A) und La Frêtaz (B); r: Korrelationskoeffizient.

abzutrennen, beschränkten wir uns auf Regenperioden mit einer Höchstdauer von vier Stunden und mehr als 0.5mm/h Niederschlag, die auf niederschlagsfreie Perioden von mindestens 24 Stunden folgten. Da die Dosisleistung noch während 2 Stunden über das Ende der Niederschläge hinaus erhöht bleibt, berücksichtigten wir die Dosiserhöhung während der Regenperiode und der anschliessenden zwei Stunden. Als Referenzwert diente das Mittel der Dosisleistung über die fünf der Regenperiode vorangehenden Stunden.

Die Figur 3 zeigt die Dosiserhöhung in Abhängigkeit von Niederschlagsmengen für zwei Stationen: Payerne (A) und La Frêtaz (B). Es zeigt sich eine deutliche Korrelation zwischen der Dosiserhöhung und der Niederschlagsmenge: Der Korrelationskoeffizient beträgt 0.95, bzw. 0.97. Die Steigung der Regressionsgeraden ist interessanterweise für die Station La Frêtaz (1.4µR/mm) ungefähr 1.5 mal so gross wie für die Station Payerne (0.84µR/mm). Dieser Effekt muss noch näher untersucht werden.

Es wurde auch beobachtet, dass der Tagesmittelwert der Dosisleistung von der Bodentemperatur abhängt (Messung in 5cm Tiefe). Um diesen Zusammenhang hervorzuheben, haben wir nur die niederschlagsfreien Tage bei trockener Bodenoberfläche betrachtet. Die Resultate für die Station Payerne für die Zeit von März bis Juli 1983 sind in Fig. 4 dargestellt. Diese zeigt, dass die mittlere Dosisleistung für Temperaturen unterhalb 20°C einigermassen konstant bleibt und oberhalb deutlich ansteigt. Dieses Verhalten ist wahrscheinlich mit dem Wassergehalt des Bodens verknüpft. Das konnte aber nicht abgeklärt werden, da der Wassergehalt des Bodens bei den meteorologischen Beobachtungen nicht berücksichtigt wird.

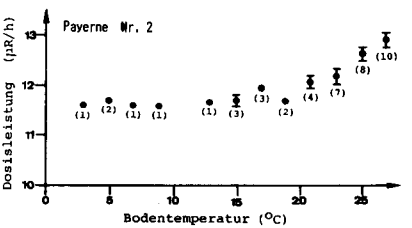

Fig. 4: Mittlere Dosisleistung in Abhängigkeit vom Tagesmittelwert der Bodentemperatur (niederschlagsfreie Tage bei trokkener Bodenoberfläche, Klassen von 2°C Breite, ( ): Anzahl der Tage pro Klasse).

Wir möchten an dieser Stelle der Schweizerischen Meteorologischen Anstalt für ihr Entgegenkommen und ihre Mitarbeit herzlich danken.

Referenzen:

/1/ O. Huber, P. Honegger, S. Prêtre, M. Baggenstos: Schutz der Bevölkerung bei einer Gefährdung durch Radioaktivität.
Sekretariat Alarmausschuss Radioaktivität, c/o SMA Zürich, 1982

/2/ L. Ribordy, B. Michaud, P. Honegger, O. Huber, F. Wicht: NADAM, ein landesweites Netz für die automatische Fernmessung der Ortsdosisleistung in der Schweiz.
Bericht der 16. Jahrestagung für Strahlenschutz e.V., FS-83-30-T/GSF Bericht A4/83, p. 353, München, 1982

/3/ Projekt ANETZ 1974-1980, Automatisches Meteorologisches Mess- und Beobachtungsnetz; Zusammenfassung der Vorträge vom ANETZ-Kolloquium vom 26. und 27. November 1980.
Schweizerische Meteorologische Anstalt Zürich.

A CONTINUOUS ON-LINE EMERGENCY AND ENVIRONMENTAL MONITORING SYSTEM

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1. Introduction

Following the Three Mile Island accident, Winfrith, in common with many other reactor sites, reviewed its arrangements for dealing with accidental off-site releases of radioactivity. It was felt that improvements could be made by installing a continuous monitoring system aimed at rapidly informing the site emergency control staff that an off-site release had in fact occurred, the probable direction and duration of the release and estimating its magnitude. At the time of this review versions of large atmospheric dispersion modelling codes were being developed for Minicomputer systems (Ref 1). In 1980 it was decided to install a system of remote monitoring stations around the Winfrith main reactor complex. These stations would continuously transmit data to a minicomputer controlled central data logger. The minicomputer would also run atmospheric dispersion codes to provide rapid predictions of probable consequences of the release on local populations. Such predictions would be modified by the data recorded. In non emergency use the system would provide environmental monitoring data to back up existing passive devices.

2. The Site

One of Winfrith's tasks is developing prototype reactor systems for the UK Civil Nuclear Power Programme. The largest such operating reactor system on-site at present is the 100 MW(E) Steam Generating Heavy Water Reactor (SGHWR). It was assessed that, if there was a major release of radioactivity from the Winfrith site, however unlikely, this would originate from the SGHWR. The initial locations of the monitoring stations were therefore evenly spaced around this facility. The SGHWR site is near the Winfrith site boundary and, because of restrictions on available land i.e. undeveloped land under Winfrith control, the spacing of the monitoring stations is closer to the reactor than that considered ideal. They are spaced on a 500 m radius (16 at 22½° spacing). The actual locations, accurately known, vary both in angular and radial spacing because of the actual land contours. Ten of the stations are outside the perimeter fence on land designated as being of special scientific interest because of the local flora and fauna. Consequently the positioning of the stations, routes and frequency of access and their environmental impact are carefully selected and controlled. For this reason, together with technical and economic factors, radiotelemetry transmission of data was selected rather than direct cable transmission. The central

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receiving station is on the main site, 2000 m from the reactor. Although the concept for the outstations is for known, semi-permanent positions, the system is designed such that any station may be moved with a minimum of effort.

3. The System

The system consists of 16 mobile outstations, communicating by a single frequency (approximately 450 MHz), simplex channel, to the base station.

3.1 The Outstations

Each outstation consists of a locked environmental box containing three calibrated geiger detectors, twin battery packs, EHT conversion circuits, a microprocessor and the radio equipment. The box supports a single dipole aerial. The box construction - a 0.5 m cube - is of light gauge steel on a frame work which is bolted to a concrete plinth. This plinth, anchored to the ground, provides shielding for one of the geiger tubes and also provides stability to the structure and protection against vandalism and theft. The station may be moved by removing the anchoring bolts or by lifting it off the plinth. The box, painted to merge in with the surrounding heathland, supports a 1.75 m pole and a 0.65 m dipole aerial. The box and all components are weatherproofed, ventillated and operate satisfactorily over a temperature range of -20°C and 40°C. The use of a simple dipole aerial lessens the environmental impact.

Two sealed 12 v, 25 amp hr lead acid batteries provide all the power to each of the outstations. Long battery life and low power consumption were considered essential and battery lifes in excess of 3 months have been obtained for normal environmental dose rates. The EHT units, sealed modified commercial units, were selected for low power consumption.

Two of the geigers, type MX 163, are intended for high dose rate use with a dynamic range of 0.001 to 100 R/hr. One of them is shielded, by the concrete plinth, from the reactor site, thus allowing corrections to be applied to the monitored plume activity for direct radiation shine from the reactor and for ground deposited activity. The third gieger, type MC70 with a dynamic range of 0.005 to 30 mR/hr, is intended for environmental background studies and is switched off above dose rate of 20 mR/hr. The communication control logic circuits were designed and interfaced to an 8 bit CMOS Microcomputer at Winfrith. Pulse outputs from the geiger are stored in temporary registers; these counts are then transferred to the microcomputer store. Upon reaching a defined level, the microcomputer enables the radio receiver and awaits its station identity polling signal from the base station. Its reply, in ASC11 code, consists of the station identity, the battery voltage state, the count from each geiger register and the check sum. After transmission the receiver is re-energised for 100 mseconds awaiting a re-poll from the base station. Re-polling is done if the check sum validity check fails and can be repeated up to three times.

3.2 The base station

The base station consists of a radio receiver and transmitter. Power levels are restricted by the operating licence. A microcomputer links the radio system to a Minicomputer. The power to the system is backed up by diesel generated emergency supplies. The base station polls the outstations at a frequency of 1 station every 5 seconds. Under normal environmental background conditions each outstation will be read every forty minutes. Under high radiation dose conditions, data updates will be as frequent as once per 5 seconds when selective station polling is installed. Validity checks of the register contents and the protocol of data transfer are controlled by the minicomputer which stores all the information.

4. The Atmospheric Dispersion Programme

The Minicomputer, a standard PDP 11/34 (256 Kbytes) with three 5 Mbyte discs, operates under RSX-11M with a Fortran compiler. It has twin functions - acting as a real time data logger from the outstations and as an interactive atmospheric dispersion modelling system.

The gamma monitors in the outstations provide information about the radioactivity passing over them and it is necessary to apply an atmospheric dispersion model in order to estimate effects further downwind. The dispersion codes, under development at SRD Culcheth are therefore incorporated (Ref 1). The modelling codes require information on the weather conditions and the source term. The weather conditions are simple observations and predictions on wind speed, wind direction, cloud cover and time of day and year. The source term is user defined from a library of predicted accident scenarios or can be built up from data on single nuclide releases. The modelling is a simple two dimensional Gaussian diffusion approach, similar to that used in the TIRION code (Ref 2) although that code uses statistical weather for risk assessment purposes.

4.1 Output

The output from the modelling codes is displayed as contours of dose to an individual overlaid on a map of the location. Local population densities are also stored to enable the site emergency controller to act on these predictions. A choice of dose effect may be displayed. Currently they are:-

(a) Effective committed dose equivalent (50 years).
(b) Whole body dose from ground deposited activity to one day.
(c) Lung dose from inhalation to one year.
(d) Thyroid dose to 50 years.
(e) Whole body dose from gamma emitters in the cloud.
(f) Acute bone marrow dose.

At present output is via a typewriter or VDU.

5. Experience with the System

The dispersion software has been operating since 1981 and has been used in the last two major site exercises, using dummy inputs, to provide experience to Health Physics staff and the site control team. The outstations and radiosystem were delayed - delivery being staged throughout 1983. Considerable experience has been gained on the early stations in proving the reliability of the system under a range of weather conditions. The transmission reliability is better than 99.9%; the battery life time is longer than 3 months and the geiger characteristics have not changed over the lifetime of the equipment to date. As well as accumulating environmental background data, the stations have been moved into active environments and have been used to monitor pipeline transit of active material into sludge settling tanks.

6. Future Developments

Developments in an advanced stage of commissioning are those for the installation of remote colour graphic terminals. Colour graphics was considered to be the optimum route to achieve rapid assessment of information available from the system. In order to provide multi-user access and to relieve the load on the minicomputer these terminals are "intelligent", data being refreshed from the minicomputer on a system priority basis. Code developments on the atmospheric dispersion models will include washout and local topography routines.

Existing emergency monitoring equipment includes passive TLD dosimeters around and remote from the Winfrith site. Such devices are also used for environmental purposes. Portable high resolution gamma spectrometers with standard RS 232 interfaces are available for rapid isotopic assessment of ground deposited activities. Such information has yet to be interfaced to the computer. Code development to enable source term modifications based on such information to be made is also a task for the future.

Local meteorology data is available and is now in a digital form suitable for computer application. The possibilities of modifying the outstations to provide wind information direct to the central system is also being considered.

References

1. G C Meggitt & R G Jackson. Seminar on Emergency Preparedness: Realtime Diffusion Models. Rome 1983.

2. Fryer L S & Kaiser G D. TIRION 4 A Computer Programme for use in Nuclear Safety Studies, UKAEA Report - SRD R 134 (1978).

Acknowledgements

Many people have contributed to this project. We would like to thank specifically R T Watts, G C Meggitt and F J Linden.

CARBON-14 SAMPLING AND MEASUREMENT IN GASEOUS RELEASES FROM THE ATUCHA I NUCLEAR POWER PLANT

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INTRODUCTION

Several reports (1-4) have discussed the production and release rates of carbon-14 at different types of nuclear power plants in the light of the growing interest in the radiological consequences of eventual discharges into the environment of this radionuclide.

According to those previous papers, the major contribution to carbon 14 production in HWPR' is the $^{17}O(n,\alpha)^{14}C$ reaction in the D_2O moderator, with predominant hydrocarbon formation due to the reducing conditions in the primary water.

However, only little information is available about measured rates for carbon-14 emission at heavy water moderated reactors (3,6).

This paper presents the preliminary results obtained while carrying out a measurement program for carbon-14 compounds in gaseous effluents at the Atucha I nuclear plant.

As the sampling method involves the retention of $^{14}CO_2$, it includes catalytic combustion of the reduced forms of carbon-14 compounds, such as CO and methane. Carbon-14 is measured by liquid scintillation counting.

ATUCHA I NUCLEAR POWER PLANT

Atucha I nuclear power plant, in operation since 1974, is equipped with a pressure vessel reactor. It uses natural uranium as a fuel and heavy water as a coolant and moderator.

The reactor thermal output is 1,179 MW, with a net electrical output of 345 MW.

The main production of carbon-14 occurs in the moderator coolant system, in forms such as CO_2, CO, and hydrocarbons and may be released to the environment after passing through the waste gas treatment system.

Stack releases include building ventilation, decay tanks and containment discharges, with an air throughput of about 150,000 m^3/h.

SAMPLING AND MEASUREMENT METHOD FOR CARBON-14 COMPOUNDS

Sampling and measurement techniques for carbon-14 compounds have been described previously by several authors (1-5).

The stack vent air of the Atucha I NPP is pumped from an isokinetic sampler through the analysis apparatus at a rate of about 0.2 l/min.

After aerosol filtering, the effluent gas is passed through wash bottles with fine-pore frits filled with 300 ml of 2M NaOH, where CO_2 is removed. The remaining gas, containing reduced forms of carbon-14, is oxidized catalytically by passing through a tubular furnace at 800°C, with CuO as a catalyzer.

The CO_2 formed is separated in further gas-washing bottles.

Combustion yield is determined by oxidation of air mixed methane gas. The sorbed CO_2 is precipitated as $BaCO_3$ by adding a 0.5 M $BaCl_2$ solution after heating.

The precipitate is filtered and dried at 120°C until constant weight is attained. From the weight of $BaCO_3$ recovered, the volume of the effluent gas sampled is calculated, based on the normal contents of CO_2 present in air (330 ppm). The values obtained are in good agreement with flowmter measurements to within 10%.

Up to 2 g of fine-ground $BaCO_3$ are mixed into a counting vial with 0.8 g of the gelifying agent (Cab-O-Sil) and with 20 ml of scintillator solution (PPO, dimethil POPOP in toluene).

After five minutes ultrasonic treatment, carbon-14 activity is determined by liquid scintillation, with a counting efficiency of about 60%, depending on the mass of carbonate weighed in the vial.

In the counting range of 10^{-9} Ci/m^3, the statistical error is negligible, while the overall indetermination was estimated to be 12%.

RESULTS

During October 1981 and January 1983, five grab samples based on short-time sampling were taken at Atucha I NPP stack, while measurements were only performed for the CO_2-bound components of carbon-14.

The values obtained laid between 1 10^{-9} Ci/m^3 and 1 10^{-8} Ci/m^3, with a mean value of 6.4 10^{-9} Ci/m^3.

Accounting for reduced forms of carbon-14, catalytic oxidation was performed in further samplings.

Moreover, sampling periods were increased to three or four days. Starting in July 1983 and until August 19, continuous sampling, with very few interruptions, was performed and nine consecutive samples were collected.

The total carbon-14 emission varied between $5.8\ 10^{-9}$ Ci/ m^3 and $1.8\ 10^{-8}$ Ci/m^3. The percentages of CO_2-bound C-14 were quite different, ranging between 43 and 81%.

Taking into account the above-mentioned air throughput, the annual release may be estimated in 16 Ci.

It must be pointed out that the Atucha I stack is the emission point for the off-gas decay tanks, for the building ventilation and for the containment discharges.

CONCLUSIONS

The marked scatter shown by the measured results obtained denotes variations in the carbon-14 concentrations due to the operating conditions of the plant, i.e., discontinuous decay-tank releases.

On the other hand, when decay tanks are not vented, a significative amount of carbon-14 is still released, indicating leakages of the coolant-moderator circuit into the containment air, and/or leakages in the auxiliary building from the primary water treatment systems. (3,7).

For this reason, only continuous sampling over prolonged periods leads to the obtention of representative measurements.

On this basis, a monthly sampling program will be carried out at the Atucha I NPP in order to integrate annual releases.

Furthermore, the main contributions to the total emission will be determined.

BIBLIOGRAPHY

1. Schwibach, J., Riedel, H. and Bretschneider, J. Commission of European Communities, 1978 Document V-3062/78-EN.

2. Moghissi, A.A. and Carter, M.W. IVth. Congress of IRPA. Paris, 24-30 April, 1977. Proceedings v.3: 949-952.

3. Kunz, C.O. 1981 CONF-810606

4. Hayes, D.W. and MacMurdo, K.W. Health Phys. (U.K.) v.32: 215-219, 1977, No. 4.

5. Tschurlovits, M., Pfeiffer, K.J. and Rank, D. Atomkernenerg. Kerntech. (Germany, F.R.) v.40: 267-269, 1982, No. 4.

6. Schuettelkopf, H. and Herrman, G. Commission of European Communities, 1978 Document V/2266/78-D.

7. Beninson, D.J. and Gonzalez, A.J. IAEA-SM-258/53.

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MEASUREMENTS OF ACTIVITY SIZE DISTRIBUTIONS OF RADIOACTIVE AEROSOLS FROM A NUCLEAR POWER PLANT*

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Abstract

The activity size distributions of artificial radioactive aerosols were measured by means of a high volume cascade impactor (Sierra Instruments) in the stack of a nuclear power plant (boiling water reactor) and in the ambient air. In the stack twelve longliving radionuclides were detected. The evaluated activity size distributions are unimodal or bimodal and show mean median diameters in the range 0.3 to 6 µm with standard deviations between 1.20 and 4.70. The size distributions of aerosols with activation products are shifted to larger diameter sizes compared with the distributions of fisson products.

In consequence of the low radioactive aerosol concentration in ambient air only aerosols with the isotopes Cs-137, Cs-134 and Co-60 were registrated.

Introduction

Knowledge of the concentration and the activity size distribution of radioactive aerosols in ambient air is of importance for the estimation of internal radiation dose and the health risk caused by respiration. Therefore the activity size distributions of natural aerosols (Radon and Thoron daughters) were studied recently (1)(2). Of general interest and especially in comparison with the natural aerosols are the activity size distributions of man-made radioactive aerosols produced for instance in medical and scientific laboratories or in nuclear power plants. The constituents of artificial radioactive aerosols in nuclear reactors are related to the isotopic composition of all individual elements invariably present in fuel, coolant and structure materials within the neutron field or introduced as impurity or additive.

Measurements and Data Evaluation

In the period march 1982 through october 1982 aerosol samples were taken with a high volume cascade impactor series 230 of Sierra Instruments Inc. in the stack of a nuclear power plant and a location situated approximately one kilometer downwind of the power plant . The impactor consists of 5 stages with rectangular jets and a backup filter. The flow rate is regulated to about 60 m^3/h. The collection efficiencies of the impactor stages, including interstage losses, were measured by means of monodisperse liquid test aerosols in the size range 0.2 µm to 7 µm. The collection media used was glasfiber filters (3).

After sampling periods of $\sim$ 70 h in the stack and about five weeks in the ambient air, the γ-activities of the filters were measured by means of a well type Ge-detector in connection with a multichannel analyser. From the measured activities and collection efficiencies of the impactor the unknown activity size distributions were determined by data inversion (comparison of simulated and measured data) using a modified simplex method according to Nelder and Mead (4). The simplex method is an iterative nonlinear optimization technique to obtain an optimal fit to the measured data. In our case a bimodal lognormal size distribution was used and the unknown parameters were determined according to the principle of minimal quadratic error (2).

The problem of all nonlinear procedures is to find the global minimum of the considered function in limited computation time and steps of iteration. Therefore the capability of the conventional simplex method was improved according to Spendley

standard deviations of these unimodal distributions differ only slightly. This is also the case for nuclides which are described by means of bimodal distributions (e.g. Co-60 and Zn-65). Therefore it is legitimate to calculate mean activity size distributions. The results are compared in figure 3.

In general the obtained mean activity size distributions are described by two modes with median diameters near 0.5 µm and 5 µm. These two modes can be explained by different mechanisms of aerosol production and their subsequent size modification.

Condensation of condensable vapours on the existing aerosols and growth by coagulation lead to the first mode with median diameters in the range from 0.4 to 0.7 µm (aerosols with fisson products Ba-140, J-131, J-133). For instance natural activity size distributions of Radon and Thoron daughters in the ambient air are also determined by these processes.

In additon to the first mode the aerosols with activation products (Co-60, Zn-65, Mn-54, Mo-99) are described primarily by a second mode with median diameters from 5 to 7 µm. The analysis of this mode indicates that the radioactive isotopes are produced by disruption of structure materials and resuspension of settled particles.

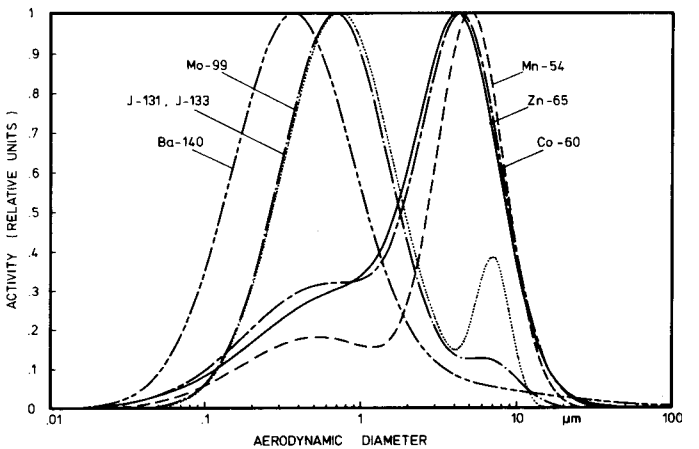

Figure 3:
Mean activity size distributions of J-131, J-133, Ba-140, Mo-99, Mn-54, Zn-65 and Co-60.

The fisson products Cs-134 and Cs-137 constitute an exception. The extreme time dependence of the size distributions are shown in figure 4. Unimodal and bimodal size distributions were found with median diameters varying from 0.5 to 5 µm. In this case mean activity size distibutions were not calculated.

Radioactive aerosols with Ce-141, Ce-144 and Np-239 were registrated in some samples, but no definite size distributions could determined since the activity concentrations were too low.

There is only one measurement in the ambient air which can be compared with the results in the stack since in june 1982 the power plant was shut down. In consequence of the low aerosol concentration only aerosols with Cs-137, Cs-134 and Co-60 were registrated. The size distributions are shifted to smaller diameter sizes; about 90 % of the radioactive aerosols have a diameter lower than 0.3 µm. The aerosols with the isotopes Co-60 show a second mode in the range near 4 µm in accordance with the size distribution found in the stack of the power plant.

(5), Bremermann (6) and Klinker (7). Furthermore the developed computer program was tested with data from hypothetical and well known size distributions (8).

Results and Discussion

The following results of the radioactive aerosols in the stack based on seven measurements with sufficient statistical accuracy. Totally twelve distributions with different longliving radionuclides were analysed. A review of some typical activity size distributions measured in april 1982 are shown in figure 1 as a function of aerosol aerodynamic diameters. The median diameters of the unimodal or bimodal

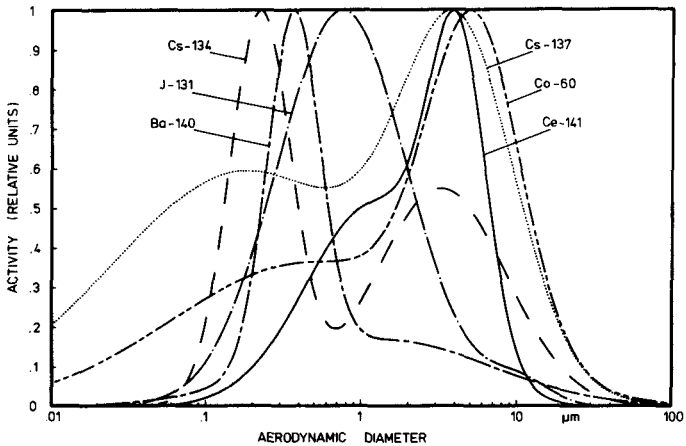

Figure 1: Activity size distributions measured in april 1982

lognormal distributions range from 0.3 to 7 µm and the standard deviations from 1.2 to 4.7. The activity size distributions of most of the nuclides (besides Cs-137 and Cs-134) are nearly time independent, although the actual composition and concentrations of the isotopes vary from time to time. As an example the measured distributions of J-131 are summarized in figure 2. The median diameters and

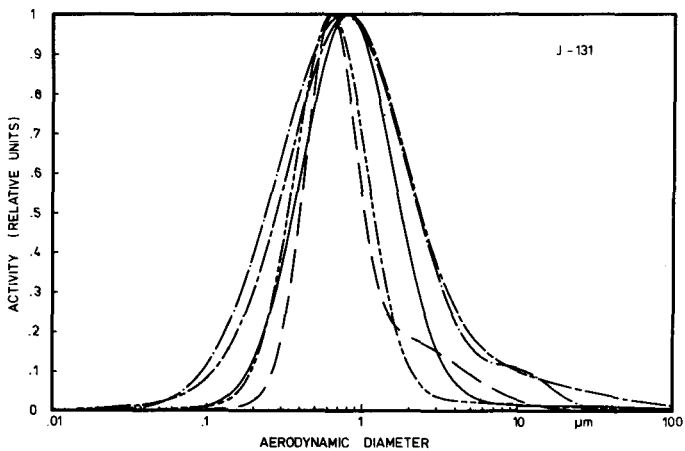

Figure 2: Measured activity size distributions of J-131

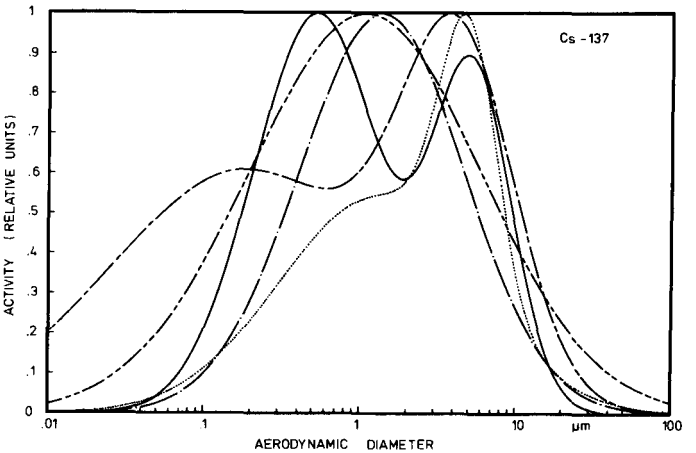

Figure 4:
Measured activity size distributions of Cs-137

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References

1. Knutson, E.O.; George, A.G.; Frey, J.J. and Koh, B.R., Radon Plateout, Part II: Prediction Model, Health Physics 45, 445 - 452 (1983)
2. Becker, K.H.; Reineking, A.; Scheibel, H.G. and Porstendörfer, J., Measurements of Activity Size Distributions of Radon Daughters Indoors and Outdoors, presented at the International Seminar on Indoor Exposure to Natural Radiation and Related Risk Assessment, Capri, Italy, October 3-5, 1983
3. Willeke, K., Performance of the Slotted Impactor, AIHAJ 683 - 691, September 1975
4. Nelder, J. and Mead, R., A Simplex Method for Function Minimization, Computer Journal 7, 308 - 313 (1965)
5. Spendley, W., Nonlinear Least Squares Fitting using a Modified Simplex Minimization, Optimization (edited by Fletcher, R.) London 1969
6. Bremermann, H., A Method of Unconstrained Global Optimization, Math. Biosciences 9, 1 - 15 (1970)
7. Klinker, M., Parameterschätzung in ökologischen Modellen durch Varianten der Simplex-Methode von Nelder und Mead, Diplomarbeit, Göttingen 1983
8. Reineking, A.; Scheibel, H.G.; Hussin, A.; Becker, K.H. and Porstendörfer, J., Measurements of Stage Efficiency Functions including Interstage Losses for a Sierra and Berner Impactor and Evaluation of Data by a Modified Simplex Method, presented at the 11th GAeF Meeting, Munich, F.R.G., September 14-16, 1983

DETERMINATION OF THE RADON DAUGHTERS CONCENTRATION IN AN ATMOSPHERE AND DISTRIBUTION OF RADON AND DAUGHTERS IN VARIOUS ENVIRONMENTS

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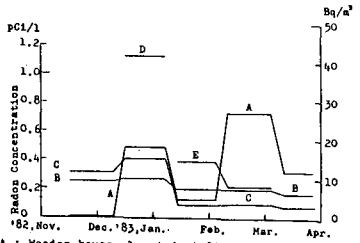
1. Radon and Radon Daughters Concentrations in Various Environment

Internal radiation exposure for members of the public may be mainly caused by the inhalation of radon daughters. The radon and its daughters concentration in an atmosphere is maintained by emanating radon from the soil, building materials and underground water,etc. The observed emanation rate from soil surface, surrounding our laboratory in the Tokyo, is 0.0030—0.0093 Bq/m^2.s and specific activity of radium-226 in that soil is estimated as 30.3 ± 5.2 Bq/kg.

In Japan, a large amount of plaster-board which is including some amounts of artificial phosphogypsum, is used for the important components of dwelling houses instead of wooden board owing its fireproof property. For instance, about 840 Bq/kg of radium-226 is observed in these plaster boards. Therefore, in the case of Japnese dwelling houses, main source of radon is estimated that the remained radium in the plaster board which is supplied from a waste of phosphoric acid manufactures and the probrem of emanated radon from igneous rock, such as granite, or other uranium containing hard materials is not serious source because of the Japanese dwelling house construction.

In the case of an uranium mine,the sources of radon are exhaust air of the mine itself and deposited mill tailings. In the Ningyo-toge mine, the mill tailings and residues of heap leaching are buried to the worked out open pit and covered with the satisfied thickness (:2.8m) of soil grained shale mixture. Therefore,the radon emanation rates from the covered soil surface are indicated comparative low value, averaged of these values is estimated as 0.057 ± 0.012 Bq/m^2.s.



A : Wooden house, located at 21 Km NW direction from centre of the Tokyo.
B : Wooden house, located at nearby centre of the Tokyo.
C : Made by prefabricated concrete, located at 19 Km WNW direction from centre of the Tokyo.
D : Closed room of the second basement in concrete building.
E : Tenth floor of the above building.

Fig.1 Variation of the radon concentrations in various dwelling houses of the Tokyo.

The continuous measurement of the averaged radon concentration in an atmosphere of the open cut working area of the Ningyo-toge mine and that of the dwelling houses in vicinity of the Tokyo is carried out by the passive type radon monitors. This monitor consists of 1.4 l stainless steel vessel with the radon removal filter, collecting centre electrode (covered with the thin aluminum foil) and high voltage supply (-800 v). The alpha radiations of the collected radon daughters are recorded on the cellulose nitrate film (LR-115, Type II) which is located inside of the aluminum foil of centre electrode. The collecting efficiency of this monitor is estimated as 70 % by experimentally. The results of the measurement are shown

in Fig. 1 and Fig. 2.

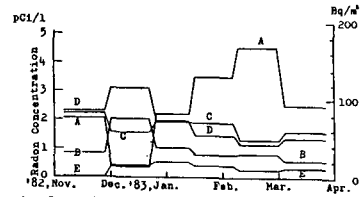
The typical results of investigation of the individual radon daughters concentration in the various locations are indicated in Table 1.

Table 1. Radon daughters concetrations in various environments.

Locations	Conditions of atmosphere	Radon daughters concentrations		
		Ra-A Bq/m³	Ra-B Bq/m³	Ra-C Bq/m³
Indoor of dwelling house in western suburbs of Tokyo	Natural ventilation, in winter	6.14± 0.96	3.63±1.33	3.89±0.74
	Under the somewhat obstructed ventilation, in winter	7.07±0.96	6.62±1.37	2.63±0.74
		6.96±1.00	6.88±1.48	3.55±0.74
Outdoor of suburbs of Tokyo	Wind speed is 1-2 m, in winter	3.03±0.67	3.18±1.30	3.29±0.67
Open cut mining area, in Ningyo-toge	Down stream of the wind, without working	29.0± 6.4	13.9± 1.9	12.2± 1.9
Heap leaching site	Wind speed is 1-2 m	16.8± 4.4	0.3 ± 1.0	6.7 ± 1.3
Environmental area	Ningyo-toge area	5.8 ± 3.7	3.3 + 1.0	4.2 ± 1.1

2.New Detecting System for Individual Radon Daughters Concentration

The presented system has been developed to determine the extreme low concentration radon daughters in an atmosphere with the comparble short measuring time. The features of this system are application of the continuous counting during the collecting operation and adoption of the additional beta counting informations of Ra-B and Ra-C to the alpha-spectroscopy technique. The system consists of "Ruggedized" silicon surface barrier (SSB) detector (ORTEC CR-28-450-100), low noise charge sensitive preamplifier, linear amplifier, multichannel or three channels pulse height analyser, special type filter holder and pumping system. The block-diagram is shown in Fig.3. The features of this detecting device are the cleanable front electrode and the much less lightsesitive characteristic than other SSB detectors. The actual depletion layer thickness of this detector is estimated as 198 μm from the resistivity of detector and applied bias voltage. It is sufficient thickness for the detection of beta rays of radon daughters. The detector is mounted in the filter holder (shown in Fig. 4) and be able to detect the emitted alpha and beta rays from the collected daughter nuclides on the membrane filter (millipore DA) during collecting operation. The geometrical arrangement of the detector and the filter is indicated in Fig.5.A typical out put pulses height distribution of the linear amplifier is shown in Fig. 6. These pulses are divided to the



A : Down stream side of the averaged winter wind direction.
B : Upper stream side of the averaged winter wind direction.
C : Periphery of the mining area, down stream side of other seasons.
D : Periphery of the mining area, upper steam side of other seasons.
E : Back ground level of the Nigyo-toge area.

Fig. 2 Variation of radon concentrations at various locations in Ningyo-toge mining area.

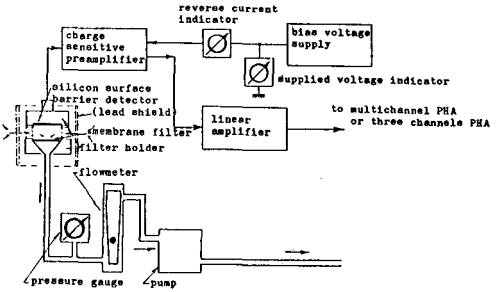


Fig. 3 The blockdiagram of the detecting system.

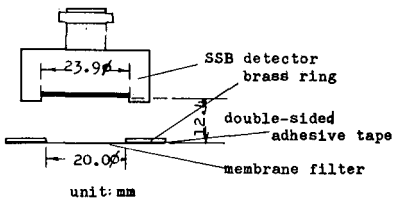


Fig. 5 Geometrical arrangement of the detector and filter.

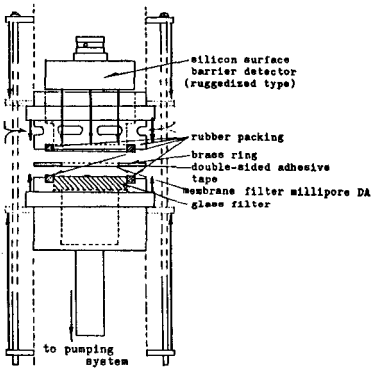


Fig. 4 Sectional view of the special filter holder.

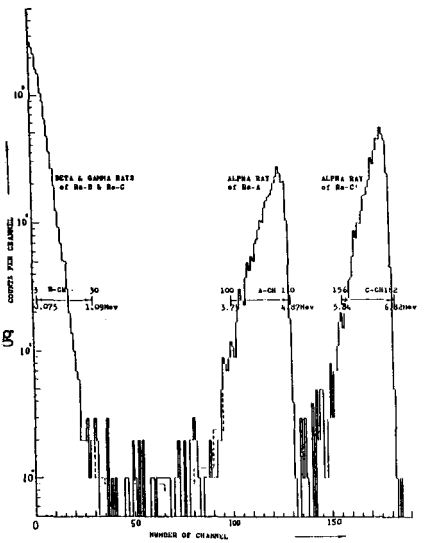


Fig. 6 Pulse height distribution of the detecting system.

following three groups. These groups are (1) beta channel (B-CH, 0.075 to 1.1 Mev-eq), (2) Ra-A alpha channel (A-CH, 3.7 to 5.0 Mev-eq) and Ra-C channel (C-CH, 5.8 to 6.9 Mev-eq). The counts of B-CH provides for the usefull information concerning to the Ra-B and Ra-C activities. The counting efficiency of each of the groups for the various radiations of daughter nuclides are estimated experimentlly as the followings, Each of the efficiency of the A-CH for alpha rays of Ra-A and Ra-C' is 10.61 % ($=\eta_{AA}$) and 0.0799 % ($=\eta_{cA}$), that of B-CH for alpha ray of Ra-A and beta rays of Ra-B and Ra-C is 0.184 % ($=\eta_{AB}$), 8.47 % ($=\eta_{BB}$) and 10.04 % ($=\eta_{BC}$) and that of C-CH for alpha ray of Ra-C' is 10.04 % ($=\eta_{cc}$), respectively. The analytical approachs to the determination of net counts of each of the channels in the appropriate collecting time are obtained from the integrated forms of the Bateman's equations as follows;

$$I_{AA} = \frac{C_a V}{\lambda_a}\left(T_c - \frac{1}{\lambda_a}\left(1 - e^{-\lambda_a T_c}\right)\right)$$

$$I_{AB} = \frac{C_a V}{\lambda_a}\left(\frac{\lambda_b}{(\lambda_b - \lambda_a)}\left(T_c - \frac{1}{\lambda_a}\left(1 - e^{-\lambda_a T_c}\right)\right) + \frac{\lambda_a}{(\lambda_a - \lambda_b)}\left(T_c - \frac{1}{\lambda_b}\left(1 - e^{-\lambda_b T_c}\right)\right)\right)$$

$$I_{AC} = \frac{C_a V}{\lambda_a}\left(\frac{\lambda_b \lambda_c}{(\lambda_b - \lambda_a)(\lambda_c - \lambda_a)}\left(T_c - \frac{1}{\lambda_a}\left(1 - e^{-\lambda_a T_c}\right)\right) + \frac{\lambda_a \lambda_c}{(\lambda_a - \lambda_b)(\lambda_c - \lambda_b)}\left(T_c - \frac{1}{\lambda_b}\left(1 - e^{-\lambda_b T_c}\right)\right) +\right.$$

$$+ \frac{\lambda_a \lambda_b}{(\lambda_a-\lambda_c)(\lambda_b-\lambda_c)}\left(T_c - \frac{1}{\lambda_c}\left(1 - e^{-\lambda_c T_c}\right)\right)\Big)$$

$$I_{BB} = \frac{c_b v}{\lambda_b}\left(T_c - \frac{1}{\lambda_b}\left(1 - e^{-\lambda_b T_c}\right)\right)$$

$$I_{BC} = \frac{c_b v}{\lambda_b}\left(\frac{\lambda_c}{(\lambda_c-\lambda_b)}\left(T_c - \frac{1}{\lambda_b}\left(1 - e^{-\lambda_b T_c}\right)\right) + \frac{\lambda_b}{(\lambda_b-\lambda_c)}\left(T_c - \frac{1}{\lambda_c}\left(1 - e^{-\lambda_c T_c}\right)\right)\right)$$

$$I_{cc} = \frac{c_c v}{\lambda_c}\left(T_c - \frac{1}{\lambda_c}\left(1 - e^{-\lambda_c T_c}\right)\right)$$

$C_A = \eta_{AA} I_{AA} + \eta_{cA} I_{cc}$, $C_B = \eta_{BB} I_{BB} + \eta_{Bc}(I_{Bc} + I_{cc}) + \eta_{AB} I_{AA}$, $C_C = \eta_{cc}(I_{Ac} + I_{Bc} + I_{cc})$

Where c_a, c_b and c_c are the averaged concentration (activities) of the Ra-A, Ra-B and Ra-C (=Ra-C'), λ_a, λ_b and λ_c are the decay constant of the Ra-A, Ra-B and Ra-C, v is the collecting rate (= flow rate) and T_c is the collecting period of time (=counting time).

I_{AA} is the activity of Ra-A nuclide, collected as Ra-A.
I_{AB} is the activity of Ra-B nuclide, collected as Ra-A.
I_{Ac} is the activity of Ra-C or Ra-C' nuclide, collected as Ra-A.
I_{BB} is the activity of Ra-B nuclide, collected as Ra-B.
I_{Bc} is the activity of Ra-C or Ra-C' nuclide, collected as Ra-B.
I_{cc} is the activity of Ra-C or Ra-C' nuclide, collected as Ra-C.
C_A, C_B and C_C are the net counts of A-CH, B-CH and C-CH.

The following equations are provided for the determination of the individual daughter nuclides concentrations and these statistical error estimations.

$c_a = (0.353\ 956\ C_A - 0.002\ 671\ 75\ C_c) \div v$, (collecting time : 10 min.)
$c_b = (-0.023\ 570\ 0\ C_A + 0.256\ 976\ C_B - 0.243\ 423\ C_c) \div v$
$c_c = (0.000\ 871\ 70\ C_A - 0.028\ 765\ 3\ C_B + 0.853\ 747\ C_c) \div v$
$WL = (1.115874 \times 10^{-4} C_A + 5.388\ 885 \times 10^{-4} C_B - 1.525\ 183 \times 10^{-4} C_c) \div v$
$\Delta c_a = (1.252\ 846 \times 10^{-1} \Delta C_A^2 + 7.137\ 750 \times 10^{-1} \Delta C_c^2)^{\frac{1}{2}} - v$
$\Delta c_b = (5.555\ 550 \times 10^{-4} \Delta C_A^2 + 6.603\ 684 \times 10^{-2} \Delta C_B^2 + 5.925\ 468 \times 10^{-2} \Delta C_c^2)^{\frac{1}{2}} \div v$
$\Delta c_c = (7.598\ 565 \times 10^{-7} \Delta C_A^2 + 8.274\ 452 \times 10^{-4} \Delta C_B^2 + 5.700\ 033 \times 10^{-2} \Delta C_c^2)^{\frac{1}{2}} \div v$
$\Delta WL = (1.245\ 175 \times 10^{-8} \Delta C_A^2 + 2.904\ 008 \times 10^{-7} \Delta C_B^2 + 2.326\ 183 \times 10^{-5} \Delta C_c^2)^{\frac{1}{2}} \div v$

$c_a = (0.083\ 678\ 7\ C_A - 0.000\ 631\ 61\ C_c) \div v$,(collecting time: 30 min.)
$c_b = (-0.007\ 897\ 8\ C_A + 0.033\ 444\ 9\ C_B - 0.031\ 644\ 4\ C_c) \div v$
$c_c = (0.000\ 389\ 89\ C_A - 0.010\ 390\ 96\ C_B + 0.038\ 740\ 5\ C_c) \div v$
$WL = (2.137\ 362 \times 10^{-5} C_A + 5.884\ 617 \times 10^{-5} C_B - 6.875\ 371 \times 10^{-6} C_c) \div v$
$\Delta c_a = (7.002\ 131 \times 10^{-3} \Delta C_A^2 + 3.989\ 274 \times 10^{-7} \Delta C_B^2)^{\frac{1}{2}} \div v$
$\Delta c_b = (6.237\ 507 \times 10^{-5} \Delta C_A^2 + 1.118\ 558 \times 10^{-3} \Delta C_B^2 + 1.001\ 371 \times 10^{-3} \Delta C_c^2)^{\frac{1}{2}} \div v$
$\Delta c_c = (1.520\ 155 \times 10^{-7} \Delta C_A^2 + 1.079\ 720 \times 10^{-4} \Delta C_B^2 + 1.500\ 830 \times 10^{-3} \Delta C_c^2)^{\frac{1}{2}} \div v$
$\Delta WL = (4.568\ 316 \times 10^{-10} \Delta C_A^2 + 3.462\ 872 \times 10^{-9} \Delta C_B^2 + 4.727\ 073 \times 10^{-7} \Delta C_c^2)^{\frac{1}{2}} \div v$

Where unit of c_a, c_b and c_c is the dpm/l, and v is the l/min. ΔC_A, ΔC_B and ΔC_c are the statistical counting error of the A-CH, B-CH and C-CH, and WL is the working level.

The relative counting error of B-CH is somewhat larger than that of other channeles owing high back ground counts of this channel. The back ground counts of B-CH is 6to7 cpm under the without shielding and is 2 to 3 cpm under the lead shielding. Some calculated precisions of the presented method is indicated in Table 2.

Table 2. The precisions for the various concentrations and collecting times (v = 16 l/min, back ground counts of the B-CH = 7 cpm.)

Concentrations	Collecting time	Ra-A	Ra-B	Ra-C	WL
1:1:1	10 min.	±18.2 %	±41 %	± 17.3 %	± 17.8 %
11.1 Bq/m³	30 min.	± 9.1 %	±12.9 %	± 8.7 %	± 4.4 %
0.002 94 WL	60 min.	± 6.3 %	± 5.9 %	± 6.7 %	± 1.8 %
1.11 Bq/m³	10 min.	±57.8 %	—	± 62.5 %	—
	30 min.	±28.9 %	±73.7 %	± 33.5 %	± 27.9 %
0.000 294 WL	60 min.	±20.0 %	±32.6 %	± 25.8 %	± 10.0 %

RADIOACTIVE POLLUTION OF ATMOSPHERIC AIR AT WARSAW FROM 1975 TO 1982

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INTRODUCTION

Since the beginning of 1975 the Central Laboratory for Radiological Protection at Warsaw has been conducting a continuous surface air monitoring programme with the objective of studying temporal variances of natural and man-made radioisotopes and toxic stable elements concentrations in surface air in Poland. The radioactivity measurement data are used for estimating the exposure of Polish population to fallout from atmospheric nuclear tests [1, 2].

MATERIALS AND METHODS

Air dust samples from about 25,000 m^3 are collected on a two-week basis at the sampling site, Warsaw /52°15'N - 21°E/, on the fiber glass FPP 15-1.7 type Petrianof filters /area 0.25 m^2/.

In the current air pollution monitoring programme four basic nuclear techniques are used for measurements. Each sample /about 5 g of air dust/ is divided in two parts - one for gamma counting, the other for X-ray fluorescence and neutron activation analysis. The first part after gamma counting is radiochemically prepared for alpha spectrometrical measurements.

Gamma spectra of air dust samples are obtained by radioactivity measurements with high resolution HPGe semiconductor detectors /coaxial and planar/ shielded in a 5 cm thick lead castle. Since the last year an anticompton spectrometer constructed in our Laboratory /germanium detector shielded with an intrinsic NaI crystal and 10 cm thick lead castle/ has been used. The spectra obtained from a multichannel analyser system Plurimat 20 are analysed by a programms written on the Multi-8 Intertechnique and HP-85 computers enabling either automatic peak searching or manual isotope identification and activity concentration calculations.

RESULTS

The results of gamma radioactivity measurements cover the period of 1975 - 1982. From these data we have estimated the radiological hazard to man from 10 radioisotopes detected during that time in the air in Warsaw, having been introduced to the environment as a result of atmospheric nuclear explosions in the northern hemisphere carried out by the People's Republic of China. Since 1974 nine tests in the atmosphere of about 9 Mt yield total have been performed.

Monthly activity concentrations between 1975 and 1982 of ^{137}Cs, ^{54}Mn, ^{95}Zr, ^{103}Ru, ^{106}Ru, ^{125}Sb, ^{131}I, ^{140}Ba, ^{141}Ce, ^{144}Ce

are presented in Fig. 1. Data for ^{7}Be - nuclide produced by cosmic rays interactions in the atmosphere are also presented. Annual mean values of activity concentrations for each of these radioisotopes are presented in Table 1.

DISCUSSION

Short-lived elements such as ^{131}I and ^{140}Ba considered to be indicative of recent nuclear atmospheric tests were detected 5 times during 1975 - 1982 and are in good correlation with the dates of Chinese nuclear tests. The concentrations of gamma radioisotopes in the air in Warsaw were highest in 1975, 1977/78 and in 1981. Only slight increase of ^{144}Ce, ^{137}Cs and ^{103}Ru was observed during 1976 and 1979/80. This increase shows that even low-yield tests of 20 kt are detectable within a week over the territory of Poland by air radioactivity measurements.

The results are in linear correlation with those from the corresponding sampling sites such as Ordfordness, Suffolk and Milford Haven, Pembrokeshire in England, and Moosonee, Ontario in Canada located at Warsaw's latitude, i.e. 52°N /correlation coefficient for ^{144}Ce and ^{95}Zr is over 0.9 for the period 1977/78/. However, our results show higher activity concentrations of these elements.

The main contribution to the radioactive air pollution at Warsaw in 1975 was given by the high yield /1 Mt/ test of 1974. Two tests in September and November, 1976, of medium /200 kt/ and high yield /4 Mt/ are responsible for the high radioactivity level of all the detected radioisotopes in 1977/78 and the test of October, 1980, /600 kt/ for 1981. Respective dose equivalents for these periods by inhalation to the lungs for Warsaw population are presented in Table 2. They were calculated from conversion coefficients [3, 4]. When compared with average dose equivalents to the lungs from the inhalation of natural occuring radionuclides in air such as ^{222}Rn, ^{220}Rn and their daughters, they constitute only a small percentage /about 0.1%/, and up to 20% of the natural long lived alpha-radionuclides inhaled.

LITERATURE

1. Jagielak J.: Pollution of atmospheric air with toxic and radioactive particulate matter investigated by means of nuclear techniques, Report INR-1744/CLOR/D/B, Warsaw 1978.
2. Jagielak J., Pietruszewski A.: Radioactive air pollution of atmospheric air at ground level in Poland during 1975-1977, Postępy Fizyki Medycznej, XIV, 3, 139-149, Warsaw 1979.
3. UNSCEAR, 1982 Report to the General Assembly - Ionising Radiation Sources and Biological Effects.
4. Protokol 12 Sowieszczenia Specialistow Stran Czlenow SEW i SFRIU po temam razdieła 3820, 16-21 maja, Moskwa 1983.

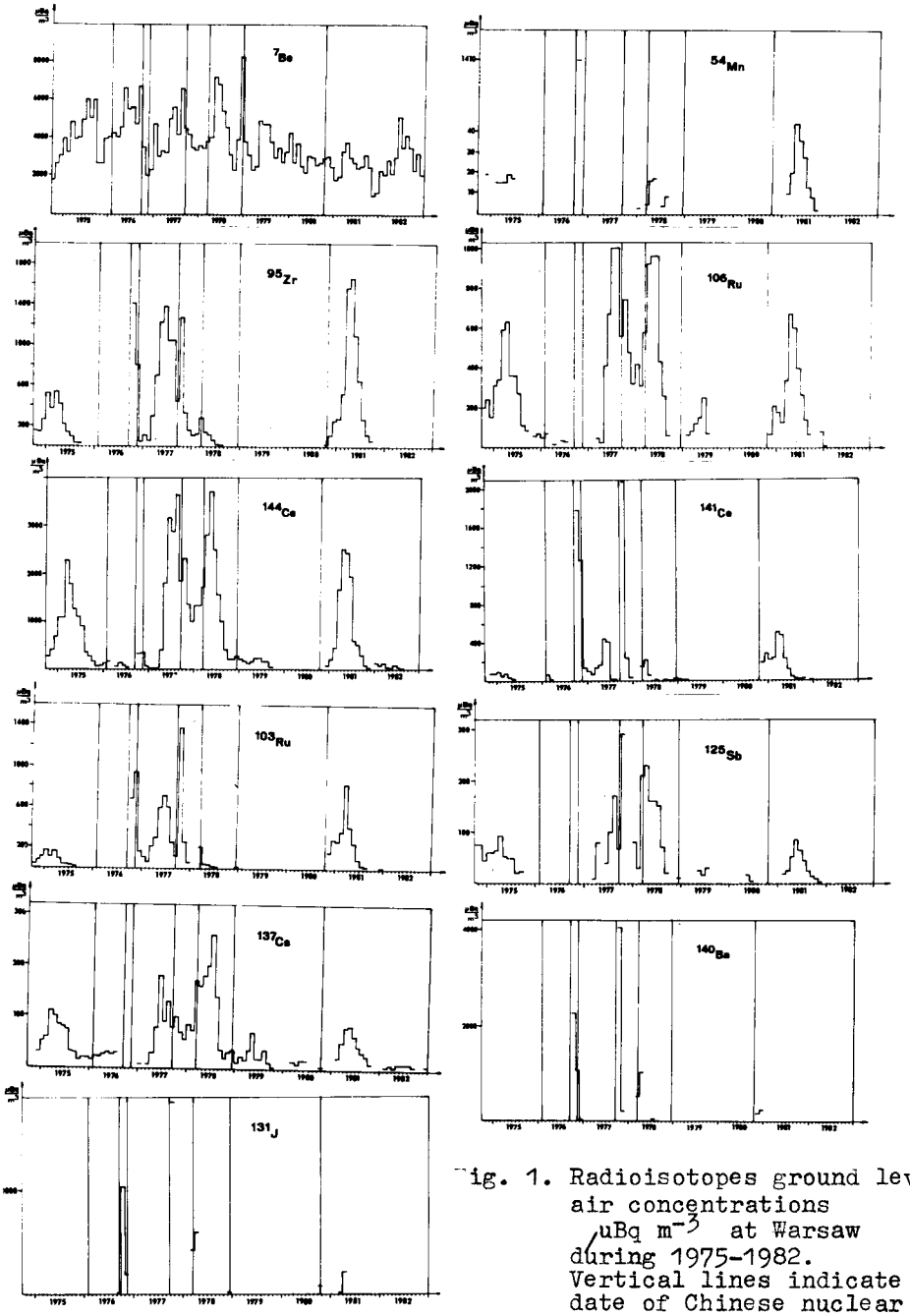

Fig. 1. Radioisotopes ground level air concentrations ($\mu Bq\ m^{-3}$) at Warsaw during 1975-1982. Vertical lines indicate date of Chinese nuclear explosions.

Table 1. Gamma activity concentrations in air near ground level at Warsaw /10^{-6} Bq m^{-3}/.

Isotope	Year							
	1975	1976	1977	1978	1979	1980	1981	1982
^{7}Be	4210	4440	4240	4670	3500	2900	2410	3000
^{54}Mn	7	120	n.d.	4	n.d.	n.d.	13	n.d.
^{95}Zr	200	190	670	70	n.d.	33	520	n.d.
^{103}Ru	48	150	380	23	n.d.	36	260	n.d.
^{106}Ru	270	17	440	410	61	23	240	n.d.
^{125}Sb	41	n.d.	63	95	6	2	28	n.d.
^{131}I	n.d.	100	160	86	n.d.	8	20	n.d.
^{137}Cs	56	12	74	115	21	5	34	6
^{140}Ba	n.d.	280	350	132	n.d.	33	n.d.	n.d.
^{141}Ce	21	270	310	37	2	40	155	n.d.
^{144}Ce	850	170	1560	1420	115	8	882	47

n.d. - not determined

Table 2. Dose equivalents /10^{-6} Sv/ from inhalation to lungs.

Isotope	Year							
	1975	1976	1977	1978	1979	1980	1981	1982
^{54}Mn	<0.001	0.006	n.d.	0.001	n.d.	n.d.	<0.001	n.d.
^{45}Zr	0.026	0.025	0.088	0.009	n.d.	0.004	0.069	n.d.
^{103}Ru	0.005	0.025	0.42	0.002	n.d.	0.004	0.029	n.d.
^{106}Ru	1.87	0.12	3.05	2.84	0.42	0.16	1.67	n.d.
^{137}Cs	0.003	<0.001	0.004	0.007	0.001	<0.001	0.002	<0.001
^{140}Ba	n.d.	0.003	0.004	0.002	n.d.	<0.001	n.d.	n.d.
^{141}Ce	0.002	0.032	0.037	0.004	<0.001	0.044	0.018	n.d.
^{144}Ce	4.66	0.93	8.55	7.78	0.63	0.044	4.83	0.26
Total	6.57	1.14	10.65	1.06	0.22	0.22	6.62	0.26

n.d. - not determined

INTERNATIONAL RADIATION PROTECTION ASSOCIATION

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THE EFFECTS OF PLUTONIUM REDISTRIBUTION ON LUNG COUNTING

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INTRODUCTION

Lung counting procedures are influenced by both interorgan and intraorgan distributions. The influence of interorgan distribution (translocation to liver, lymph nodes, etc.) is widely recognized and calibrations have been made for such distributions. Although nonuniform distribution within the lung is recognized, calibrations generally assume a uniform distribution since the magnitude of nonuniformities is unknown. At a meeting on the measurement of heavy elements in vivo (Swinth 1976), it was pointed out by several authors that "reasonable" nonuniform distributions could lead to changes of 200-300% in the calibration factor as compared to a uniform distribution. The purpose of this paper is to show that a nonuniform distribution is a reasonable expectation and to indicate the influence of a nonuniform distribution on assessment of plutonium depositions in the lungs by external counting techniques.

BACKGROUND

The spatial distribution of plutonium within the lungs is not uniform. The initial distribution depends on health, the breathing pattern at time of deposition and the particle size. Deposited materials do not remain at or near the site of deposition for the duration of their residence in the lung. McInroy (McInroy et al. 1975) measured concentration ratios ranging from 10 to 37 times as great in the subpleural region as the parenchyma (internal or central area) of the lung. Nelson (Nelson et al. 1972) reported a concentration ratio of 59:1. These results are for workers with 25 and 26 years of potential exposure, respectively. Results from an autopsy one month post exposure (McInroy et al. 1975) did not show a difference in the subpleural and parenchymal region concentrations.

Data from animal studies (Rhoads, Mchaffey and Sanders 1982) was analyzed to determine the change in concentration with time for the subpleural and internal regions of the lungs of rats. It was found that the relative particle concentration in the subpleural areas increased to 122% of the original after 180 days while the concentration in the rest of the lung decreased to 92% of the original. Calculating the subpleural to internal ratios for individual animals, one obtains a best-fit line of $y = 1.35 + 0.0029t$ where t is in days.

MATHEMATICAL MODEL

To further study the influence of non-uniform distributions on lung counting, a mathematical model of the thorax and phoswich counting system was developed paralleling the work of Dudley and ben Haim (Dudley and ben Haim 1968). The model assumes an elliptical thorax and was refined to study nonuniform distributions and different tissue types. Two 5-inch diameter scintillation counters (Phoswich detectors) are positioned over the upper chest to simulate the commonly used counting geometry.

The model assumes linear attenuation with the attenuation for a given tissue type being the combination of photoelectric and Compton effects

($\mu_{pi} + \mu_{ci} - \mu_i$, for tissue type i). Detector response for a given configuration of lung, chest wall, and detector is calculated by the sum of the transmission fraction for each path from the lung inversely weighted by the square of the total path length. This assumption is performed for a matrix of points throughout the chest model.

$$\text{Detector response} = \sum_{\text{Detector}} \sum_{\text{Lung}} \frac{p(x,y,z)e^{-\Sigma_i \mu_i d_i}}{(\Sigma_i d_i)^2}$$

where $\sum_{\text{Lung}} = \sum_x \sum_y \sum_z$

$$\sum_{\text{Detector}} = \sum_{i=1}^{\text{Number of detector points}}$$

$\Sigma_i \mu_i d_i$ = summation of path-length absorption product over tissue type i

d_i = distance of vector through tissue type i

$p(x,y,z)$ = intensity at point (x,y,z)

The tissue types included for study are bone, adipose tissue, muscle, lung, and soft tissue. Assumptions concerning tissue parameters are the following: bone covers 45% of frontal area with a uniform thickness; the chest wall is made up of 20% adipose tissue, 80% muscle tissue; and the density of lung tissue is 0.3 that of soft tissue. The attenuation coefficients used are identical to those in Newton et al (Newton, Taylor and Anderson 1978).

The model was verified by reproducing a configuration used in an empirical ^{103}Pd study (Newton, Taylor and Anderson 1978) and comparing the values for the response ratio, R (R is the ratio of ^{103}Pd x-ray intensity to the ^{238}Pu x-ray intensity). The maximum error in the three cases used for verification was 2% which was felt to be adequate verification.

In order to disperse unit activity in the model for different cases in a manner simulating the expected concentration ratios, calculation of the volume and number of mathematical sample points in a tissue for a given condition was required. The activity is assigned to each point as a function of: a) concentration ratio, b) volume of the respective tissue, and c) the number of sample points in each tissue. For the lung-lymph conditions the same method was used to assure the correct concentration ratios. The basic assumptions were a lymph tissue mass of 15 g, a lung of 900 g and densities for the respective tissues of 1.0 and 0.3 g/cc.

RESULTS

Figure 1 shows the change in detector response to the L x-rays compared to the uniform case response for various subpleural to lung concentration ratios. There is a rapid increase until a concentration ratio of 20:1 is reached at which point the increase tends to level out. At 20:1, approximately 64% of the activity is subpleural for the 3-mm case and 75% for the 5-mm case in our model. At 60:1, the respective percentages increase to

84% and 90%. Two subpleural thicknesses are shown to indicate the effect of increasing thickness on the count rate ratios. In actual tissue samples the subpleural activity will be concentrated near the surface and thicker samples will decrease the observed concentration ratio.

Curves for the ^{241}Am gamma ray (60 keV) are similar, but the curves do not increase as rapidly since the contribution from the deep lung (less attenuation) is greater.

Figure 2 shows the effect of movement to the tracheobronchial lymph nodes on detector calibration. Movement to the subpleural tissues leads to an increased detector response while movement to the lymph nodes leads to a decreased response compared to the same activity uniformly distributed in the lung volume. The change at 60 keV will not be as large as indicated on the figure due to the detection of scattered photons which were not considered in these calculations.

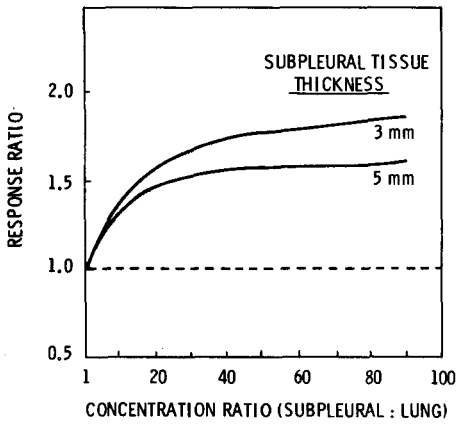

FIGURE 1: Change in Detector Response at X-Ray Energies with Increasing Concentration in the Subpleural Region of the Lung

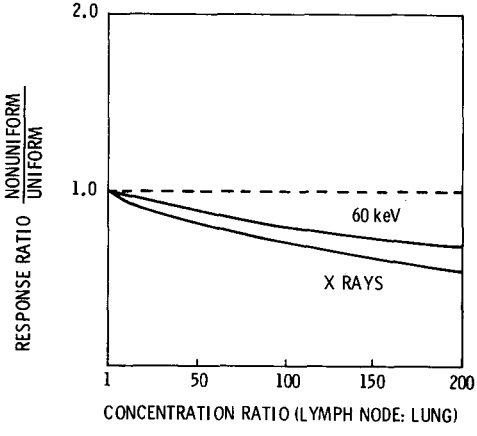

FIGURE 2: The Effect of Movement to the Trachebronchial Lymph Nodes on Lung Counter Response

Table 1 combines the features to show a modeled detector response at x-ray energies for time postexposure based on a linear extrapolation of data from Nelson (Nelson et al. 1972). This indicates that at increasing times postexposure the translocation of material will cause an increase in observed count rate compared to the uniform calibration phantom. This will result in overestimation of retained lung burdens by as much as 50%.

Although the uniform deposition model is adequate at short times postexposure, it is clear that translocation of material will be important in the continued analysis of lung depositions.

The uncertainties in a lung burden estimate caused by subject background estimation, counting statistics, and errors in chest wall attenuation corrections have been estimated to cause a collective uncertainty of approximately 80% in lung burden estimation (Swinth 1976). This is based on an individual containing 16 nCi with a 2.5 cm thick chest wall. Counting time is assumed to be 2000 seconds and errors were estimated at the 2σ level. Based on the current research, the effect of nonuniform distribution is to add an additional biased error of +50%. This would change the uncertainty to +95% and -80%. For a first order correction the regression equation

TABLE 1. Change in Count Rate for L X-Rays Compared to a Uniform Tissue Distribution at Various Times Postexposure

Years Postexposure	Assumed Concentration Ratios Subpleural to Lung	Lymph Nodes to Lung	Normalized Count Rate Ratios
0	1:1	1:1	1.0
5	10:1	30:1	1.26
10	20:1	60:1	1.35
15	35:1	90:1	1.43
25	60:1	140:1	1.48

$y = 1.35 + 0.0029t$ could be used if the time since exposure is known and a correction is necessary. When the other sources of error are considered, the overall impact of nonuniform distribution is not a dominant factor based on the present model. This ignores the effect of illness, breathing pattern, and aerosol characteristics on the initial deposition pattern and translocation rates.

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REFERENCES

Swinth, K. L., ed. 1976. Proceedings of the Workshop on Measurement of Heavy Elements In Vivo. BNWL-2088. Battelle, Pacific Northwest Laboratories, Richland, Washington.

McInroy, J. F., H. A. Boyd, M. W. Stewart, B. C. Eutsler and G. L. Tietjen. 1975. Biomedical and Environmental Research Program of the LASL Health Division. January-December 1975, pg 19. LA-6313-PR, Los Alamos Report.

Nelson, I. C., K. R. Heid, P. A. Fuqua and T. D. Mahony. 1972. "Plutonium in Autopsy Tissue Samples," Health Physics, pg 22, 925-930.

Rhoads, K., J. A. Mahaffey and C. L. Sanders. 1982. "Distribution of Inhaled $^{238}PuO_2$ in Rat and Hamster Lung." Health Physics, 42, pg 645.

Dudley, R. A. and A. ben Haim. 1968. "Comparison of Techniques for Whole Body Counting with NaI(Tl) Detectors,: Phys. Med. Biol., 13, pg 181.

Newton, D., B. T. Taylor and A. L. Anderson. 1978. "X-Ray Counting Efficiencies for Plutonium in Lungs, Derived from Studies with Inhaled Palladium-103," Health Physics, 34, pg 573.

A NEW TECHNIQUE FOR DISCRIMINATION OF INTERNAL AND EXTERNAL CONTAMINATION IN WHOLE-BODY COUNTING

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In whole-body counting, problems may arise when the subjects undergo external contamination. In many cases, especially after accidents, the contamination both of the skin and of the hair cannot be removed completely by the first decontamination treatment. Thus, for the first estimation of radioactivity intake the complex spectrum from internal and external contamination must be separated.

At the KfK a new technique for discrimination of internal and external contamination was developed. The technique is based on the analysis of photopeak asymmetries in the spectra of NaI (Tl) scintillation detectors. The photopeak asymmetry is caused by Compton scattering in the forward direction and thus giving information about the attenuation of the radiation. For description of the asymmetry a modified Gaussian fit can be used. This fit is based on four parameters for the photopeak, the fourth of which being related to the thickness and the absorption coefficient of the material between the radiation source and the detector.

The paper describes the application of the fit for the evaluation of whole-body counting spectra. The problems encountered by uniform external contamination and inhomogeneous internal contamination are discussed.

ESTIMATION OF METABOLIC PARAMETERS FROM INHALED ACTIVITY DISTRIBUTION IN THE HUMAN BODY

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1. INTRODUCTION

The ICRP has proposed a new metabolic model in Publication 30 which makes it possible to calculate individual organ doses due to inhaled radioactive aerosols (1). However the model needs five metabolic parameters, i.e. the inhaled particle size, chemical form, time after inhalation, type of radionuclides and retained activity in the body for each case. These parameters are seldom available from the usual monitors installed in nuclear power plants, except for the type of nuclide and retained activity which can be measured by a whole body counter.

This paper proposes a method to estimate the first three of these parameters for individuals. The metabolic parameters define the behaviour of the inhaled aerosols and consequently, their distribution in the body.

2. RETENTION MODEL

Inhalation of ^{60}Co aerosols was considered as an example. Based on the ICRP model the metabolic process after inhalation is as follows:

(1) Inhaled particles deposit in three distinct regions of the respiratory system. The regional deposition fractions are dependent on the particle size, i.e. the activity median aerodynamic diameter (AMAD).
(2) Elimination from the respiratory system is through two routes. One is solution into body fluids in which the elimination rate depends on chemical form of the particles. The other is transport to the gastrointestinal tract in which the transportation rate depends on the region of initial deposition. The chemical form of the particles is defined as inhalation class in the ICRP model.
(3) Some particles, transported to the gastrointestinal tract, are dissolved into body fluids, while the others are eliminated from the body. The dissolved fraction is related to the chemical form of the particles.
(4) The dissolved particles are distributed via routes dependent on the type of nuclide. Some particles accumulate in particular organs, and others are eliminated from the body.

Behaviours of inhaled aerosols were calculated on the basis of the ICRP model. Figure 1 shows the organ retentions of inhaled ^{60}Co oxide particles as a function of time after inhalation. Furthermore, the retentions depended strongly on other metabolic parameters such as particle size.

3. ESTIMATION METHOD

The present method assumes that the metabolic parameters can be estimated from distribution patterns of inhaled aerosols in the body. While it is well

A NEW TECHNIQUE FOR DISCRIMINATION OF INTERNAL AND EXTERNAL CONTAMINATION IN WHOLE-BODY COUNTING

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The present method assumes that the metabolic parameters can be estimated from distribution patterns of inhaled aerosols in the body. While it is well

known that retentions in individual organs are sensitive to the metabolic parameters, it is not practical to measure these individual retentions. Therefore, the body was divided at the diaphragm into lower and upper regions. The ratio of radionuclide retentions in the lower to upper regions was tested as a practical index of distribution patterns.

While the proposed retention ratio is very simple, it can be assumed to reflect closely the ratio of retention in the gastrointestinal tract to that in the respiratory system. The former retention ratio is designated as the retained activity ratio and can be measured by a scanning type whole body counter (2).

4. RESULTS AND DISCUSSION

Dependencies of the retained activity ratio on the metabolic parameters were simulated by a computer program. Calculations were performed for inhalation of $^{60}C_o$ compounds and their results are summarized in Figs. 2 and 3.

Time dependencies of the retained activity ratio are shown in Fig. 2. The curves appear to be a cubic function. Two or three time values correspond to a single retained activity ratio. It is difficult to identify the correct time after inhalation corresponding to the retained activity ratio measured. However, this might not be a problem for practical applications in which times after inhalation are estimated as from one to 90 days. In this region, the retained activity ratios decrease monotonically with time after inhalation.

Particle size dependencies of the retained activity ratio are shown in Fig.3. The retained activity ratios increase monotonically with particle sizes in AMAD. There is no many-valued function problem, as in the first case. Increments in the retained activity ratios are nearly constant for times ranging from 5 to 90 days after inhalation. The small increase in one day after inhalation is caused by a time delay in the transport of particles from the respiratory system to the gastrointestinal tract. Retained activity ratios decrease with time after inhalation since the time constant for elimination from the gastrointestinal tract is smaller than that from the respiratory system.

From the results described above it is seen possible to estimate the metabolic parameters for inhaled aerosols. In practical applications, chemical forms of aerosols might be estimated from working environment conditions. Then the time after inhalation and particle size can be target parameters for estimation.

5. CONCLUSION

A method was proposed to estimate metabolic parameters for inhaled aerosols, i.e. elapsed time after inhalation, inhaled particle size, and chemical form. The parameters for individuals are estimated from inhaled activity distribution retained in the body.

Metabolic behaviours of inhaled $^{60}C_o$ oxide were calculated on the basis of the ICRP model. Results of computer simulations showed the ratio of retained activities in the lower to upper half of the body was a good index for the activity distribution.

ACKNOWLEDGEMENTS

Authors are pleased to express their gratitude to Dr. Akira Doi of Energy Research Laboratory, Hitachi, Ltd. and Dr. Hideo Matsuzawa of National Institute

of Radiological Sciences for their guidance and encouragement throughout this work.

REFERENCES

(1) Vennart, J.: Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, Part 1, 23-34, Pergamon Press (Oxford) (1978).
(2) Tanaka, E. and Nohara, N.: Oyo Butsuri (in Japanese), 31 (11), 943-948 (1962).

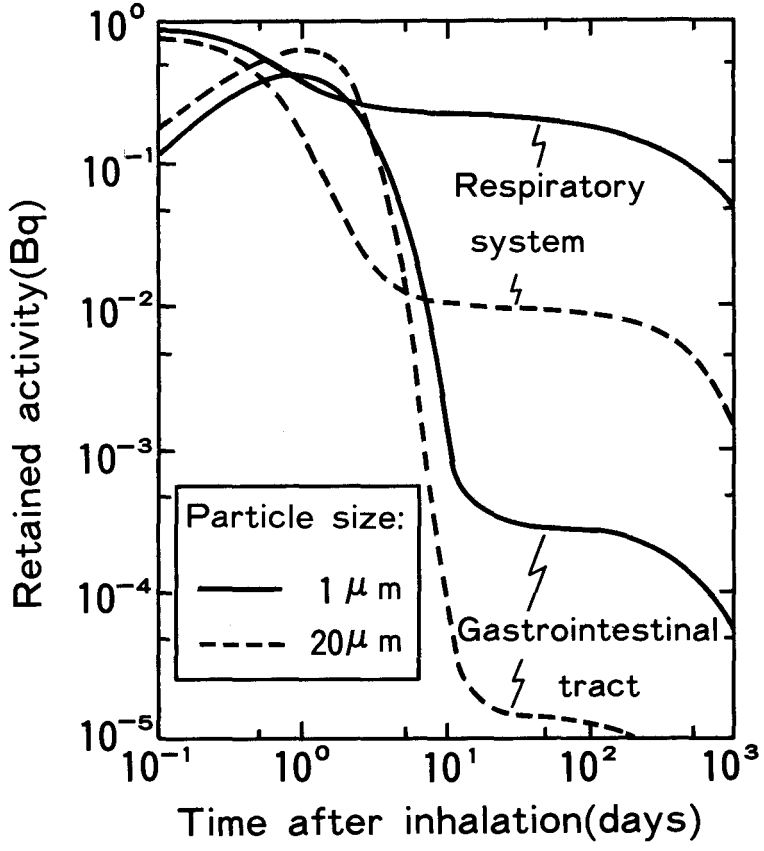

Fig. 1 Time Dependence of Co-60 Retained Activity in the Human Body

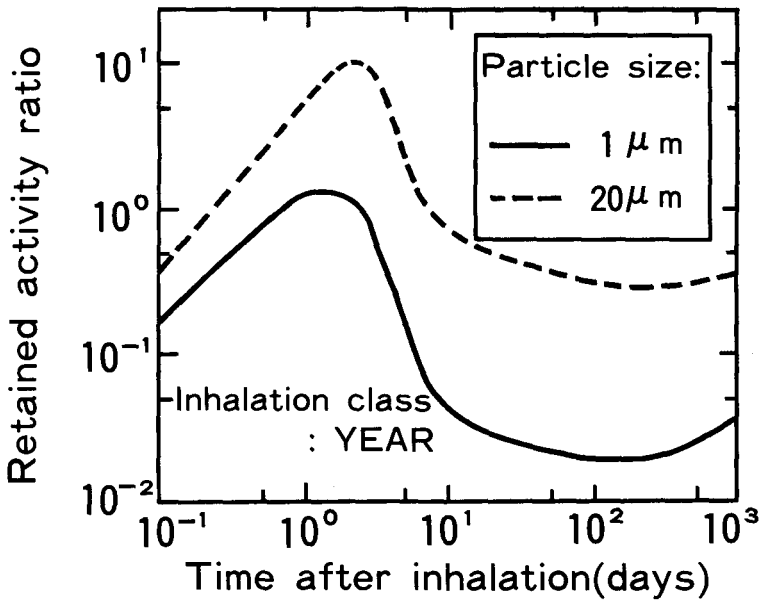

Fig.2 Retained Activity Ratio vs. Time After Inhalation

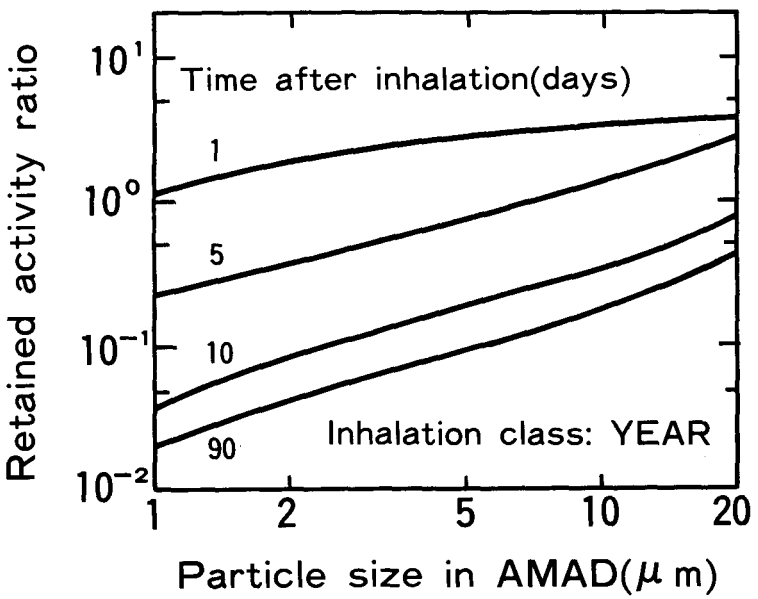

Fig.3 Retained Activity Ratio vs. Particle Size

AN INVESTIGATION OF TISSUE EQUIVALENT MATERIALS BY DETERMINING THEIR BACKSCATTER FACTOR*

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In a general way the backscatter factor $B_{\mathcal{M}}^{(\mathcal{N})}$ can be defined as the ratio of kerma $K_{\mathcal{M}}^{(\mathcal{N})}$ in a material $\mathcal{M}$ at the plane surface of a semi-infinite phantom of a material $\mathcal{N}$ to the kerma $K_{\mathcal{M}}^{(o)}$ at the same point in absence of the phantom. One particular backscatter factor is closely related with the conversion factor $k_a^s = H_s/K_a^{(o)}$ from air kerma $K_a^{(o)}$ in free air to the skin dose equivalent H_s in the ICRU sphere. For a field diameter of 30 cm and when higher order effects are neglected - like the finite curvature of the sphere and photon absorption up to the depth of 7 mg/cm^2 - the following relation holds

$$k_a^s = B_{TE}^{(TE)} \, (\overline{\mu_{tr}/\rho})_{TE}/(\overline{\mu_{tr}/\rho})_a \quad Q \, (1 - g_{TE}), \qquad (1)$$

where (μ_{tr}/ρ) is the mass-energy transfer coefficient averaged over the primary energy fluence spectrum and the index TE refers to the tissue equivalent material as given in ICRU report 19 [1]. Q is a quality factor, which has the value one for photon radiation, and the term $(1 - g_{TE})$ takes account of bremsstrahlung losses, which in the energy region up to 300 keV considered here can be approximated by a value of one. Eq. 1 demonstrates that for radiation protection purposes a determination of $B_{TE}^{(TE)}$ allows quantitative conclusions to be drawn on the extent of the tissue equivalence of the material $\mathcal{N}$. In slight deviation from eq. 1 water kerma instead of tissue kerma will be used in the following. This will affect the absolute values of some of the quantities investigated, it has not, however, any influence on material differences which are the subject of this paper.

The present work contains experimental as well as theoretical results from Monte Carlo (MC) simulations. Details of the two experimental methods [2, 3] and of the MC calculation [4] are given elsewhere.

Fig. 1 shows the energy dependence of the backscatter factor for a paraffin type substance named M3 [5] and for polymethylmethacrylat (PMMA), by trade name referred to as Perspex, Lucite or Plexiglas. The results were obtained in an experiment using very heavily filtered radiation corresponding to ISO specifications [6] with a beam diameter of 10 cm. M3 was chosen as a material because it has been shown in recent MC calculations that its backscatter factor is practically indistinguishable from that of water [7]. Hence fig. 1 demonstrates that PMMA is by no means a "good" substitute for water. For energies below about 60 keV the use of a PMMA phantom will produce errors of the order of 10 %. For higher energies M3 and PMMA become progressively similar and in many cases the remaining differences may be disregarded for energies above 200 keV.

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*This work was supported by the Commission of the European Communities under contract number BIO-218-80D(B).

All the results for M3 and a selection of the results for PMMA in fig. 1 were also obtained with the MC calculations which, however, used water and PMMA as phantom materials. Excellent agreement between both methods was found with deviations ranging always below 3.5 % and on an average below 1 %. The agreement for PMMA implies that both methods yield truly comparable information and the agreement between the MC results for water and the experimental results for M3 confirms the earlier conclusion [7] that M3 is a good substitute for water. An additional variation of the field diameter between 2 cm and 10 cm for both materials established the same quality of agreement. The quicker and more versatile MC calculations are therefore inferred to provide reliable information also in cases where agreement has not been explicitly verified. As the exact properties of the ICRU material [1] have not yet been realized as a solid or liquid material, the degree of equivalence between the ICRU material and any other substance can only be assessed by means of calculations, which, in the present case, have the advantage of being backed experimentally.

The results of the MC calculations for TE material, water and PMMA for a field diameter of 30 cm are shown in fig. 2. Qualitatively water and PMMA display similar differences as in fig. 1 reaching more than 10 % for some energies. On the other hand, over the entire energy range from 10 keV to 300 keV, the results for water and TE material are well within the statistical uncertainty of ±1 %. These results clearly justify the use of either water or M3 for experimental investigations on dose equivalent quantities.

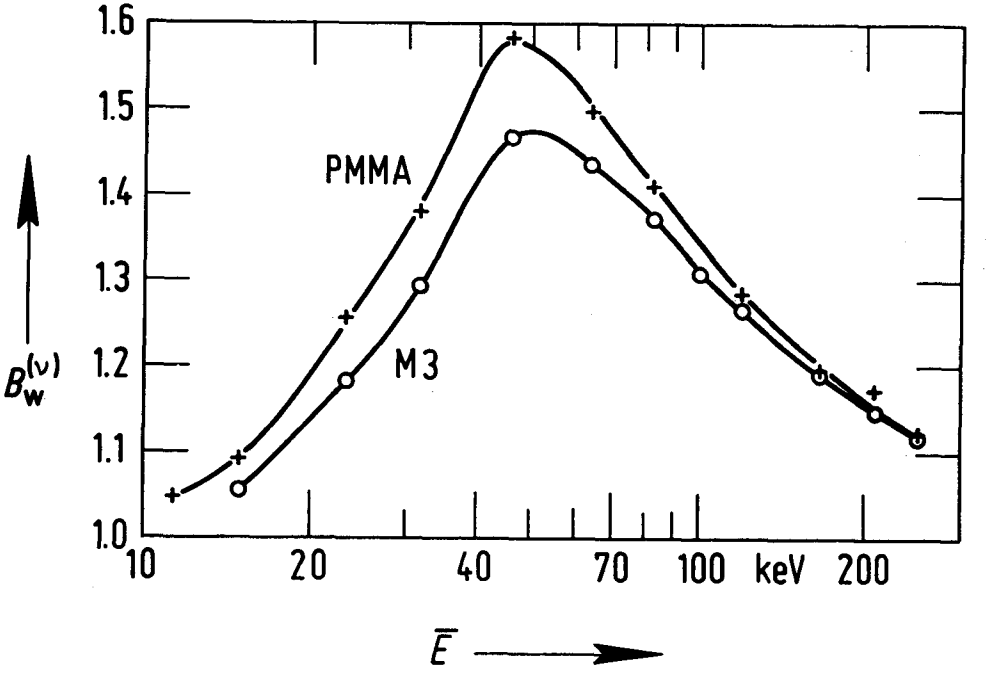

Fig. 1 Energy dependence of the backscatter $B_w^{(M3)}$ and $B_w^{(PMMA)}$ for a field diameter of 10 cm. $\bar{E}$ denotes the mean energy averaged over the primary spectral water kerma of very heavily filtered primary X-rays. The experimental uncertainty is estimated to be below ±1 %.

According to eq. 1 tissue equivalence enters twice into k_a^s: first via the backscattering properties of the TE substitute as discussed so far (upper index); the other question refers to the material which the kerma is measured in (lower index). For an experiment a detector calibrated in tissue kerma will hardly be available; instead a detector calibrated in air kerma is most likely to be used. In this case a conversion from air kerma to tissue kerma is required. Assuming an isotropic detector with energy-independent response with respect to air kerma, this can be achieved by simply multiplying the result in air kerma with $\overline{(\mu_{tr}/\rho)}_{TE}/\overline{(\mu_{tr}/\rho)}_a$, where the average has to be taken over the entire energy fluence spectrum consisting of both primary and backscattered contributions. While the spectrum of the primary radiation may be known reasonably well, e. g. from the catalogue by Seelentag et al. [8], this is not the case for the scattered radiation. This problem was solved by calculating conversion factors of the type

$$f_a^{\mathcal{M}} = (B_{\mathcal{M}}^{(\nu)} - 1)/(B_a^{(\nu)} - 1) \qquad (2)$$

transforming the backscattered contribution from air kerma to kerma in material $\mathcal{M}$. The use of $f_a^{\mathcal{M}}$ is as follows. If it is assumed that the total air kerma $K_a^{(\nu)}$ on the surface of a phantom is known together with the backscatter factor $B_a^{(\nu)}$ then the corresponding tissue kerma at this point, which is essentially the dose equivalent H_s, can be written as

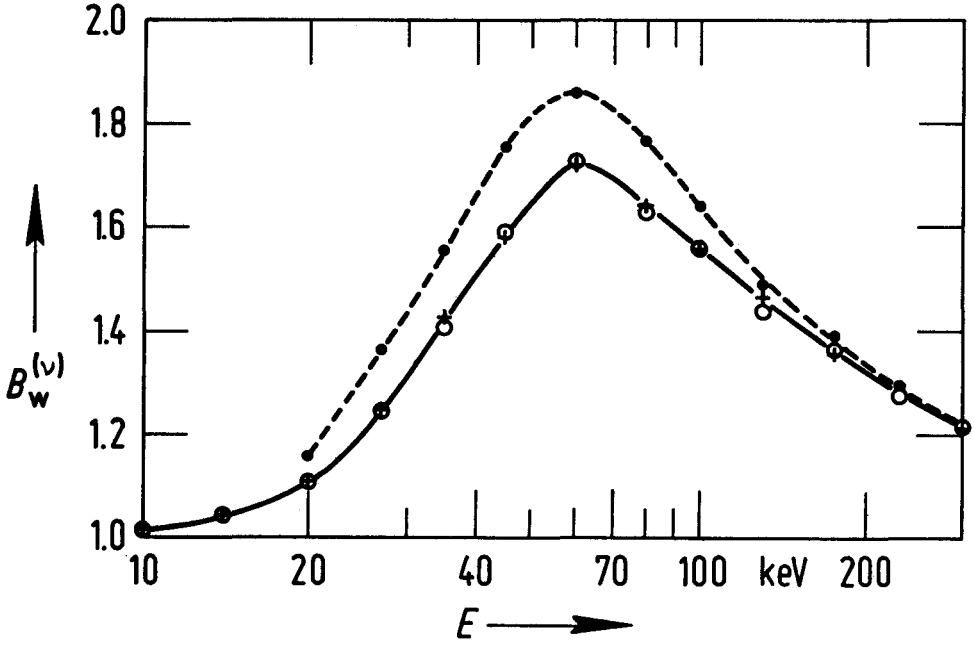

Fig. 2 Energy dependence of the backscatter factor as determined in MC calculations for a field diameter of 30 cm. The solid line refers to both TE material (+) and water (o), the dashed line refers to PMMA (•). The statistical uncertainty is estimated to be below ±1 %.

$$H_s = \left[1 + f^{TE}(B_a^{(\nu)} - 1)\right]\left[(\overline{\mu_{tr}/\rho})_{TE}/(\overline{\mu_{tr}/\rho})_a\right]\left[K_a^{(\nu)}/B_a^{(\nu)}\right]\left[Q(1-g_{TE})\right], \qquad (3)$$

where (μ_{tr}/ρ) has to be averaged over the (known) primary spectrum. Fig. 3 shows the energy dependence of f_a^{TE}; for the sake of completeness the corresponding curve for f_a^w is presented as a dashed line.

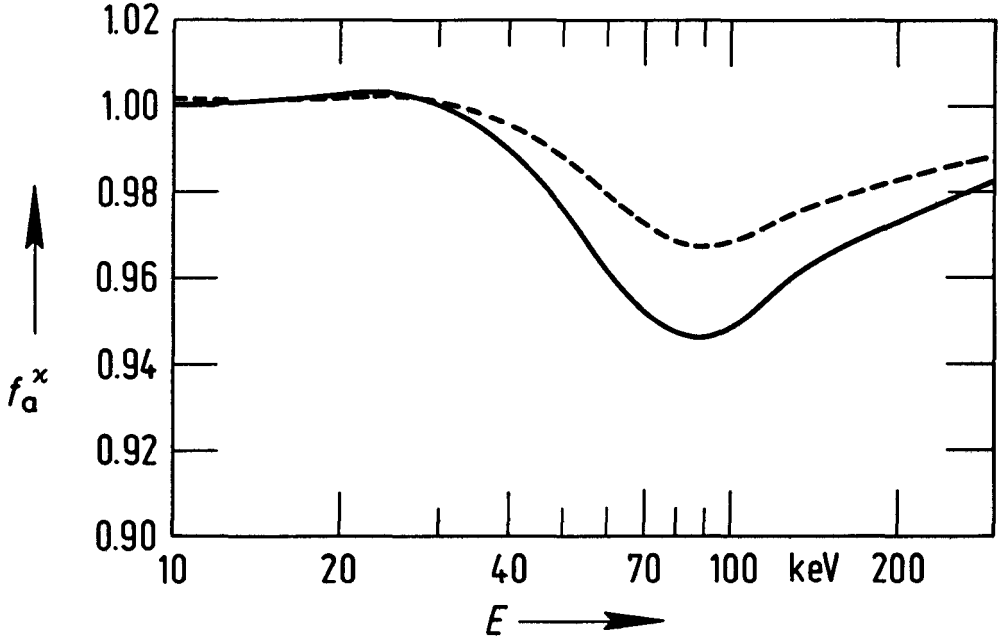

Fig. 3 Energy dependence of $f_a^{\mathcal{M}} = (B_{\mathcal{M}}^{(\nu)} - 1)/(B_a^{(\nu)} - 1)$, which converts the backscattered contribution from air kerma into kerma in material $\mathcal{M}$. For the solid and dashed lines $\mathcal{M}$ signifies TE material and water, respectively.

Literature

[1] ICRU 19, Radiation Quantities and Units, ICRU, Washington DC, 1971.

[2] H. M. Kramer, B. Grosswendt and K. Hohlfeld, to be published.

[3] H. M. Kramer, B. Grosswendt and K. Hohlfeld, to be published.

[4] B. Grosswendt and H. M. Kramer, to be published.

[5] B. Markus, Strahlentherapie, 101 (1956) 111-131.

[6] International Standard ISO 4037, X and reference radiations for calibrating dosemeters and dose ratemeters and for determining their response as a function of photon energy, (1979).

[7] B. Grosswendt, Phys. Med. Biol., (1984) in press.

[8] W. W. Seelentag, W. Panzer, G. Drexler, L. Platz and F. Sautner, A catalogue of spectra for the calibration of dosimeters. Gesellschaft für Strahlen- und Umweltforschung mbH, München, GSF-Bericht 560 (1979).

Fabrication of a Set of Realistic Torso Phantoms for Calibration of Transuranic Nuclide Lung Counting Facilities*

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The calibration of whole body counters for accurate measurement of transuranic nuclides deposited in the lungs requires highly realistic phantoms. The realism includes faithful simulation, not only of the torso and organ morphology, but the composition of appropriate tissues--muscle, bone, adipose, cartilage and lung. Recognizing this fact, the U. S. Department of Energy supported fabrication of three tissue equivalent torso phantoms to be used in an interlaboratory intercomparison program.[1] The phantoms included a human rib cage and removable tissue equivalent lungs, heart, liver and lymph nodes. The tissue equivalent materials, including the foamed lungs, are based on a polyurethane-calcium carbonate composition.[2] Only the rib cage was real tissue.

As the intercomparison program proceeded, it became clear that each of the laboratories involved required the use of the phantoms for longer periods of time than the program schedule allowed. In fact, a number of laboratories expressed interest in having phantoms that could be used as permanent parts of their individual calibration programs. Therefore, we agreed to fabricate a second set of phantoms for each of the laboratories on a cost recovery basis. As a result, 16 phantoms were made for laboratories in England, Canada, the United States and the International Atomic Energy Agency.

The new phantoms were identical to the original three, with the exception that a simulated rib cage would replace the real bone used in the original phantoms. This required development of a bone equivalent material for the ribs and backbone. The ribs are structurally complex with a solid outer layer and a trabecular core of fine bone and marrow. Therefore, choice of a single reference tissue for simulation is not obvious. However, we selected the "rib bone" composition proposed by White[3] as our reference tissue.

Molds existed from the original phantoms for the torso, chest plates and interior organs. The simulated bones for the replica phantoms are made using casts of the real rib cage incorporated in the third of the original phantom set. Following casting, the plastic rib cage is assembled on a plaster torso cavity mandrel (Fig. 1). The mandrels are cast with fine grooves indicating the proper placement of each bone. The torso is cast with the mandrel in place. The mandrel is removed destructively, after the torso has cured and the cut is made for removal of the chest plate (Fig. 2). The organs and chest overlays are cast using conventional casting techniques.[1] With the exception of the bones, lungs and cartilage segments used to join ribs to the sternum, all organs have a muscle equivalent composition. The chest overlays have been made to simulate one of three different chest wall compositions: 100% muscle; 50% muscle, 50% adipose (breast); or 13% muscle, 87% adipose. Table 1 presents the simulant composition summary.

**This work was performed under the auspices of the U. S. Department of Energy by Lawrence Livermore National Laboratory under contract No. W-7405-ENG-48.

Fig. 1. Plaster mandrel of torso interior with simulated bone rib cage in place.

Table 1. Composition of Tissue Simulants Used in Realistic Phantoms

Tissue	Elemental Composition - Wt %[1] H	C	N	O	Ca
Muscle	9.00	61.9	2.98	24.4	1.72
50% Adipose-50% Muscle	9.23	63.0	3.03	23.9	0.84
87% Adipose-13% Muscle	9.46	64.1	3.14	23.3	0.00
Cartilage	8.89	61.1	2.98	24.7	2.33
Bone	6.38	47.2	2.12	31.3	13.0
Lung	8.91	61.6	2.95	24.3	2.24

[1]Nominal composition. Actual composition may vary slightly due to variations in commercial polyurethane supply.

Reproducibility is an important aspect in the use of the realistic phantoms. Moreover, the value of the phantom set is enhanced significantly if there is a precise method of intercomparing results obtained with different phantoms in the series. Therefore, we adopted a marking system for the front surface of the phantoms and each of the chest overlays. The most important positions for external detector measurement are centered over each lung and the liver. A pattern of concentric circles extending, in one-inch increments, to five inches in diameter was applied to the phantom and overlay surfaces at these three positions. In addition, we used a coordinate pattern with 2 cm

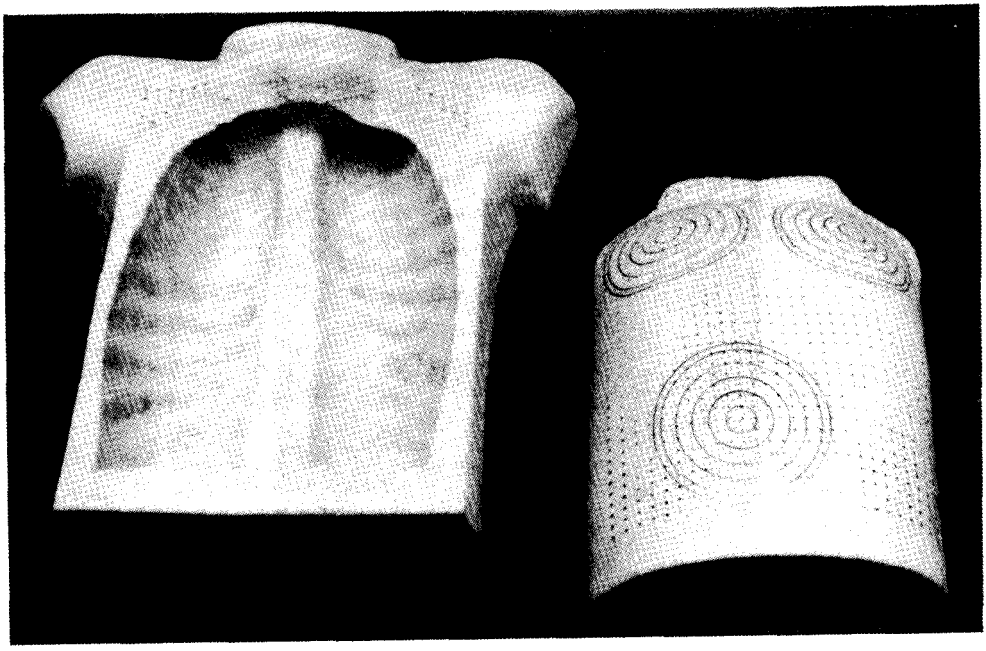

Fig. 2. Torso with chest plate removed.

grid spacing projected on the same surfaces. The patterns have an alpha-numeric labelling system that gives each coordinate intersection a unique identification referencable on any other phantom or chest overlay.

The primary goal of the fabrication of the new phantom set was duplication of the original phantoms as closely as possible. A major concern was the effect of the simulated bone rib cage on transmission of low energy photons such as the 17 keV uranium X-rays from plutonium decay. Therefore, we performed measurements using lungs loaded with ^{238}Pu both in the original phantoms and phantoms selected from the new series. The results of the comparison are illustrated in Fig. 3. The new phantoms have chest walls that are thinner than the original set by about 4 mm. However, the new phantoms match the performance of the originals closely, and the duplication is considered successful.

Summary

A set of 16 tissue equivalent torso phantoms has been fabricated for use by major laboratories involved in counting transuranic nuclides in the lung. These phantoms, which have bone equivalent plastic rib cages, duplicate the performance of the DOE Realistic Phantom set. The new phantoms (and their successors) provides the user laboratories with a highly realistic calibration tool. Moreover, use of these phantoms will allow participating laboratories to intercompare calibration information, both on formal and informal bases.

References

1. Griffith R.V., Dean P.N., Anderson A.L. and Fisher J.C. (1979), "A Tissue Equivalent Torso Phantom for Intercalibration of In-Vivo Transuranic-Nuclide Counting Facilities," IAEA-SM-229/56, Advances in Radiation Protection Monitoring, Stockholm, Sweden 493-502.

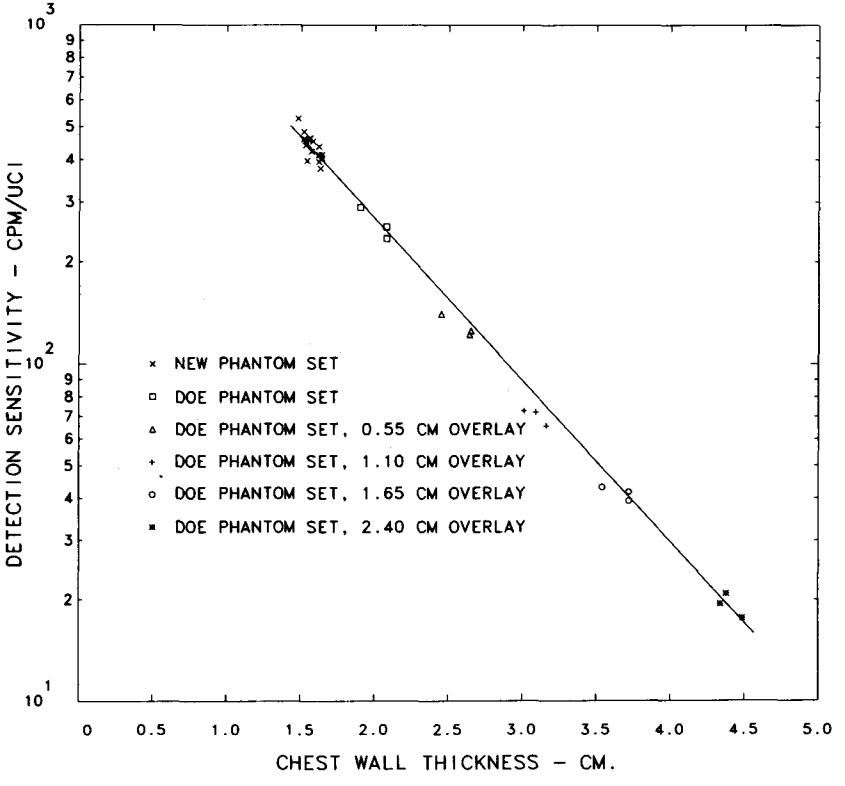

Fig. 3. ^{238}Pu calibration factors for realistic phantoms using dual 4.5 inch diameter phoswich lung counters.

References (continued)

2. Griffith R.V. (1980), "Polyurethane as a Family of Tissue Equivalent Materials," Proc. 5th International Congress of the International Radiation Protection Association, Vol. II, 165-168, Jerusalem.

3. Newton D. and White D.R. (1978), "Attenuation of 13-20 keV Photons in Tissue Substitutes and Their Validity for Calibration Purposes in the Assessment of Plutonium in Lungs," Health Physics, 35, No. 65, 699-703.

LIMITATIONS FOR THE USE OF ANTHROPOMORPHIC PHANTOMS IN X-RAY DIAGNOSTICS

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Radiographs of the anthropomorphic phantom used in our laboratory for dosimetric purposes show remarkable visual deviations in bone absorption compared with x-ray images of the human body taken in the same position. This is demonstrated by radiographs of the lumbar spine from our phantoms in anterior, posterior and lateral positions. The radiation absorption within the single vertebrae is lower than in the surrounding soft tissue. This anthropomorphic phantom is an Alderson-Rando phantom which is composed of different plastic material radioequivalent to corresponding human tissues as muscles, lung tissue, etc.. The phantom contains a natural skeleton. Apparently air is being retained in the bone structure during the preparation procedure. To confirm the observation, the radiation attenuation was measured under small beam conditions (1) through one section with x-rays of diagnostic qualities between 50 and 110 kv. The radiation was filtered in such a way that the half value layer is the same as that for a homogeneous radiation of half the energy, so called "normal radiation". From radiographs we know sections of the phantom in which one vertebra is covering the entire thickness. In our phantom, this is section 25.

The radiation is attenuated by the bone structure itself and the soft tissue filling the trabeculae of the bone. The entrance dose d_o is reduced to

$$D = D_0 e^{-\mu_b d_b} e^{-\mu_s d_s}$$

where d_b of the bone material and d_s of the soft tissue are the thickness $d = d_e + d_s$, and μ_b and μ_s the relevant linear attenuation coefficients. If the soft tissue is replaced by air, the second exponential term is approximately 1, due to the low density of air which is nearly by a factor 1000 lower than the density of the bone marrow tissue. In this extreme case, none of the viscous plastic substances have infiltrated the bone structure during the preparation procedure. This is valid for our phantom, as can be seen from the experimental results. Perhaps it is a casual event that for the lumbar spine inhomogeneities of the bone are compensated by air-filled gaps in such a way that the phantom works like a homogeneous phantom. However, this may contribute to the fact that deviations from natural body composition are not easy to discover, especially when dosimeter probes are used which integrate over a larger area. The x-ray film is the best indicator for detecting small local changes in absorption properties.

This type of anthropomorphic phantom is widely used in many laboratories, also for research in the field of x-ray diagnostics.

The observed deviation may cause errors, if the results from dose measurements are used for dose estimates inside or behind bone in human bodies. A review of published information on construction details confirms our observation. It is described that skeletons are received in a dried condition and that skeletal preparation with soft tissue equivalent plastic material cannot suffuse bone as completely as in the living body (2).

This presentation is meant to alert the attention of users of the same type of phantom showing similar or comparable deviations in relation to real body absorption.

(1) DIN 6845, Teil 1: Prüfung von Strahlenschutzstoffen für Röntgen- und Gammastrahlung; Röntgenstrahlung bis 400 kV, Ausgabe Februar 1980

(2) Alderson-Rando Phantom System, Technical Bulletin No. 431 Januar 1969

INFLUENCE OF CHARGED PARTICLE BUILD-UP ON DOSIMETRIC QUANTITIES IN THE SURFACE LAYER OF THE ICRU SPHERICAL PHANTOM

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INTRODUCTION

The specified depth dose equivalent H_d has recently been proposed as the operational quantity for radiation protection measurements for photon, electron and neutron radiation /1/. This quantity would replace the dose equivalent index H_I which has been shown to be a questionable quantity /2/. Both quantities H_d and H_I are secondarily limited quantities and should provide a conservative estimate for the effective dose equivalent H_{eff} defined in /3/. In /3/ H_I has been explicitly suggested as the limiting quantity for external exposure to penetrating radiation, if information on the actual distribution of dose equivalent in the body is lacking.

The depths to be specified for H_d are 70 µm, 3 and 10 mm /1/ below the ICRU sphere surface. At these depths charged particle build-up (CPB) may play an important rôle, as can be inferred from /4/ for photons and seen from /5/ for neutrons. The air surrounding the ICRU sphere may influence the CPB. This is significant for photons /4/ and will be discussed below for neutrons. The influence of the CPB on H_d for photons and neutrons will be studied and implications for the suitability of H_d as the secondarily limited quantity will be discussed.

The CPB is due to the finite range of secondary charged particles (SCP). Its magnitude can be studied by comparing correct calculations of H_d with kerma approximations $H_{d,k}$ where zero range of SCP is assumed.

PHOTON RADIATION

Table 1 displays results of calculations assuming the ICRU sphere in vacuo and irradiated by a broad parallel beam of monoenergetic photons. In order to check the quality of our calculations, $\hat{H}_3$ and $\hat{H}_{10}$, i.e. the maxima on the corresponding shell, are compared with results obtained by /4/. In the last two columns our results for H_I are compared with those of /4/. With the exception of $\hat{H}_3$ for 2 MeV, one finds that our calculations agree for $\hat{H}_3$ and $\hat{H}_{10}$ well within 5 % which we quote as our calculational error. Details of our code can be found in /6/. The results for H_I up to 3 MeV agree within 3 %. At higher energies, however, our results are increasingly higher than those of /4/. In /7/ various calculations of H_I are compared with experimental results. The values of H_I given in /4/ underestimate the experimental and other calculational results above 4 MeV. Our results for H_I at energies above 4 MeV support the values of /8/ as shown in /7/. The ratio of H_3 to its kerma approximation $H_{3,k}$ - second and third column in table 1 - decreases with incident photon energy from 0.96 to 0.08. The trend with energy is due to the increasing range of the secondary electrons. For $H_{10}/H_{10,k}$ - sixth and seventh column - one correspondingly finds values from 0.97 to 0.30. Calculations of H_{eff} by /9/ - not shown in table 1 - show that H_{10} underestimates H_{eff} for frontal irradiation (AP) for all energies above 2 MeV. At 10 MeV

H_{10}/H_{eff}-AP is 0.43. The kerma approximation $H_{10,k}$ is a conservative estimate of H_{eff}-AP. Our results support the suggestion of /3/ that H_I can be used as a conservative estimate of H_{eff}. For the photon energies shown in table 1 the ratio H_I/H_{eff}-AP is found to vary between 1.35 at 0.662 MeV and 1.23 at 10 MeV.

These results clearly show that the influence of CPB on H_3 and H_{10} for high energy photons is considerable. This is important in practical radiation protection. Higher photon energies above 2 MeV occur, for instance, at nuclear reactors, the $^{16}O(n,p)^{16}N$ reaction where ^{16}N emits photons of 6.13 and 7.12 MeV is an example of this.

It is interesting to note in this context that calculations by /4/ for the ICRU phantom in air indicate that H_{10} is then a conservative estimate of H_{eff}-(AP). This is due to the fact that H_{10} then will increase to approximately $H_{10,k}$ since the CPB takes place in air.

NEUTRON RADIATION

A study of CPB for neutrons is given in /5/. There, however, CPB has been examined using a slab phantom and only a surface layer of 5 mm has been examined. In table 2 the ratios of H_d to $H_{d,k}$ are intercompared. The values in /5/ have been read from a graph which displays smoothed curves. Our values /6/ have not been smoothed and contain calculational errors of up to 7 %. With the exception of the values for 0.07 mm at 5 MeV, and for 3 mm at 20 MeV, the agreement is quite satisfactory. The influence of CPB is considerable only for d = 0.07 mm.

The quantity H_d also depends on the direction of the incident neutrons (see e.g. /10/). It is interesting to study the influence of CPB on the "ear effect" (i.e. the fact that for certain incident radiations the maximum of the dose equivalent distribution within the ICRU sphere occurs at 90° instead of 0°, see e.g. /4/). The magnitude of CPB is again studied by comparing H_d with $H_{d,k}$. The effect which is most pronounced at d = 0.07 mm is shown in table 3. The data clearly exhibit an "ear effect" which is also observed for photons /4/. The sharp drop for angles above 90° (back half of sphere) is typical of neutrons and not seen in photon fields. The CPB increases with energy and decreases with angle. This effect was observed to be less pronounced for d = 3 mm and E_n = 14 and 20 MeV. Similar results given in /11/ do not show the ear effect since the subdivison of the phantom was not sufficiently fine.

These results given in tables 2 and 3 show that with the exception of 20 MeV neutrons, the CPB influences $H_{0,07}$ only. This CPB effect, however, would not disappear if the ICRU sphere were embedded in air, since the hydrogen content of even moist air is negligible as compared with that of the phantom which contains 10.1 % H by weight.

Returning to the question of the suitability of H_d as a secondarily limited quantiy, we give in /10/ a detailed comparison of H_{10} at zero degrees with H_{eff}-AP as a function of energy. From our data given in /10/ and /6/ it can be seen that H_d (for d = 0.07, 3 and 10 mm) is a conservative estimate of H_{eff}-AP for $E_n \geq 8$ keV, if the ICRU sphere is in vacuo. This result is practically

unaffected by embedding the sphere in air.

CONCLUSION

In calculating H_3 and H_{10} the CPB is not negligible for photon irradiation with energies $E_\gamma \geq 2$ MeV. This influence could be remedied e.g. by specifying greater depths (d ~ 2 - 4 cm) for those high energies. In calibrating dosimeters therefore CPB necessitates careful considerations. In practical applications measurements with and without build-up layers (1 - 3 g/cm^2) should be made and the higher reading be used.

The CPB in the ICRU phantom for neutron radiation is only of significance for d = 70 µm and energies $E_n \geq 5$ MeV, and for 3 and 10 mm at 20 MeV. In contrast to photons, the surrounding air would not remedy the problem. However, since all H_d (d = .07, 3 and 10 mm) for $E_n \geq 8$ keV are conservative estimates of H_{eff}-AP, no problem is caused by CPB.

References

/1/ Burlin, T.E.:"Practical Determination of Dose Equivalent", Proceedings of the 7th Int. Congress of Rad. Res. Amsterdam, July 3 - 8, 1983, Paper E 1-20, Martinus Nijnhoff, Amsterdam 1983

/2/ Wagner, S.R.: "Is the Dose Equivalent Index a Quantity to be Measured?", Proceedings of the 5th Congress of Int. Rad. Prot. Soc., Jerusalem, March 1980, Pergamon Press, Oxford (1980) p. 307 - 310

/3/ ICRP, Publication 26, Pergamon Press, Oxford, New York, Frankfurt 1977

/4/ Dimbylow, P.J. and Francis, T.M.: "The Effect of Photon Scatter and Consequent Electron Build-up in Air on the Calculation of Dose Equivalent Quantities in the ICRU Sphere for Photon Energies from .662 to 10 MeV", Phys. Med. Biol. 28 (1983) p. 817 - 828

/5/ Chen, S.Y. and Chilton, A.B.: "Depth-Dose Relationships Near the Skin Resulting from Parallel Beams of Fast Neutrons", Rad. Res. 77 (1979) p. 21 - 33

/6/ Hollnagel, R.; Jahr, R. and Siebert, B.R.L.: "Dosimetric Quantities in the ICRU Sphere for Neutron Irradiations with Energies between thermal and 20 MeV", Report FMRB-101, PTB, Braunschweig (1983)

/7/ Hohlfeld, K. and Großwendt, B.: "Conversion Factors for Determining Dose Equivalent Quantities from Absorbed Dose in Air for Photon Radiation", Rad. Prot. Dos. 1 (1982) p. 277 - 283

/8/ Kramer, R. and Drexler, G: "The Dose Equivalent Index (dei) as a Function of Angular Distribution of Photons", Health Phys. 38 (1980) p. 426 - 428

/9/ Kramer, R. and Drexler, G.: "On the Calculation of the Effective Dose Equivalent", Rad. Prot. Dos. 3 (1982) p. 13 - 24

/10/ Siebert, B.R.L.; Hollnagel, R. and Jahr, R.: "Comparative Study of Radiation Protection Quantities for Neutron", this conference

/11/ Chen, S.Y. and Chilton, A.B.: "Calculation of Fast Neutron Depth-Dose in the ICRU Standard Tissue Phantom and the Derivation of Neutron Fluence-to-Dose-Index Conversion Factors", Rad. Res. 78 (1979) p. 335 - 370

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Table 1
Intercomparison of various fluence to dose equivalent conversion factors (pSv cm^2) as function of photon energy E_γ (MeV); (a) corresponds to ref. /4/

E_γ	$H_{3,K}$	H_3	$\hat{H}_3$	$\hat{H}_3$(a)	$H_{10,K}$	H_{10}	$\hat{H}_{10}$	$\hat{H}_{10}$(a)	H_I	H_I(a)
.662	3.64	3.48	3.89	3.77	3.62	3.51	3.87	3.78	3.9	3.82
1.25	6.16	5.06	6.34	6.39	6.08	6.34	6.34	6.19	6.4	6.19
2.0	8.53	4.59	7.85	8.64	8.43	8.36	8.84	9.02	8.8	9.02
3.0	11.3	3.93	10.5	11.1	11.2	9.86	11.3	11.6	11.6	11.6
5.0	16.5	3.82	14.4	15.1	16.3	9.59	15.1	15.7	16.8	16.2
7.0	22.4	3.80	17.8	18.0	22.5	9.79	19.5	19.6	22.0	19.9
10.0	30.7	3.78	22.8	23.6	30.8	10.2	25.6	24.9	29.7	25.9
15.0	48.4	3.78	31.7	-	48.1	14.2	34.8	-	66.5	-

Table 2
Ratio H_d to its KERMA approximation as a function of neutron energy E_n (MeV) and depth d (mm) as given by /5/ and the authors /6/

d	Ref.	E_n /MeV			
mm		5	10	14	20
0.07	/5/	0.42	0.69	0.69	0.77
	/6/	0.67	0.69	0.73	0.71
3.0	/5/	1.00	1.00	1.00	0.97
	/6/	0.98	1.02	1.00	0.87
10.0	/6/	1.03	1.04	1.03	0.88

Table 3
Fluence to specified depth dose conversion factors (pSv cm^2), in kerma approximation and correct as function of neutron energy E_n (MeV) and angle (degree)

Angular Range	E_n/MeV							
	5		10		14		20	
	$H_{0.07,k}$	$H_{0.07}$	$H_{0.07,k}$	$H_{0.07}$	$H_{0.07,k}$	$H_{0.07}$	$H_{0.07,k}$	$H_{0.07}$
0°-10°	380	254	457	303	537	389	715	507
10°-20°	{ 386	{ 268	455	300	529	388	741	526
20°-30°			455	304	539	396	746	533
30°-50°	391	289	463	323	543	412	745	541
50°-70°	406	331	476	364	560	450	765	576
70°-90°	429	387	500	416	591	507	813	640
90°-110°	267	262	350	314	431	392	625	518

INVESTIGATION LEVELS OF RADIOISOTOPES IN THE BODY AND IN URINE FOLLOWING INHALATION OF RADIOACTIVE MATERIALS - CONSEQUENCES OF THE 1977 ICRP RECOMMENDATIONS

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This work is an extension (and update) of a study published by the authors in 1980 (1). In the earlier work investigation levels (IL) of radioisotopes in the body following ingestion of radioactive materials were calculated. These calculations were necessary due to the changes in the basic recommendations of the ICRP (2) and the introduction of the dose limitation system and the concept of the annual limit of intake (ALI). In this work similar calculations are presented for the investigation levels in the body and in urine following penetration of radioactive materials into the body via inhalation. The dosimetric model of the ICRP for the respiratory system is used as a basis for the calculations. This model is described in detail in ICRP Pub. 30 (3).

The activity of a radioelement in the body at any time t after intake by ingestion of a unit of activity is given by its retention R(t):

$$R(t) = F_1 \exp(-\lambda t) \sum_i A_i \sum_j B_{ij} \exp(-\lambda_{ij} t) \qquad (1)$$

where F_1 is the coefficient expressing the fraction of the intake transferred to the transfer compartment; A_i are coefficients expressing the fractions transferred from the transfer compartment to the i-th organ; B_{ij} are the coefficients of the linear combination of exponentials with decay constants λ_{ij} representing the retention in the i-th organ (3,4) and λ is the physical decay constant.

When radioactive material enters the human body via inhalation the retention function is much more complex. The reason for this is that the respiratory system acts as a source of material for the other organs over relatively long periods after intake. The rate of change of the amount of material in the body organs is governed in this case by input from the lungs and lymph nodes (directly or via the G.I. tract) in addition to the changes caused by physical decay and biological excretion (the only two factors which affect the retention function in the case of intake by ingestion).

The total body activity at time t after intake by inhalation can be expressed:

$$R(t) = R_B(t) + R_L(t) \qquad (2)$$

$R_B(t)$ is the activity in all body organs except the respiratory system and $R_L(t)$ is the activity in the respiratory system. The two components in Eq. (2) can be expressed in detail as follows:

$$R_B(t) = \sum_i A_i \sum_j B_{ij} V_{ij} \, e^{-\lambda t}$$

$$R_L(t) = \left\{ (0.25(1-f)e^{-\lambda_2 t} + \frac{FD_3\lambda_2}{\lambda_3-\lambda_2} (e^{-\lambda_2 t} - e^{-\lambda_3 t}) \right.$$

$$\left. + FD_4 \frac{\lambda_2}{\lambda-\lambda_2} (e^{-\lambda_2 t} - 1) \right\} e^{-\lambda t}$$

where

$$V_{ij} = FD_1 e^{-\lambda_{ij}t} + FD_2 \frac{\lambda_2}{\lambda_{ij}-\lambda_2}(e^{-\lambda_2 t} - e^{-\lambda_{ij}t})$$

$$+ FD_3\lambda_2\lambda_3 \left\{ \frac{e^{-\lambda_2 t}}{(\lambda_3-\lambda_2)(\lambda_{ij}-\lambda_2)} + \frac{e^{-\lambda_{ij}t}}{(\lambda_2-\lambda_{ij})(\lambda_3-\lambda_{ij})} \right.$$

$$\left. + \frac{e^{-\lambda_3 t}}{(\lambda_{ij}-\lambda_3)(\lambda_2-\lambda_3)} \right\}$$

$$FD_1 = 0.3(a + F_1 b) + 0.08(C + F_1 d) + 0.25F_1 f$$

$$FD_2 = 0.25(e + F_1 g)$$

$$FD_3 = 0.25\ hi$$

$$FD_4 = 0.25\ hj$$

The symbols a - j are the conventional notations used by the ICRP (in its lung model description)for the relative uptake of the material in the different compartments of the respiratory system[3]. FD_i (i=1,2,3,4) are coefficients used to describe the fractions of the material removed from the respiratory system by different modes: FD_1 is the part that is removed from the lungs within the first 24 hours, FD_2 is the part removed with a decay constant λ_2, FD_3 is the fraction that is transferred to body fluids from the lung via the lymph nodes with a decay constant λ_2 for the transfer from compartment h to the lymph nodes and λ_3 for transfer from the lymph nodes to the body fluids; FD_4 is the part which stays in the lymph nodes practically infinitely.

The amount of activity U(t) excreted in the urine at any time t after the intake of a unit of activity is given by the first derivative of the biological retention function, multiplied by F_u the fraction of the excretion that is excreted through the urine[5,6]:

$$U(t) = F_u \exp(-\lambda t) \sum_i A_i \sum_j \lambda_{ij} B_{ij} V_{ij}$$

When the decay constants λ are expressed in $days^{-1}$ then U(t) is the daily excretion. The average daily urine volume is 1.4 liters (5); thus devision by 1.4 yields the concentration of the radioelement per liter.

The investigation level at any time t after intake was defined as the concentration of activity in the urine arising from an intake of 1/20 of an ALI (6). An analogous definition is used here for the total body investigation level.

A computer code was written which includes the clearance parameters from the respiratory system to the body fluids directly or via the GI-tract. The computer code receives as input for each isotope the ALI value, parameters related to transfer of the material

from the respiratory system and GI tract to the various organs (via the body transfer compartment), retention times in the organs and the fraction excreted through the urine. The program uses these data to calculate investigation levels of the radioisotopes in the body and in urine. Tables of these levels for different times after inhalation are then printed out. IL values for some selected radioisotopes calculated by the program are presented in Tables 1 and 2.

TABLE 1: Total body investigation levels as a function of time after inhalation

Isotope	Class*	Compound	A**	Investigation level (μCi) days after inhalation			
				2	7	30	60
^{22}Na		-	TB	25	19	6.6	1.6
^{67}Ga	W	Oxides and	TB	73	22	0.12	1.3×10^{-4}
		hydroxides	L	52	17	0.09	1.1×10^{-4}
	D	Others	TB	180	44	0.15	1.4×10^{-4}
^{75}Se	W	Element	TB	5.5	4.7	3.0	1.8
			L	3.9	3.6	2.3	1.3
^{125}I	D	-	TB	0.52	0.46	0.32	0.2
^{131}I	D	-	TB	0.22	0.14	0.017	0.001
^{239}Pu	Y	Dioxide	TB	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}
			L	1.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	1.1×10^{-4}
	W	Others	TB	1.1×10^{-4}	1.1×10^{-4}	9.6×10^{-5}	8.3×10^{-5}
			L	7.9×10^{-5}	7.4×10^{-5}	5.6×10^{-5}	3.8×10^{-5}
^{241}Am	W	-	TB	5.7×10^{-5}	5.5×10^{-5}	4.8×10^{-5}	4.2×10^{-5}
			L	4.0×10^{-5}	3.7×10^{-5}	2.8×10^{-5}	1.9×10^{-5}

*ICRP classes of retention in the lungs: D = Days, W = Weeks, Y = Years

**Investigation levels: Total body = TB, part in lung = L

TABLE 2: Investigation level in urine as a function of time after inhalation

Isotope	Class	Compound	Investigation level (μCi/ℓ) days after inhalation			
			2	7	30	60
^{3}H		HTO	87	62	12	1.6
^{32}P	W	Some phosphates	0.23	0.054	6.4×10^{-3}	6.8×10^{-4}
	D	Most phosphates	0.83	0.18	0.017	1.3×10^{-3}
^{35}S	W	Element	1.8	0.17	0.084	0.038
^{90}Sr	Y	Titanate	2.7×10^{-5}	5.7×10^{-6}	7.7×10^{-7}	5.8×10^{-7}
^{125}I	D		0.023	0.001	6.8×10^{-4}	4.2×10^{-4}
^{131}I	D		0.01	2.9×10^{-4}	3.6×10^{-5}	2.3×10^{-6}
^{238}U	Y	Oxides	1.4×10^{-7}	3.7×10^{-8}	1.2×10^{-8}	8.1×10^{-9}

*ICRP classes of retention in the lungs: D = Days, W = Weeks, Y = Years.

References

1. Y. Shamai, M. Tirkel and T. Schlesinger, 5th International IRPA Congress, Jerusalem, 1980, Vol. 1, p. 197.
2. Annals of the ICRP, Publ. No. 26, 1977, Vol. 1.
3. Annals of the ICRP, Publ. No. 30, 1979, Vol. 2.
4. N. Adams, B.W. Hunt and J.A. Reissland, NRPB R-82,1978.
5. Annals of the ICRP, Publ. No. 23, 1974.
6. Annals of the ICRP, Publ. No. 10, 1968.

VORTEIL EINES BRAGG-MONOCHROMATORS ZUR BESTIMMUNG DER ENERGIEABHÄNGIGKEIT VON DOSIMETERN

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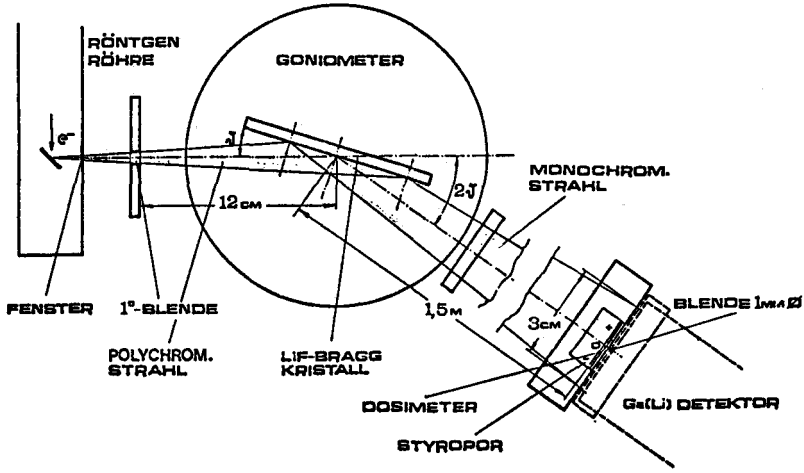
Die Bestimmung der Energieabhängigkeit jeder Art von Dosimetern ist eine der wichtigsten Messungen der experimentellen Dosimetrie /1/. Diese Messungen werden in der Regel mit gefilterter Bremsstrahlung durchgeführt, wobei das Bremsstrahlungsspektrum durch die Röntgenspannung, der Filterung, der Halbwertsschichtdichte und dem Homogenitätskoeffizienten definiert wird /2,3/.

Der Photoeffekt als dominierender Prozeß im Energiebereich < 50 keV bedingt eine starke Abhängigkeit des Dosimeteransprechvermögens von der effektiven Ordnungszahl des Dosimetermaterials. Die durch die Filterung des Bremsspektrums erzeugte "monochromatische" Strahlung zeigt eine breite Energieverteilung für den Bereich maximaler Intensität und einen nicht zu vernachlässigenden Anteil niederenergetischer Photonen /2/. Um diese inherenten Fehler zu vermeiden, wurde eine Kalibriermethode mit Bragg-Monochromator entwickelt /4/.

Die Abbildung zeigt die schematische Darstellung der entwickelten Kalibriereinrichtung. Der Röntgenstrahl einer Röhre mit W-Anode (10 - 60 kV) wird durch eine Blende definiert und an einem LiF-Kristall um einen Winkel ϑ Bragg reflektiert. Nach der Bragg-Reflexion sind die Röntgenstrahlen mit einer Halbwertsbreite von $\sim$0,5 keV monochromatisiert.



Schematische Darstellung der Kalibriereinrichtung zur Bestimmung der Energieabhängigkeit von Dosimetern.

Die Energie E der Strahlung und der zur Monochromatisierung einzustellende Bragg-Winkel ϑ sind durch die Beziehung

$$\vartheta = \arcsin \frac{nhc}{2dE}$$

verknüpft, wobei d der Netzebenenabstand des Monochromatorkristalls und n die Ordnung der Beugung ist.

Mit der in /4/ beschriebenen Anordnung ist es möglich, mit Hilfe eines Efficiency-kalibrierten Ge(Li)-Detektors /5/ die Energieverteilung und den Photonenfluß am Ort des zu kalibrierenden Dosimeters zu bestimmen. Aus diesen Werten wird die absorbierte Dosis im entsprechenden Dosimetermaterial bestimmt /6,4,7/. Da auch bei dieser Methode ein niederenergetischer Photonenfluß zur vom Dosimeter registrierten Anzeige beiträgt, wurde ein Verfahren entwickelt, welches die Abschätzung dieses Beitrages ermöglicht. Dazu wurde eine Spektralzerlegung des niederenergetischen Flußes im Energiebereich von 5 keV Breite vorgenommen und die jeweiligen Beiträge zur Dosimeteranzeige abgeschätzt. Diese Methode wurde für verschiedene TL-Dosimeter angewendet.

Für das stark energieabhängige $CaSO_4$ betrug der systematische Gesamtfehler der Dosisbestimmung aus dieser Fehlerquelle und damit die Korrektur 6,5%. Es wurde aus dem in diesem Beitrag gewonnenen Photonenfluß, unter Berücksichtigung der für das entsprechende Dosimetermaterial in diesem Energiebereich gültigen Ansprechfunktion, der Beitrag zur Dosimeterenergie abgeschätzt.

LITERATUR

/1/ K.Becker: Solid State Dosimetry, CRC Press, Cleveland, 1960

/2/ W.W.Seelentag, W.Panzer, G.Drexler, L.Platz and F.Santner: A Catalogue of Spectra for the Calibration of Dosimeters, GSF-Bericht 560/1979

/3/ ISO 4037/DAD1: X and γ reference radiations for calibrating dosimeters and dose ratemeters and for determining their response as a function of photon energy

4/ N.Vana and H.Aiginger: Acta Phys.Acad. Scientiarum Hung. 52, (1982) 333

/5/ E.Unfried: Calibration of Semiconductor-Detectors, Thesis Technical University of Vienna, 1977 (in Deutsch)

/6/ E.Strom and D.W.Lier: Health Phys. 23 (1972) 73

/7/ N.Vana, H.Aiginger, W.Erath and T.Michev: Acta Phys. Acad. Scientiarum Hung. 52 (1982) 341

ASSESSMENT OF ABSORBED DOSE IN THE FEMALE BREAST FROM A SOURCE IN THE LUNGS.

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INTRODUCTION

Because of it's radiosensitivity, the breast is one of the organs that have recieved it's own high weighting factor for the calculation of the effective dose equivalent (1). At present the most common way to calculate absorbed doses from internal sources is to follow the MIRD-formalism, using "S-values" (absorbed dose per unit cumulated activity) or absorbed fractions (2,3). These data give, however, no information useful for calculation of the absorbed dose to the breast from a radionuclide source in another organ. For the calculation of the effective dose equivalent, the absorbed dose to the breast is usually approximated with the absorbed dose to "other tissue" or muscle. In most cases of internal irradiation it can be assumed that this approximation slightly overestimates the effecitve dose equivalent. This is, however, not true for the lungs who are situated close to the breast.

The purpose of this work is to see to what degree the assumption, that the breast recieve the same absorbed dose as "other tissue", underestimates the effective dose equivalent, when the radiation originates mainly form the lungs. Data useful for making a more accurate assesment of the absorbed dose in the breast, from a radionuclide depositied in the lung, will also be presented.

CALCULATIONS

Monte-Carlo calculations, simulating photon transport, has been compared with measurements of absorbed dose in a phantom using TL-dosimeters.

ICRP publication 23, reference man (4), reports typical dimensions and mass of the female breast, see Table 1. However, the mass of the breast as given in ICRP23, and the mass based on a calculation using the dimensions from the same report, differ by around a factor of two. The size of the adult female breast is, as most other organs, varying greatly throughout the world, and an variation with age may also be seen. This is probably the explanation of the huge variation, by a factor of four, that can be seen in the literature (4-13). I have chosen to stay to the lower limit of the dimension as given in ICRP23 (4), and to use the mass calculated for this size. The influence on the absorbed dose from the breast size, at interal irradiation, was found to be very low in most cases, this has also been studied by Cristy (13).

The female breast can be represented by portions of two ellipsoids attached to the trunk, which is described by MIRD (3). The transverse radius of the ellipsoid is set to 6 cm, the cephalocaudal: 5 cm, and the protrusion from the chest wall: 5 cm. The central point of the breast is found at the height of the fifth rib (8). Based on the MIRD reference skeleton (3) this leads to a Z-coordinate of 55, in the MIRD coordinate system. The midpoints are also assumed to be half-way out to the sides from the central axis. At this point the Y-coordinate for the surface of the trunk is -8.7. The equation for the left breast can be written:

$$\left(\frac{X-10}{6}\right)^2 + \left(\frac{Y+8.7}{5}\right)^2 + \left(\frac{Z-55}{5}\right)^2 < 1$$

For the right breast the "X-10" is changed to "X+10". The portion of the ellipsoid that is inside the trunk must be subtracted. The volume of one breasts can be calculated to

325 cm^3. The lung, which is the source organ is described in MIRD, as a half ellipsoid with an anterior section removed (3).

The trunk of the reference man is a male torso, a female has smaller dimensions. The organs will therefore in a female be closer to the breast. Use of a male torso might therefore lead to a slight underestimation of the absorbed dose in the female breast. For source organs situated higher or lower than the breast, this can be compensated for by increase or decrease the Z-coordinates of the breast a few centimeters. The lungs have their center of mass at Z=52.3, a three cm lowering of the breast, will in this case increase the calculated absorbed dose in the breast with in general less than 10%. This small change in the result is negligible, compared to the relatively great variation in absorbed dose among individuals due to form, size, etc. of the breast and body. An error is also introduced since the equations only approximately can describe an average individual. I have therefore chosen not do decrease the Z-value, and the calculations are performed with the center of the breast situated at Z=55.

A Monte-Carlo program has been developed for simulating photon transport. The mathematical phantom, comprises a MIRD trunk with lungs included, and breast, as described above, attached. The program choses a start point within the lungs, and a direction, at random. It traces all scattering points of the photon until it is absorbed, or escapes from the phantom. Attenuation coefficients has been calculated according to White and Fitzgerald (14). The density of the lung is 0.296 g/cm^3 (3). In order to speed up computer calculations no regard has been taken to the skeleton, which has a higher density than the rest of the body. This approximation will increase the absorbed dose from low energy photons with around 10%. The random number generator of the computer was not reliable enough, therefore generator found in the NAG program library was utilized (15).

EXPERIMENTS

Phantoms of trunk, lung, and breast was produced. The absorbed dose at different points within and around the breast phantom was measured with 1 mm x 5 mm (diam.) LiF-teflon dosimeters. The measurements were performed with 20-30 GBq of Tc-99m in the left lung filled with water or sawdust (density: 0.16 g/cm^3). Sawdust was chosen as phantom material in order to investigate the effect of the low density of the lung on the absorbed dose to the breast. In this case the activity was introduced at several scattered points within the lung, in order to resemble a homogeneous distribution. In the case of water the distribution of the radionuclide was checked with a scintillation camera, and, except for the first few minutes after introduction, no inhomogeneity could be seen. There were some slight deviation between the experimental and the mathematical breast phantom, therefore, for comparison, separate Monte-Carlo calclulations were made with a modified mathematical phantom, with a density of the lung equal that of water or sawdust.

RESULTS AND DISCUSSION

The results of the calculataions are presented in Table 1 in the form of specific absorbed fractions for different energies. The coefficient of variation in this table varies between 3 and 6%. In Table 2 the absorbed dose per unit cumulated activity, S-values, for some radionuclides of interest in nuclear medicine lung investigations, are presented. From this table it can be seen that the absorbed dose to the breast from a source in the lungs, is in general a factor of two greater than that to "other tissue" or muscle. The error in the effective dose equivalent, due to the use of the absorbed dose in "other tissue" instead of breast is, however, insignificant. In the most extreme case, that is, when the source is solely located within the lungs, the underestimation of the effective dose equivalent will, in practically all cases be less than 5%, and is

TABLE 1. Specific absorbed fractions for the breast with a source in the lung.

Energy (MeV)	0.020	0.030	0.050	0.100	0.200
SAF (g^{-1})	4.00×10^{-6}	1.53×10^{-5}	1.45×10^{-5}	1.00×10^{-5}	9.66×10^{-6}

Energy (MeV)	0.500	1.000	1.500	2.000
SAF (g^{-1})	9.74×10^{-6}	9.46×10^{-6}	8.82×10^{-6}	8.15×10^{-6}

TABLE 2. S-values in Gy/decay (rad/μCi-h), with a source in the lungs.

	Target				
Nuclide	Breast (this work)		Other tissue, MIRD11 (2)		Breast / 0.tissue
^{11}C	1.6×10^{-15}	(2.1×10^{-5})	7.5×10^{-16}	(1.0×10^{-5})	2.1
^{15}O	1.6×10^{-15}	(2.1×10^{-5})	7.5×10^{-16}	(1.0×10^{-5})	2.1
$^{81}Kr^{m}$	2.0×10^{-16}	(2.7×10^{-6})	9.8×10^{-17}	(1.3×10^{-6})	2.0
$^{99}Tc^{m}$	2.0×10^{-16}	(2.6×10^{-6})	9.8×10^{-17}	(1.3×10^{-6})	2.0
^{111}In	6.2×10^{-16}	(8.3×10^{-6})	3.2×10^{-16}	(4.2×10^{-6})	1.9
$^{113}In^{m}$	4.1×10^{-16}	(5.4×10^{-6})	1.9×10^{-16}	(2.5×10^{-6})	2.2
^{123}I	2.9×10^{-16}	(3.8×10^{-6})	1.5×10^{-16}	(2.0×10^{-6})	1.9
^{131}I	5.9×10^{-16}	(7.9×10^{-6})	2.8×10^{-16}	(3.8×10^{-6})	2.1
^{127}Xe	4.5×10^{-16}	(6.0×10^{-6})	2.3×10^{-16}	(3.1×10^{-6})	2.0
^{133}Xe	9.1×10^{-17}	(1.2×10^{-6})	5.1×10^{-17}	(6.8×10^{-7})	1.8

in most cases less than 2%. The reason for this is that the contribution to the effective dose equivalent from the absorbed dose in the lung itself is overwhelming, and that the lung also has about the same, relatively high, weighting factor, 0.12, compared to 0.15 for the breast (1). Other, relative to the breast, close located organs are the liver and the thyroid. Monte-Carlo calculations have been performed also with radionuclide sources in these organs. For the liver as a source, is was found that the absorbed dose to breast is of the same order as that to "other tissue", and for the thyroid as a source, the absorbed dose to "other tissue" was found to be higher.

The result of the LiF-measurements, and the comparing Monte-Carlo calculations, are presented in figure 1. As can be seen, the calculated and measured values are relatively well in accordance with each other. In the measurement, no significant difference, in the absorbed dose to the breast, could be observed between the "sawdust-lung" and the "water-lung". For the left breast, however, a difference can be seen in the calculated mean absorbed dose between the both cases. There might be several reasons for this discrepancy, except for measuring errors, mainly due to the positioning of the TLD, one reason may be that, also in the case of sawdust, interaction cross section for water was used in the calculation.

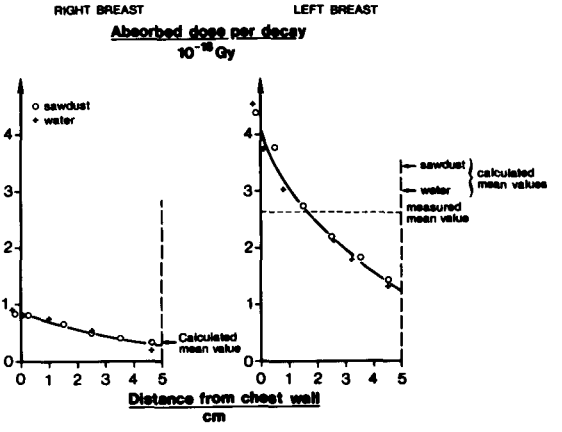

Figure 1. Result of phantom measurements of the absorbed dose at different points within the left and right breast presented in units of 10^{-16} Gy. The radionuclide source, $^{99}Tc^{m}$, is located in the left lung, which is filled with water or sawdust.

REFERENCES

1. International Commission on Radiological Protection: Recommendations of the ICRP. Annals of the ICRP Vol. 1, No. 3, 1977, ICRP Publication 26
2. Snyder WS, Ford MR, Warner GG, et al.: "S" absorbed dose per unit cumulated activity for selected radionuclides and organs. MIRD Pamphlet No. 11, Society of Nuclear Medicine, New York, 1975
3. Snyder WS, Ford MR, Warner GG: "Estimates of specific absorbed fractions for photon sources uniformly distributed in various organs of a heterogenous phantom", MIRD pamphlet no 5, revised. Society of Nuclear Medicine, New York, 1978.
4. International Commission on Radiological Protection, Report of the Task Group on Reference Man, ICRP Publication 23, Pergamon Press, Oxford, 1975.
5. Cristy M: Health Phys. 43 (1982) 930-932
6. Anson BJ (ed.): Morris Human Anatomy, 12:th ed., McGraw-Hill, New York, 1966.
7. Goss CM (Ed): Anatomy of the Human Body, 27:th ed., Lea & Febiger, Philadelphia, 1959.
8. Hytten FE, Leitch I: The Physiology of Human Pregnancy, F A Davis, Philadelphia, 1964.
9. Katariya RN, Forrest APM, Gravelle IH: Br. J. Cancer 29, (1974) 270-273
10. Rehman I: in: Gallager HS, Leis,Jr. HP, Snyderman RK et al. (Eds.): The Breast, pp 3-22, The C. V. Mosby Company, Saint Louis 1978.
11. Katch VL, Campaigne BC, Freedson P, et al.: Am. J. Phys. Antropol. 53 (1980) 93-100
12. Kramer R, Williams G, Drexler G: Health Phys. 43 (1982) 932-935
13. Cristy M: "Calculation of annual limits of intake of radionuclides by workers: significance of breast as an explicitly represented tissue" (submitted to Health Physics)
14. White DR, Fitzgerald M: Health Phys. 33 (1977) 73-81
15. Numerical Algorithms Group, NAG Central Office, Mayfield House, 256 Banbury Road, Oxford OX2 7DE, England.

COMPUTATIONAL METHOD FOR REALISTIC ESTIMATES OF THE DOSE TO ACTIVE MARROW*

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INTRODUCTION

Calculation of absorbed dose to active marrow from photon radiation is a complex problem because electronic equilibrium may not exist in the vicinity of soft tissue–bone mineral interfaces.[1–5] Snyder et al.[6] recognized the intractable geometry of trabecular bone in their studies of photon transport in the body and formulated marrow dose estimates in a conservative manner. Other investigators[7–10] have noted that this approach leads to overestimates by factors of 3 or more at low photon energy. In this paper the absorbed dose is formulated in terms of physical and anatomical parameters defining the energy deposition in the marrow space.

ABSORBED DOSE FORMULATION

Consider the trabeculation of a bone experiencing a fluence, $\Psi(E)$, of photons of energy E. Let m(TB) and m(RM) denote the mass of bone (trabeculae) and marrow comprising the trabeculation. If we index the type of photon interaction by i and the region in which it occurred by r, r = TB or RM, then the absorbed dose in active marrow per unit photon fluence, $D(RM)/\Psi(E)$, can be expressed as

$$\frac{D(RM)}{\Psi(E)} = \sum_r \frac{m(r)}{m(RM)} \sum_i \int_0^\infty \phi(RM \leftarrow r, T_i)\ (i/\rho)_r\ n_r(T_i)\ T_i\ dT_i \quad , \tag{1}$$

where

$\phi(RM \leftarrow r, T_i)$ is the absorbed fraction in RM from r for electrons of energy T_i,

$(i/\rho)_r$, i = τ, σ, and k, denotes the mass attenuation coefficients in medium r for the photoelectric, Compton, and pair-production interactions, respectively,

$n_r(T_i)dT_i$ denotes the number of electrons of energy between T_i and $T_i + dT_i$ liberated in region r per interaction i.

This formulation separates the energy transfer process from the process of energy dissipation by secondary electrons. With this approach the mathematical analogue of man,[6] with its homogeneous skeleton, can be retained in photon transport calculations and the energy dissipation can be addressed on a microscopic scale. The energy dissipation is embodied in the absorbed fraction quantity.

ENERGY DISTRIBUTION OF SECONDARY ELECTRONS

Photons transfer energy to electrons through three major interactions: the photoelectric effect, the Compton effect, and pair-production. Photon cross-section data of Hubbell[11] and elemental composition data of Kerr[12] were used in evaluating the energy transfer. Photoelectrons were assumed to be of discrete energy corresponding to the incident photon energy. The energy distribution of Compton electrons was calculated from the Klein-Nishina relationship,[13] and the positron-electron energy distribution was derived from the Bethe-Heitler theory of pair-production.[14]

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ABSORBED FRACTIONS FOR MONOENERGETIC ELECTRONS

Because the geometry of trabecular bone could not be described in simple terms, Spiers introduced a method of calculating energy deposition using the path-lengths traversed by particles.[15] These path-lengths are based on chord-length distributions for trabeculae and marrow cavities obtained by optically scanning the trabeculation.[16] Absorbed fraction data for monoenergetic electrons, as required in Eq. (1), were computed as outlined by Whitwell and Spiers.[17] Data for the parietal bone and lumbar vertebra of the skeleton of a 44 year-old male are shown in Fig. 1. Dose factor quantities for beta emitters calculated from our monoenergetic absorbed fraction data were in excellent agreement with values reported by Whitwell and Spier.[17]

RESULTS AND DISCUSSION

The complete results of our calculations for various bones of the skeleton are given in Table 1. These data can be applied to photon fluence estimates derived from Monte Carlo transport calculations in mathematical analogues of the body to estimate absorbed dose. Variations with incident photon energy in the ratio of absorbed dose in active marrow to the equilibrium dose (kerma) in soft-tissue are indicated in Fig. 2. These ratios are maximal at photon energies in the region of 50 to 60 keV and are higher for the thick trabeculae and small marrow cavities of the parietal bone than for the thinner trabeculae-larger marrow cavities of other bones. The ratios at low energy conform to the general features indicated by Spiers.[5] However the parietal bone exhibits a substantially higher enhancement of the marrow dose than other trabecular bones. This enhancement should be considered in deriving skeletal average values for the diagnostic x-ray region. Enhancement of dose in the high energy (pair production) region is also indicated in our calculations. Enhancement is small, about 5%, for most trabecular sites but approaches 20% for the parietal bone. Considering the highly stylized analogue of the skeleton used in photon transport calculations, we recommend that the skull be treated as a separate bone region and data for the parietal bone in Table 1 be applied to estimate marrow dose. The lumbar vertebra appears to be representative of other trabecular sites.

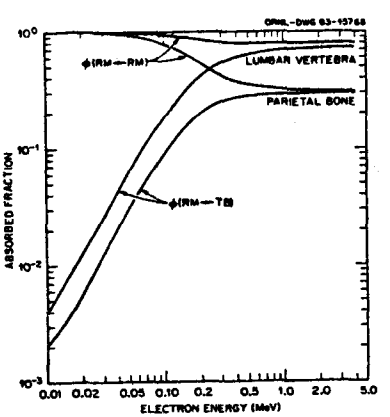

Figure 1.

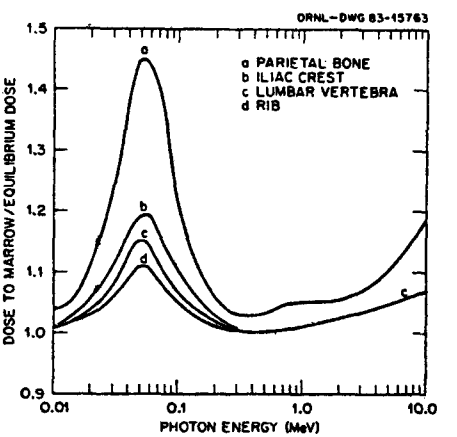

Figure 2.

Table 1. Absorbed dose in active marrow, D(RM), per unit fluence, $\Psi(E)$, of monoenergetic photons in trabecular bones of the skeleton of a 44-year-old male.

Photon energy (MeV)	D(RM)/Ψ(E), Gy per photon/m^2						
	Parietal bone	Cervical vertebra	Lumbar vertebra	Rib	Iliac crest	Head of femur	Neck of femur
0.010	6.30e-16	6.15e-16	6.14e-16	6.12e-16	6.16e-16	6.12e-16	6.12e-16
0.015	2.71e-16	2.62e-16	2.61e-16	2.59e-16	2.63e-16	2.60e-16	2.59e-16
0.020	1.53e-16	1.45e-16	1.43e-16	1.41e-16	1.45e-16	1.42e-16	1.41e-16
0.030	7.49e-17	6.60e-17	6.44e-17	6.29e-17	6.61e-17	6.39e-17	6.31e-17
0.040	5.04e-17	4.27e-17	4.11e-17	3.99e-17	4.28e-17	4.08e-17	3.99e-17
0.050	4.18e-17	3.45e-17	3.31e-17	3.20e-17	3.45e-17	3.27e-17	3.21e-17
0.060	3.93e-17	3.26e-17	3.11e-17	3.01e-17	3.24e-17	3.08e-17	3.01e-17
0.080	4.15e-17	3.58e-17	3.45e-17	3.36e-17	3.57e-17	3.44e-17	3.37e-17
0.10	4.79e-17	4.33e-17	4.22e-17	4.14e-17	4.33e-17	4.21e-17	4.15e-17
0.15	7.16e-17	6.83e-17	6.74e-17	6.68e-17	6.83e-17	6.72e-17	6.70e-17
0.20	9.88e-17	9.63e-17	9.57e-17	9.52e-17	9.64e-17	9.53e-17	9.53e-17
0.30	1.57e-16	1.54e-16	1.54e-16	1.53e-16	1.54e-16	1.52e-16	1.53e-16
0.40	2.15e-16	2.12e-16	2.10e-16	2.10e-16	2.12e-16	2.07e-16	2.10e-16
0.50	2.72e-16	2.67e-16	2.66e-16	2.65e-16	2.68e-16	2.60e-16	2.65e-16
0.60	3.28e-16	3.20e-16	3.19e-16	3.17e-16	3.20e-16	3.10e-16	3.18e-16
0.80	4.28e-16	4.17e-16	4.15e-16	4.14e-16	4.17e-16	4.04e-16	4.14e-16
1.0	5.19e-16	5.06e-16	5.03e-16	5.01e-16	5.06e-16	4.88e-16	5.02e-16
1.5	7.12e-16	6.95e-16	6.90e-16	6.88e-16	6.94e-16	6.69e-16	6.90e-16
2.0	8.77e-16	8.52e-16	8.45e-16	8.43e-16	8.50e-16	8.19e-16	8.45e-16
3.0	1.16e-15	1.11e-15	1.10e-15	1.09e-15	1.11e-15	1.06e-15	1.10e-15
4.0	1.41e-15	1.33e-15	1.31e-15	1.29e-15	1.32e-15	1.26e-15	1.30e-15
5.0	1.64e-15	1.52e-15	1.49e-15	1.46e-15	1.51e-15	1.43e-15	1.48e-15
6.0	1.87e-15	1.69e-15	1.65e-15	1.62e-15	1.68e-15	1.58e-15	1.64e-15
8.0	2.33e-15	2.02e-15	1.94e-15	1.89e-15	1.99e-15	1.86e-15	1.93e-15
10.0	2.79e-15	2.32e-15	2.21e-15	2.14e-15	2.29e-15	2.10e-15	2.19e-15

REFERENCES

1. F. W. Spiers, Brit. J. Radiol. 12, 521 (1949).
2. F. W. Spiers, Brit. J. Radiol. 26, 38 (1953).
3. D. E. Charlton and D. V. Cormack, Radiat. Res., 17, 34 (1962).
4. J. L. Howarth, Radiat. Res. 24, 158 (1965).
5. F. W. Spiers, "Transition-zone dosimetry," in Radiation Dosimetry Vol. 3 (Academic Press), ed. F. H. Attix and E. Tochlin, 809-867 (1969).
6. W. S. Snyder, M. R. Ford, and G. G. Warner, "Estimates of Specific Absorbed Fractions For Photon Sources Uniformly Distributed In Various Organs of a Heterogeneous Phantom, Medical Internal Radiation Dose Committee, Pamphlet No. 5, Revised (Society of Nuclear Medicine, New York; 1978).
7. M. Rosenstein, Organ Doses in Diagnostic Radiology, U.S. Department of Health, Education and Welfare Report FDA 76-8030 (Washington, DC: U.S. Government Printing Office: 1976)
8. R. Kramer, Ermittlung von Konversionsfaktoren Zwischen Korperdosen und Relevanten Strahlungskenngrossen bei Externer Rontgen- und Gamma-Bestrahlung, Gesellschaft fur Strahlen-und Umweltforschung GSF-Bericht-S-566 (1979).
9. T. Ashton and F. W. Spiers, Phys. Med. Biol. 24, 950 (1979).
10. G. D. Kerr, Health Phys. 39, 3 (1980).
11. J. H. Hubbell, Int. J. Appl. Radiat. Isot. 33, 1269 (1982).
12. G. D. Kerr, Photon and Neutron-to-Kerma Conversion Factors for ICRP-1975 Reference Man Using Improved Element Compositions for Bone and Marrow of the Skeleton, Oak Ridge National Laboratory Report ORNL/TM-8318 (1982).
13. R. D. Evans, "X-Ray and γ-ray interactions," Radiation Dosimetry (edited by F. H. Attix and W. C. Roesch) Vol 1, pp. 93-155 (Academic Press; New York, N. Y.; 1968).
14. W. Heitler, The Quantum Theory of Radiation, 3rd edition, Oxford Press, London, 1964.
15. F. W. Spiers, "Beta dosimetry in trabecular bone," Delayed Effects of Bone-Seeking Radionuclides (edited by C. W. Mays et al.) pp 95-108 (U. of Utah Press; Salt Lake City, UT 1969).
16. A. H. Beddoe, P. J. Darley, and F. W. Spiers, Phys. Med. Biol. 21, 589 (1976).
17. J. R. Whitwell and F. W. Spiers, Phys. Med. Biol. 21, 16 (1976).

INTERNAL DOSIMETRY MONITORING AT THE CHALK RIVER NUCLEAR LABORATORIES

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INTRODUCTION

Employees at the Chalk River Nuclear Laboratories (CRNL) operate and maintain high flux reactors, zero power research reactors, radioisotope production facilities, accelerators, experimental fuel laboratories and waste management facilities. In addition much of the research and development carried out at CRNL involves work with unsealed radioactive materials. Internal contamination may occur during the course of this work, and monitoring for internal contamination (internal dosimetry monitoring) is a necessary complement to workplace monitoring to ensure employees are not contaminated, or if they are, to quantify the doses associated with these contaminations so that appropriate actions can be taken. Currently approximately one half of CRNL's 2400 employees are monitored by at least one method each year. This paper briefly reviews the process of selecting the employees to be monitored, the monitoring techniques used, and the reporting, recording and interpretation of monitoring results.

SELECTION OF EMPLOYEES TO BE MONITORED

Employees are selected for internal dosimetry monitoring by their supervisory staff in consultation with staff from the Radiation and Industrial Safety (R&IS) Branch. The selection is made after reviewing such factors as the amount and form of activity handled. The results of this review are used to judge whether intakes by individuals in a year could exceed a significant fraction of the annual limit on intake (ALI), usually taken as 0.1 ALI. This judgement of what intakes could be is very subjective and hence conservative assumptions are used. It is therefore very unlikely that an activity exceeding 0.1 ALI would be taken in by an employee who has not been selected to take part in the internal dosimetry monitoring program.

The routine monitoring frequency depends on the likelihood of contamination, the derived investigation level (DIL - see below), and the sensitivity of the monitoring technique being used. Typical frequencies used are bi-weekly, monthly and bi-monthly for tritium and fission products in urine; semi-annually, annually and bi-annually for fission products in vivo; monthly and bi-monthly for uranium, thorium and plutonium in urine; and semi-annually and annually for uranium, thorium and plutonium in vivo. In addition to this routine monitoring, operational and special monitoring is performed as required.

The names and locations of each employee selected to take part in the monitoring program, along with the type of monitoring required, and its frequency are stored in a computer file (frequency file). A list of employees to be monitored in the following week is produced from this file each week, and once an employee's name appears on this list, it will continue to appear each week until the

requested monitoring has been performed, or the request is cancelled. Once a month, lists of employees' names with their current monitoring frequencies are sent to supervisors. This list is reviewed and any changes in monitoring requirements are used to update the frequency file.

MONITORING TECHNIQUES

Urinalysis Tritium concentrations in urine are measured with a standard liquid scintillation counter at a sensitivity better than 10^4 Bq L^{-1}. Calibration standards and quality control samples are analysed each day and the computer evaluated results are compared to the known activities. Once a month, or more often if changes in operation are made, a quench (calibration) curve is generated and compared to the previous quench curve. The urine tritium concentration results are automatically stored in the data base, from which they are available for subsequent dosimetry calculations.

The screening method (gross beta) that is used for beta emitting fission products in urine consists of an oxalate precipitation and counting in a standard planchet counter. This method will detect Sr, Pm, Ce and Ba (1) in urine at a level of about 60 mBq L^{-1}. The quality control on this method is performed by calibrating the counter each day, and by analysing a "blind" sample supplied by the Quality Control Manager (QCM) each week.

Gamma emitting fission products and other gamma emitting radionuclides are detected in urine in a Marinelli flask using a 5 x 5 cm NaI (Tl) detector inside a lead castle (1). The sensitivity of this method for a standard counting time of 800 seconds and 0.4 L of urine is better than 500 Bq (gamma) L^{-1} over the range of gamma energies encountered. The efficiency and energy calibration of the detector is checked daily.

A gross alpha analysis is done on urine from persons working with Th, Pu, Am, Np or other actinides (excluding uranium) (1). The prepared sample is counted using a ZnS screen/photomultiplier tube detector and if any activity is detected, an alpha spectrum using a Si surface barrier detector is taken to aid in the identification of the radionuclide(s) present. The sensitivity of this method is about 0.5 mBq per sample. The quality control on this method is achieved by analysing blind samples supplied by the QCM each week.

Analysis for uranium in urine is done either fluorometrically for natural uranium (2) or radiochemically (3). The fluorometric analyses are calibrated by spiking aliquots of each sample. The radiochemical procedure is checked periodically with blind samples prepared by the QCM.

In addition to the above analyses that are performed routinely, analytical techniques have been developed for Ni-63 (4); radium (5); S-35 and P-32 (6); Cl-36 (7); Tc-99 (8); and Ce-144/Pm-144, Sr-90 and Y-90 (9), for use as required.

In Vivo Monitoring Whole body monitoring is performed in a specially constructed low background building (10) with a shadow

shield monitor. This monitor has been calibrated using standard bottle phantoms filled with various radionuclides (10). The sensitivity of this system is about 40 Bq for Cs-137. The efficiency, energy calibration and background spectrum of the monitor are checked daily. All spectra are routed to the main CRNL computer for analysis and storage of spectra and results.

Routine thorax (lung) monitoring is carried out with phoswich detectors in the low background building (11). These detectors have been calibrated using a thorax phantom based on the well known Livermore phantom (12). These monitors will detect about 3 mg of U-Nat, 1 mg of Th-Nat and a variable amount of plutonium, depending on body build and plutonium burnup and time since separation (11).

Routine thyroid monitoring for I-125 (other radioiodines are measured with the whole body monitor) is performed with a 2.5 x 0.1 cm NaI (Tl) detector. In addition various other detectors (including a 20 cm diameter phoswich detector optimized to detect U-Nat and Th-Nat) are available for non-routine monitoring as required.

Thoron-in-Breath Monitoring A thoron-in-breath monitor has been recently developed (13) for monitoring employees working with thoria powders. Calibration of this monitoring technique is difficult and work is continuing on this problem, but its sensitivity is much superior to in vivo monitoring or urinalysis.

RECORDING AND REPORTING OF RESULTS

A computerized data base system has been constructed in order to efficiently record and report the approximately 12,000 monitoring results on approximately 1200 employees obtained each year and to allow easy retrieval of these results at a later date if required. The data base is constructed using the indexed sequential filing system available in the main CRNL computer. Employee identification and status (i.e. retirement, etc.) is kept up-to-date by access to the main personnel files. The programs used to access (input, retrieve, correct, etc.) records are interactive and contain various checks to help ensure that the monitoring results that go into the data base are correctly assigned to the right employee. A report of the monitoring results obtained each week is produced from the data base and sent to supervisory and R&IS staff. This report contains the results of each test, and whether any internal contamination measured by the test was judged to be at a negligible, minor, caution or removal level (see below). In addition, supervisory and R&IS staff are immediately informed if any result exceeds the caution or removal level. Each month, a summary report is produced that lists all minor, caution or removal levels for use by the Site Safety Committee during their monthly review. The only internal contamination that is routinely present at non-negligible levels is tritium. Doses associated with tritium contaminations are routinely calculated (14, 15) and are reported bi-weekly, quarterly and annually.

INTERPRETATION OF RESULTS

Except for tritium, very few monitoring results exceed the detection limit of the monitoring procedure. Those that do are compared to pre-derived levels so that further actions can be taken if necessary. The highest of these levels is the removal level and

when a result exceeds this level the employee is removed from sources of further significant contamination for evaluation. Nominally, the removal level (16) is based on an intake of 1/20 of the ALI for the radionuclide in question, calculated via GENMOD (17) with the assumption that the contamination occurred immediately after the previous measurement. This level is called the derived investigation level (DIL), and differs somewhat from that recommended by ICRP (18, 19). Actual removal levels for some radionuclides are considerably below the DIL if past results indicate that this level can be easily maintained without disrupting the efficient operation of the laboratory in question (ALARA). In other instances, the DIL is below the detection limits of the available monitoring techniques (notably for the actinides) and the removal level is set at any activity that significantly (at approximately the 95% confidence level) exceeds the detection limit. In most instances when a removal level is exceeded, subsequent monitoring reveals that the dose resulting from the contamination is negligible. If effective or organ dose equivalent exceed 2.5 mSv, they are reported and recorded.

Caution levels are normally set at one half the removal level, and if this level is exceeded, the employee is subjected to further monitoring to evaluate the significance of the contamination. Minor levels are set at somewhere between 1/2 and 1/100 of the caution level, the purpose of the minor level is to bring them to the attention of the responsible persons so that any appropriate action may be taken.

SUMMARY

The internal dosimetry monitoring program at CRNL has been reviewed. This program has evolved from a very simple one in the late 1940's to the comprehensive one described above. Results obtained with this program indicate that measures taken to control internal contamination at CRNL are effective.

REFERENCES

1. G.H. Kramer, S.E. Gardner and J.R. Johnson, AECL-7608*.
2. G.H. Kramer, J.R. Johnson and W. Green, AECL-8251.
3. N. Desai and G.H. Kramer, AECL-7986, 1983.
4. G.H. Kramer, AECL-7248, 1981.
5. G.H. Kramer and P.C. Beaulieu, AECL-7979, 1983.
6. G.H. Kramer, AECL-7162, 1981.
7. S. Joseph and G.H. Kramer, AECL-7512, 1982.
8. G.H. Kramer, Can. J. Chem. (in press).
9. G.H. Kramer and J.M. Davies, Anal. Chem. 54, 1428, 1982.
10. B.F. Peterman, Atomic Energy of Canada Limited, CRNL-2582, 1983.
11. J.R. Johnson, AECL-5621, 1976.
12. R.V. Griffith, et al. IAEA Int. Symp. Stockholm, June 1978.
13. B.F. Peterman, CRPA Conference, Toronto, Ontario, May 1983.
14. J.R. Johnson, AECL-5507, 1976.
15. J.R. Johnson, Rad. Prot. Dos. 2, (4), 245, 1982.
16. G. Cowper, J.R. Johnson and A.M. Marko, IAEA International Symposium, IAEA-SM-258/49, Madrid, October 1981.
17. J.R. Johnson and D.W. Dunford, AECL-7919, 1983.
18. ICRP Publication 26, Pergamon Press, Oxford, 1977.
19. ICRP Publication 35, Pergamon Press, Oxford, 1983.

* AECL-XXXX numbers refer to Atomic Energy of Canada Limited, published reports.

COMPARISON OF PHANTOM AND IN VIVO DOSAGE MEASUREMENTS IN DENTAL RADIOGRAPHY

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SUMMARY

For the assessment of doses in diagnostic radiography often phantoms are used, made of tissue equivalent material. The literature about dosimetry in dental radiodiagnosis shows a large divergence of results. This divergence can partly be explained by the application of different projection techniques and other technical parameters. The anatomical variations in the head and neck region contribute to the variance as well. It is necessary to compare phantom measurements and patient measurements when using phantom measurements for the prediction of patient doses. In this study dose measurements were carried out in an Alderson-Rando phantom with LiF-chips. Corresponding measurements were done in patients.

The patient measurements showed a larger variance. The entrance doses were lower, the exit doses higher to the patients as compared to the phantom. The study reveals that phantom measurements cannot simply be used for the determination or prediction of doses to patients.

INTRODUCTION

For the assessment of doses in diagnostic radiography often phantoms are used, made of tissue equivalent material. Many publications have appeared, concerning dose measurements and risk estimations in dental radiography based on such phantom studies. The literature about dose measurements in dental radiography shows a large divergence of the results. This divergence can partly be explained by the application of different projection techniques and other technical parameters. Another source of this divergence however are the patients themselves. The anatomical structures in the maxillo-facial region are very complicated. Slightly different radiographic techniques may result in highly varying measurements as a result of the influence of the anatomical structures on the dose distribution in the applied technique. This applies the more for the anatomical variance in the population as a whole and its influence on dose measurements. These effects can not be determined by means of phantom measurements only. Particularly in dental radiography the absorbed dose within a limited area may be rather high. The dose distribution is often very complicated by the complex anatomical structures in the head region and the small beam geometry. This makes it important to have a better understanding of the significance of phantom measurements in dosimetry for dental radiodiagnosis for the determination of patient doses.

AIM OF THIS STUDY

An investigation is being undertaken in our department aiming at the determination and comparison of doses in a number of radiographic techniques as used in dentistry (granted by the Dutch Government and the Foundation for Health Prevention). This investigation consists of two parts. The first part is based on phantom measurements for the determination of the dose distribution in several

radiographic techniques [1,2]. In continuation an investigation is undertaken in a number of dental practices in order to find the frequence and variance of the applied techniques.

In order to rate the results of the phantom measurements of the first part of this investigation at its true value for predicting patient doses, a comparative study has been caried out. The objective of this study was to determine the agreement of in vivo and in vitro dose measurements in dental radiography.

METHODS

Phantom dose measurements were carried out in an Alderson-Rando phantom. Selected LiF-chips (Harshaw, TLD-100) have been used for the dose registration. The X-ray source was a General Electric 1000 dental X-ray apparatus at 15 mA and 75 kVp. The first half value layer was determined at 3.2 mm Aluminium. The field diameter was 6 cm at the entrance site of the patient. The focus-skin distance was about 32 cm.

The phantom measurements were done ten times. The positioning of the X-ray beam was adjusted before each measurement in order to simulate the conditions in daily practice. Corresponding measurements were done in ten patients in places of the head and neck region accessible for TLD chips. The patient measurements were done during routine X-ray examinations.

The first part of the study was concerning entrance and exit doses consequent on the radiation of two bite-wing radiographs taken unilateral in the molar and the bicuspid region. Bite-wing radiographs give diagnostic information about the crown and upper part of the root of teeth and about the bone level of the supporting bone.

The second part was concerning measurements of doses consequent on the radiation burden of a full mouth x-ray survey. A complete full mouth survey consists of 16 to 20 X-ray pictures and gives diagnostic information about all teeth and the supporting bone structures. Chips were placed on fifteen locations on the skin surface in the head and neck region and in the region of the gonads.

The study was based on (the simulation of) a half full mouth survey of nine X-ray pictures in order to obtain information about the contribution of both primary and secundary radiation to the dose. Some regions are situated in the primary beam and receive primary radiation and the secundary radiation of subse-

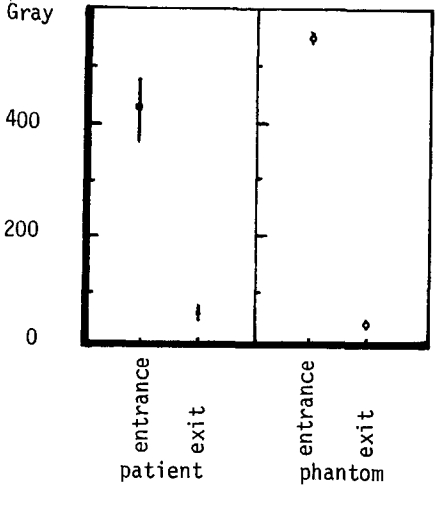

figure 1, Dose measurements in patients and phantom for bite-wing radiographs.

quent radiographs; opposite regions are located outside the primary beam of one or more X-rays and receive secondary radiation only.

Measurements were corrected for back scatter. The primary aim of this study was not the calculation of dose values, but rather the evaluation of the relative tissue equivalence of the phantom material as compared with human tissue material. Therefore, results are given in Gray, but they have no absolute value for dose estimations as such.

RESULTS

a. Bite-wing Radiographs

The back-scatter factor for the phantom material proved to be 1.20, which agrees with data from literature for human tissue material [3]. In this study back scatter for human tissue material was less than expected.

The results of the bite-wing study are shown in figure 1. The entrance doses in patient measurements prove to be lower than those of the phantom measurements. The inverted relation exists with regard to the exit doses. The standard deviation for the patient measurements is much higher, both for entrance and for exit doses, as compared to the phantom measurements.

b. Full Mouth Survey Radiographs

Results of the study of full mouth survey radiographs are given in figure 2. It is clear that the standard deviation of the patient measurements is larger as

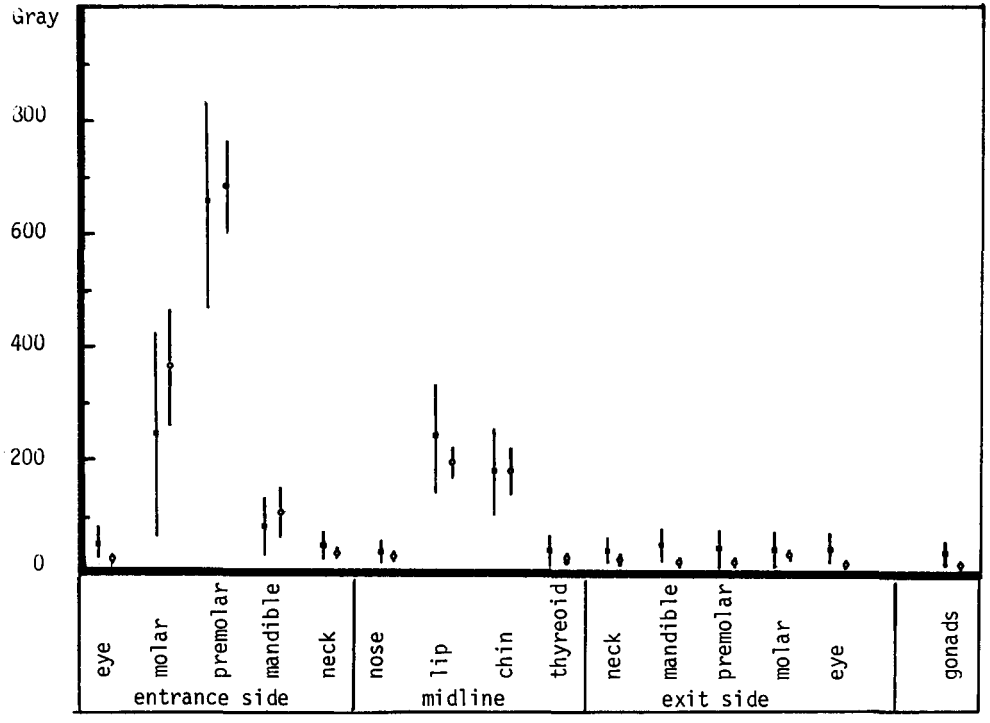

figure 2, Dose measurements in patients (■) and in phantom (●). Full mouth survey.

compared to the results of the phantom measurements. Phantom values are equal or exceed patient values when situated in areas which can be supposed to have received primary radiation of one or more X-rays of the survey.

Phantom values decrease after attenuation of the primary beam and in regions which have received only secondary radiation.

CONCLUSIONS

The bite-wing study and the study concerning the radiation dose of a full mouth survey show the same results. In the bite-wing study the entrance doses of the phantom measurements exceed the patient measurements. This fact can be explained from the production of the different amount of back scatter radiation in the phantom material and patient tissue material. This means that the phantom material is not fully tissue equivalent in the energy range used in this study.

The results of the full mouth survey study are partly masked by the fact that some regions receive both primary and secundary radiation in succesive radiographs. It is clear that the effects are more strength for regions far from primary radiation and in the center of the exposed field of one or more radiographs. Regions in the midline of head and neck show a more moderate effect.

As can be expected the variance of the patient measurements is larger than those of the phantom measurements. The influence of the anatomical structures comes into play strongly.

DISCUSSION

Especially in the low-energy range (50-100 kVp) of the radiation as applied in dental radiography the effects of secundary radiation may be enormous. The difference of the results of phantom and patient measurements can be the result of the slightly different composition of the phantom material as compared to human tissue material. The difference between entrance and exit doses in patient and phantom material makes it desirable to have more data between entrance and exit sites. Further investigation deals with this problem and is intended to give more information about the distribution of radiation within the head and neck region by measurements within the mouth. For this aim equipment has been developed for placement of TLD-chips within the mouth cavity.

The study reveals that the prediction of doses and risk estimations based on phantom measurements must be corected with in vivo measurements in the population in question, both for the assessment of the variability and for the magnitude of the average doses.

LITERATURE

[1] P.N. Ruys, P.F. van der Stelt and H. Boersma; The Ratio of Mass Energy Absorption Coefficients of Dosemeter and Tissue Material in Oral Dosimetry; J Dento-Maxillo-Facial Radiology, suppl. 5, p 19, 1983.

[2] H.T. Kwee, P.F. van der Stelt et al.; Comparison of Phantom Measurements and in vivo Measurements for Dosage Assessment in Oral Radiodiagnosis; J Dento-Maxillo-Facial Radiology, suppl. 5, p.20, 1983.

[3] F. Wachmsman and G. Drexler; Graphs and Tables for Use in Radiology; Springer Verlag, Berlin, 1976.

INFLUENCE OF THE ELECTION OF THE REMAINDER IN THE ASSESSMENT OF THE EFFECTIVE DOSE EQUIVALENT

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INTRODUCTION

The effective dose equivalent, H_E, was defined in ICRP Pub. 26 (1) for stochastic effects on the basis of the principle that risks arising from whole-body uniform irradiations and from non-uniform irradiations are equal. The H_E is the weighted mean whole-body dose equivalent defined by means of equation /1/. In addition to those organs explicitly quoted in Table 1, this equation involves a non-specific set of organs called "remainder". This has been defined as the set of 5 organs with the highest dose equivalents, excluding those explicitly indicated.

This paper raises the fact that the remainder definition is not precise and that this leads to great indeterminations in the value of the H_E for the case of non-homogeneous exposures.

ANTECEDENTS OF THE REMAINDER PROBLEM

Several authors have made different interpretations of the definition of the remainder. Bengtsson, L. (2) considered the mean trunk dose equivalent as the remainder dose equivalent. Kramer, R. (3) used a fixed remainder model composed by stomach, liver, upper large instestine (ULI), small intestine (SI) and pancreas. Whøni,T. (4) considered the GI dose equivalent as the remainder dose equivalent. ICRP Pub. 30 (5) used the list of organs shown in Table 2 as a reference set. Maruyama, T. (6) considered brain, stomach, liver, rectum and ULI. Finally, in another work (7), the author used a different set composed by SI, LI, stomach, kidneys, pancreas, spleen, liver, adrenals, brain, bladder and heart, as a reference set.

These diverse interpretations arise because the ICRP has not explicitly defined the set of reference organs from which the remainder is to be selected. ICRP Pub. 26, paragraph 66, reads that certain irradiated tissues may be ignored for radiation protection purposes, but it does not specify which are those tissues.

Since the remainder is the "organ" with the maximum weight factor, as may be seen in Table 1, its influence needs to be studied when the H_E must be assessed. Kramer, R. (8) demonstrated that, for whole-body irradiations in photon parallel fields, the differences in the H_E calculated on the basis of fixed or variable remainder models are below 17% for energies higher than 50 KeV. In cases of non-homogeneous exposures, the differences may be far higher, as shown in this paper through two practical cases described below.

ASSESSMENT OF THE EFFECTIVE DOSE EQUIVALENT

The influence of the remainder selection for occupationally exposed workers was analyzed in two practical cases. The first case refers to a routine gynecological practice with Ra-226, for which an assessment was made of the H_E absorbed by the ginecologist during a given practice. The physician performs the implantation seated behind a lead shield and is mainly exposed in upper limbs, head and upper thorax. Details on the development of this routine practice have been published by Eckerl, H. (9) and Spano, F.(10)

The second case refers to the H_E resulting from an event occurred during maintenance work performed on an x-ray equipment unit. A technician stood for 20 minutes facing an x-ray tube at 170 kVp and was exposed in head and upper thorax. Although the H_E was not defined for its application in this type of events, its value was calculated in order to show the influence of the remainder selection in cases of strongly non-homogeneous irradiation.

The H_E was defined in ICRP Pub.26(1) by means of the equation:

$$H_E = \sum_T W_T H_T \qquad /1/$$

where:

W_T is the weighting factor representing the proportion of the stochastic risk resulting from tissue T to the total risk, when the whole body is irradiated uniformly. The male W_T factors used in this work are shown in Table 1.

H_T is mean dose equivalent in tissue T.

Table 1

Male risk coefficients

Tissue	Absolute (10^{-6} rem^{-1})	Relative
Testes	40	0.283
Red bone marrow	20	0.141
Lungs	20	0.141
Thyroid	5	0.035
Bone surface	5	0.035
Skin	1	0.007
Remainder	50	0.354

The mean organ dose equivalent absorbed by the physician was obtained through the development of a mathematical model based on the application of the Monte Carlo method to photon transport in a MIRD V phantom developed by Spano, F.(10) In the case of the x-ray incident, the distribution of the dose equivalent was obtained by using a Rando phantom and 3 x 3 x 1 mm^3 LiF and CaF:Dy thermoluminiscent dosimeters.

The H_E was assessed on the basis of four different remainder models described in Table 2. The first one is a fixed remainder defined by Kramer, R.(3) The second one considers the target tissues given in Table 4.1 of ICRP Pub. 30 (5) (except for muscle and

the organs included in Table 1) as a reference organ set. This paper introduces, as a third reference organ set, a selection of the original organ list in ICRP Pub. 23. The fourth model considers the head mean dose equivalent as the remainder dose equivalent.

Table 2

Remainder Models

Kramer:
Stomach, ULI, SI, liver and pancreas.
ICRP30:
ST wall, SI wall, ULI wall, LLI wall, kidneys, liver, pancreas, spleen, thymus, uterus, adrenals and bladder wall.
Organs selected from the ICRP 23 list:
Adrenals, brain, esophagous, eyes, gall bladder, stomach, SI, ULI, LLI, kidneys, larynx, liver, tonsils, pancreas, parathyroid, pineal gland, pituitary gland, prostate gland, salivary glands, spleen, thymus, trachea, ureters, urethra and urinary bladder.
Mean Head Dose Equivalent: as the remainder dose equivalent.

Tables 3 and 4 show the H_Es, as functions of the four selected remainder models, for the physician performing the practice with Ra-226 and for the technician who suffered the x-ray exposure respectively.

Table 3

Effective Dose Equivalent: Physician in Radiumtherapy

Remainder Model	Effective Dose Equivalent (mrem)
Kramer	2.1
ICRP 30	2.7
Organs selected from ICRP 23 list	4.3
Mean head dose equivalent	3.2

Table 4

Effective Dose Equivalent: Technician in X-ray Incident

Remainder Model	Effective Dose Equivalent (mrem)
Kramer	6.466
ICRP 30	8.667
Organs selected from ICRP 23 list	56.995
Mean head dose equivalent	23.270

CONCLUSIONS

The differences found in the H_E absorbed by the physician during the radiumtherapy, as a function of the adopted remainder model, reach a factor of 2.1. This factor is obtained by comparing the values resulting from the application of a fixed remainder defined by Kramer, R. (3) and of a variable remainder based on the organ list in ICRP Pub. 23 (11), as shown in Table 2. The difference obtained from the same comparison performed on the technician who suffered the x-ray incident reached a factor of 8.8 in the H_E values.

Similar variations may be found for the H_E resulting from non-homogeneous irradiations, specially when involving head and upper thorax exposures.

The results obtained indicate that the present definition of the remainder is not precise and that it leads to great indeterminations in the value of the H_E. Therefore, the set of organs from which the remainder is to be selected should be explicitly defined.

BIBLIOGRAPHY

1. ICRP Publication 26. Oxford, Pergamon Press, 1977. Annals of the ICRP, Vol. 1: No. 3, 1977.
2. Bengtsson, L.G., Jensen, M. and Lindberg, L. IAEA Proceedings Series STI/PUB/521, Vienna, 1979.
3. Kramer, R. 1979 GSF-S-556.
4. Wohni, T. and Strander, E. Health Phys. (U.K.) v.36: 71-73, 1979, N° 1.
5. ICRP Publication 30. Oxford, Pergamon Press, 1979. Annals of the ICRP v.2: No. 3-4, 1979.
6. Maruyama, K. et al. J. Radiat. Res. (Japan) v.22: 182-203, 1981, No. 2.
7. Thomasz, E. et al. Nucl. Instrum. Methods. Phys. Res. (Netherlands) v.175: 196, 1980, No. 1.
8. Kramer, R. and Drexler, G. Radiat. Prot. Dosim. (U.K.) v.3:13-24, 1982, No. 1/2.
9. Eckerl, H., Thomasz, E. and Drexler, G. Strahlentherapie (Germany, F.R.) v.158: 422-426, 1982, No. 7.
10. Spano, F. and Thomasz, E. VIth. Congress of IRPA, Berlin, May 7-12, 1984.
11. ICRP Publication 23. Oxford, Pergamon Press, 1974.

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TOWARDS THE ASSESSMENT OF PLUTONIUM IN LUNG USING OBSERVED DISTRIBUTIONS OF ACTIVITY WITHIN THE LUNG*

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Systems for the direct measurement of plutonium 239 in the human lung are subject to many sources of error and have high associated values of minimum detectable activity (MDA). The main sources of error are counting statistics, subject background and calibration. In recent years improvements in equipment and techniques have resulted in reductions in the errors associated with counting statistics and subject background. Thus meaningful conclusions are now possible for most subjects when assessing plutonium oxide levels in the lungs using a routine programme(1). Calibration errors, which arise from a combination of several factors, remain largely unexplored. In particular, the errors associated with varying distributions of activity within the lungs or with different lung sizes often are unassessed for a measurement on an individual subject. Estimates as to the magnitude of distribution errors vary but are the order of a factor of 3 in calibration for any particular individual. Inhalation studies give some guide as to these variations. Work with chest phantoms used at Winfrith indicates that a factor of 2 could be expected based on differing lung sizes alone. Such errors are not easily quantified and are usually not included in the quoted error on a measurement.

One method of detecting the variations in distribution within the lung is to use the responses from a number of detectors to predict a distribution of activity in the lungs. Each detector would view an area of the chest with known structural details. This approach requires an array of detectors and a seldom available detailed knowledge of each individual's internal chest structure. Such an approach is being followed in these laboratories and this paper reports on some preliminary studies. In order to understand and estimate what improvements might be achieved by such an approach, we describe briefly some observations made from 39 measurements on a small population of subjects using just two detectors and attempting to compare responses from two areas i.e. the front left, and front right chest. These results are contrasted with the results obtained when the measurements are interpreted with less subject specific calibration information as is the normal practice in a routine monitoring programme.

Subject Background Prediction

There are a number of ways to predict subject background(2). This laboratory has developed a computer based tool to investigate various relationships between observed background in the plutonium band (12-25 keV) and the count rate at higher energies for any given measurement, using a continually growing data base of responses from known "clean" subjects. Over 200 measurements on subjects were used

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to form the currently preferred relationship for each of two phoswich detectors. This is of the form:-

$$B = pe^{-qt}D \quad - - - - (1)$$

where B is the predicted background count rate for the plutonium band of 12-25 keV, p and q are constants for a given detector, t is the chest wall thickness (CWT) of a given subject and D is the count rate at a higher energy band 80-126 keV. The inclusion of an exponential term in the relationship gives a better fit to the observed data than the simple linear expressions used previously. Such a term might be expected to crudely describe the attenuation of scattered radiation from higher energies.

Subject Efficiency Predictions

Efficiency predictions are based on measurements using a realistic chest phantom supplied by Lawrence Livermore Laboratories (LLL)(3). This phantom has a selection of tissue equivalent overlays for calibrations using different values of tissue thickness, and is complete with artificial lungs containing a homogeneous distribution of plutonium 239 activity. The equivalent tissue thickness at points across the torso and overlays is well documented.

The derived efficiency for any subject is of the form:-

$$A(be^{-cT} + de^{-fT} + ge^{-hT}) \quad - - - - (2)$$

where the three exponential expressions correspond to the three principle X ray photons from the decay of plutonium, b, d, g are constants for a given detector, and c, f, h are absorbtion coefficients in soft tissue. T is the difference in CWT between the phantom and subject and A is a constant describing the response from unit activity of plutonium in the phantom. In general form this expression may be used for a combined response from both lungs or for a response from a single lung. With the detector geometry used at the energies of interest, the contributions to a detector from the opposite lung are minimal (estimated at maximum 7%).

Chest Wall Thickness Estimations

With the adoption of the LLL phantom, Winfrith has abandoned the early concept of a mean soft tissue thickness (MSTT) in favour of CWT. For routine measurements CWT is estimated from the subject's weight and height. The estimation at present is based on that used at LLL and has the form:-

$$\text{Chest Wall Thickness (cms)} = 11 \times \frac{\text{Weight (Kg)}}{\text{Height (cm)}} - 2.0 \quad - - - (3)$$

This gives one single average estimate for a subject's CWT (front of chest only) which can be compared to a single average CWT measurement for the phantom. For the purposes of these investigations we derive additional average CWTs for each side of the basic phantom. There are significant differences in these numbers

i.e. The CWT from the left and right sides are 1.81 and 1.59 cms respectively compared to the mean value of 1.63 cms.

For each of the subjects used in this study, actual measurements of CWT using ultrasonics were made along single lines on each side of the chest. These lines were taken for each subject to extend down the chest from the mid-clavical point and parallel to the sternum.

Discussion

Two separate estimates of plutonium activity were calculated for each measurement, i.e. one estimate based on each of the two detector responses available. Each estimate was obtained by correcting a detector response for subject background and then applying the appropriate subject efficiency.

Results from measurements on clean subjects show that in most cases the modulus of the estimate from the detector viewing the left side of the subject was greater than the modulus of the estimate from the detector viewing the right side of the subject. This observation held true even for subjects having negative estimates as part of a general distribution of results around zero. Furthermore, the differences in estimates between the detectors for a given subject were greater than expected after taking account of the relevant statistical errors of the measurements. These differences would certainly mask the small effects expected from any differential distribution of plutonium between the lungs. To study and attempt to explain the above, the results for the population of subjects were calculated in three different ways.

(a) The responses were interpreted using a single average derived CWT for each subject and a single CWT for the phantom.

(b) The responses were interpreted using the single CWT for each subject but with different phantom CWTs for each side of the chest.

(c) The responses were interpreted using different CWTs on each side of the chest for both phantom and subject. In this case two separate CWTs for each subject were obtained by distributing the single CWT as derived from equation 3 according to the ratio of the ultrasonic measurements made on each side of the chest. Thus an estimated CWT was present in each of the three alternative methods of calculation

Comparison of Methods

The method of comparing the data was to calculate, for each measurement, a weighted square of the difference in estimates from the left and right sides of the chest. These differences were summed for each of the three methods of interpretation.

For each measurement, W the weighted square of difference is defined as:-

$$W = (X-Y)^2 \, C^{-2} \qquad - - - - (4)$$

and $$C^2 = (x^{-2} + y^{-2})^{-1} \qquad - - - - (5)$$

where $X \pm x$ is the estimate from the left side and $Y \pm y$ the estimate from the right side and where C is the error on the weighted mean of X and Y.

W is tabulated in Table 1. It can be seen that, for the group of measurements, agreement between estimates for each side of the chest improve as more account is taken of variation in chest wall thickness in either phantom or subject.

TABLE 1

INTERPRETATION METHOD	W
a	953
b	471
c	245

Conclusions

The development of techniques which will be sensitive to various distributions of activity within the lungs is the aim of the project. Such techniques, operating at or around the lower limits of present detection systems, can be used to calculate a more realistic calibration factor specific to the individual and the contaminant. This paper shows that, even in the simple case of an assumed homogeneous distribution of activity throughout the lungs, significant improvements can be obtained by measuring and assessing separately the contents of each lung. These improvements arise from deriving specific CWTs for each side of the chest both for the subject and the phantom.

It is emphasised that for some individuals within the group studied, application of this simple technique worsens the agreement between the positive estimates for the separate lungs. It could be concluded that this is evidence of non-uniform distribution although in some cases we have observed unusual rib spacing, which is being investigated further.

The approach using two values of CWT, will be routinely used for deriving calibration factors for plutonium exposed subjects as an initial step prior to completion of work on multi-detector systems using smaller defined regions of the lung and a CWT specific to each detector/source geometry.

References

1. The Accuracy of a Routine Plutonium in Lung Assessment Programme - D Ramsden. IRPA Jerusalem, Israel, 9-14 March 1980.
2. Measurement of Transuranic Elements In-Vivo at CRNL - Johnson J R AECL 5621 1980.
3. Fabrication of a Tissue - Equivalent Torso Phantom for Intercalibration of IN-VIVO Transuranic-Nuclide Counting Facilities - R V Griffith, P N Dean, A L Anderson and J C Fisher. IAEA Stockholm, Sweden, 26-30 June 1978.

RADIATION HAZARDS FROM INTERNAL EMITTERS: CALCULATION OF ABSORBED DOSE AND EFFECTIVE DOSE EQUIVALENT

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Introduction

The absorbed dose from radioactive material incorporated in man is rarely, if ever, uniform throughout the body. The true non-uniform radiation exposure can be converted mathematically to an equivalent whole body dose without changing the risk of damage induced by radiation. It would be more logical to calculate the risk caused by radiation directly. However, the absolute risk factors of the late effects are not known accurately; the uncertainty may even be of the order of magnitude. Thus it is preferable to approach the risk assessment with relative risk factors or weighting factors (w). The International Commission on Radiological Protection, ICRP, has used this technique to convert the dose equivalents H_T (Sv) of single organs to the effective dose equivalent $H_E = \sum_T w_T H_T$.

Detailed information on the harmful effects of the different types of radiation is needed to convert the doses absorbed by the organs to the dose equivalents or to the risk of somatic and genetic harm. At present, however, there is no unambiguous way to compare the hazard of one type of radiation with another; the comparison depends on the biological end point. The ability to kill growing cells and the effectiveness of malignant transformation depend greatly and in different ways, on the radiation quality, e.g. on LET. In addition to the absorbed dose, many other factors - biological and environmental variables - have a great influence on the risk of clinical cancer.

Dose Calculation

The present system of dose calculation makes good use of precalculated S factors (the absorbed doses per unit cumulative activity) on the basis of the work of the MIRD committee.[2] The S factors are presently available only for the reference man (70 kg). Extensive additional Monte Carlo calculations would be required to give individual, or nearly individual S factors. In some special cases, however, the S factors calculated for the reference man can be converted fairly easily to individual values. For for the thyroid, a simple correction, proportional to the cube root of the mass of the gland, gives results nearly equal to those of more sophisticated methods; however, a discrepancy is detected in cases of enlarged thyroids.[3]

In the following formulation, different quality factors (Q_{np}, Q_p) are used for non-penetrating and penetrating radiation; however, they are omitted from the computer program because no reliable information is available. In fact, the method gives an effective absorbed dose rather than an effective dose equivalent. The goal is to avoid the direct calculation of any index for the biological hazard which contains too many poorly known or even unknown factors.

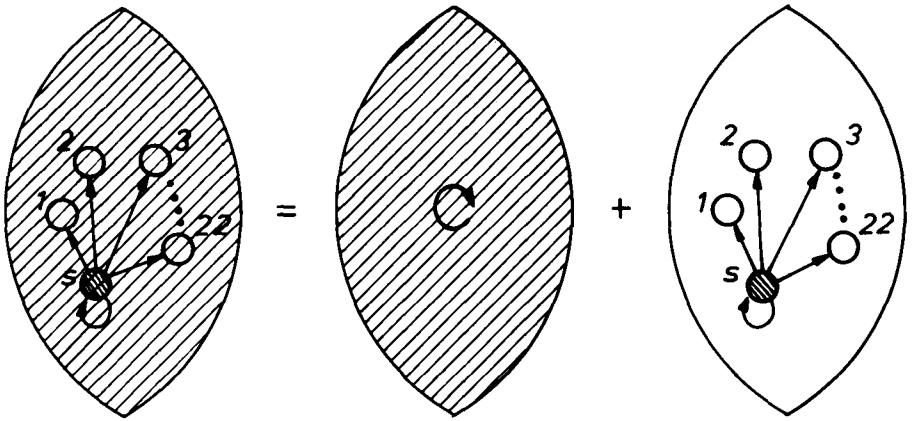

Fig. 1. Calculation of the effective dose equivalent (see Eq. (1)). The dose is obtained in two parts: from the uniform distribution of the activity and from the source organ(s) s.

Let us assume that the residence time τ_s (the cumulative activity per unit intake of radioactive material) in the source organ s and the residence time τ_{WB} in the whole body are known (measured). The activity outside the source organs is assumed to be uniformly distributed throughout the body. The effective dose equivalent is now calculated from the contribution of the uniform activity in the body and from the activity of the source organs:

$$H_E = \tau_u \left[Q_{np} S_{np}(WB \leftarrow WB) + Q_p S_p(WB \leftarrow WB) \right] + \sum_s \sum_i w_i \tau_s^* \left[Q_{np} S_{np}(i \leftarrow s) + Q_p S_p(i \leftarrow s) \right] \quad (1)$$

The residence times τ_s have to be corrected; otherwise part of the activity would be included in the equation twice:

$$\tau_s^* = \tau_s - \frac{m_s}{m_{RB}} \tau_{RB} \quad (2)$$

where the subscript RB refers to the remaining body, with residence time

$$\tau_{RB} = \tau_{WB} - \sum_s \tau_s \quad (3)$$

The residence time for the uniform activity in the whole body is simply

$$\tau_u = \tau_{RB} \frac{m_{WB}}{m_{RB}} \quad (4)$$

In addition to the 23 organs considered in the program, other parts of the body are also exposed to radiation from the source organs. The contribution of this component - the remaining part (RP) of the body - could be assessed as follows:

$$D(RP) = \sum_s \tau_s^* \, S_p(RP \leftarrow s) \quad (5)$$

An expression for the factor $S_p(RP \leftarrow s)$ was derived by Toivonen.[3] However, the contribution of Eq. (5) to the effective dose equivalent is small and can thus be excluded from the dose calculation.

The present computer program has been developed on the basis of the CAMIRD/III program.[1] CAMIRD/III is written in FORTRAN IV and was originally implemented on IBM 370/168. Using CAMIRD/III, the S factors were calculated (UNIVAC 1100) for 22 nuclides important in nuclear medicine. These results were transferred to an ECLIPSE S-140 computer. A PASCAL program (SMIRD) was developed to cope with the dose calculation. SMIRD makes use of files containing precalculated S factors for each radionuclide. A file comprises about 6500 ASCII characters. SMIRD is an interactive program including various types of checking of the input data. The absorbed dose is calculated by SMIRD for the desired target organ and the result is given separately for each source organ. In addition, the total absorbed doses of all organs are calculated and, for clear reporting, they are printed in descending order. The effective dose equivalent is calculated with the aid of ICRP weighting factors. These, however, have been normalized to give a weight of one for all the somatic effects; the gonad doses are calculated separately but they are not included in the effective dose equivalent. SMIRD is a fairly small program, about 450 PASCAL lines, and it can be easily implemented on different types of computer systems (available for VAX-11, PDP-11, PC-325, PC-350, CP/M-86, CP/M-80; Mimarobe Ltd, Tampere).

Application to ^{131}I with Stable Iodine

The residence times of a radionuclide in the source organs are the input data needed by the SMIRD program. The kinetics of the nuclides vary widely in different persons; the individual variability can be taken into account only by the direct measurement of the behaviour of the radioactive material in the body. However, the model studies can provide useful information to compare completely different exposure situations or to assess quantitatively the effect of different treatments or protection schedules. In the present study the kinetics of ^{131}I were simulated by a set of differential equations (the model can be used for all isotopes of iodine). The details of the model are given by Turai and Toivonen.[4]

Table 1 shows an example of the calculations of the prophylactic efficacy of 30 mg of potassium iodide (KI) given 0.5 h after the intake of ^{131}I. The model studies show that although the thyroid

gland can be protected effectively with KI (ratio 0.062), the total number of nuclear disintegrations in the body is reduced much less drastically (ratio 0.239). The reduced total risk of late effects is not described by the thyroid doses but rather by the effective dose equivalents. However, it is not at all clear, using the present weighting factors, that the effective dose equivalent is the true indicator of the radiation hazard.

Table 1. Residence times and radiation doses of different organs for ^{131}I; (a) unperturbed kinetics, (b) 30 mg potassium iodide given 0.5 h after the intake of radioiodine.

	(a)	(b)	Ratio (b/a)	
Residence times (h):				
thyroid	63.69	3.940	0.062	
blood (inorganic)	9.073	11.75	1.295	1.039
blood (organic)	2.377	0.147	0.062	
bladder	1.906	2.464	1.293	
lungs	0.164	0.164	1.	
total	77.21	18.47	0.239	
Radiation doses (μGy/MBq):				
thyroid	374000	23100	0.062	
bladder	646	831	1.286	
blood	330	283	0.858	
thymus	225	30.2	0.134	
lungs	82.2	36.5	0.444	
muscle	80.4	20.3	0.252	
red marrow	59.7	21.3	0.357	
skeleton	52.9	17.8	0.336	
skin	49.2	11.5	0.234	
uterus	39.3	45.7	1.163	
ovaries	25.6	28.4	1.109	
testes	20.5	23.0	1.122	
effective somatic dose equivalent (μSv/MBq)	15100	1040	0.069	

References:

1. Bellina C.R. and Gruzzardi R., CAMIRD/III: A Revised Version of the CAMIRD/II and MIRD S Packages for Internal Dose Calculation: Concise Communication. J. Nucl. Med. 1980(21), 379-383.
2. Snyder W.S., Ford M.R., Warner G.G. and Watson S.B., "S", Absorbed Dose per Unit Cumulated Activity for Selected Radionuclides and Organs. Medical Internal Radiation Dose Committee, MIRD pamphlet No. 11, N.Y., Society of Nuclear Medicine, 1975.
3. Toivonen H., Calculation of Patient Doses from Radioisotopes. Institute of Radiation Protection: STL-B47, Helsinki, 1983 (in Finnish).
4. Turai I. and Toivonen H., Radiohygiene of Fission Isotopes of Iodine: Experiments in Rats and Model Studies in Man. Institute of Radiation Protection: STL-A42, Helsinki, 1983.

CALIBRATION OF A PHOSWICH SYSTEM FOR THE IN VIVO MEASUREMENT OF 239-Pu AND 241-Am ACTIVITIES

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INTRODUCTION

The radiotoxicity of transuranic elements, particularly that of 239-Pu and 241-Am, has led to the development of a measuring system allowing for assessing low levels of lung inhalation by workers at the various restricted areas using these nuclides.

Estimations based on radiochemical analyses of excreta require the use of models whose transfer parameters, at a human level, are hard to assess.

Thus, direct measurement allows for insuring, both quickly and in a precise manner, the evaluation of a given internal contamination, as well as for avoiding the use of inaccurate metabolic parameters.

The use of phoswich detectors, with pulse shape discrimination (PSD), has led to attaining appropriate detection levels as per the regulations presently in force. Therefore, these systems have been preferently considered when dealing with the elements available for the control of internal contamination with 239-Pu and 241-Am.

MEASURING SYSTEM

The system utilized is composed by two phoswich detectors [3 mm NaI(Tl) and 50 mm CsI(Tl)], 127 mm in diameter, and by an electronic chain associated with each one of them.

The preamplifier is of the charge-sensitive type and has a special output that inhibits the system after a signal exceeding its dynamic rate. The PSD operation is based on the use of the crossover-crossover technique applied on a signal produced by two amplifiers with double delay lines and different delay times (1), (2), (3).

The system was implemented in such a way that the spectra of low-energy NaI(Tl) and high-energy CsI(Tl) crystals may be simultaneously stored, for each detector, at different positions in the multichannel analyzer core memory.(6)

The system is completed with a 6 m^3 monitoring room, shielded by a 15 cm iron wall. The inner side of the hall is shielded with 3 mm lead and 0.3 mm cadmium in order to improve the background of the low-energy area, plus a ventilation system with an absolute filter at the air inlet.

The detectors were compared individually with a NaI(Tl) crystal, with a multiwire proportional counter (4) and with the same counter plus an associate PSD. In the latter case, the constant fraction method was applied on the decay time of the signal produced by an amplifier with simple delay line.

The results in Table 1 show the advantages of using phoswich detectors in this type of measurements.

Table 1

	Efficiency $\left[\frac{cpm}{X \text{ or } \gamma \text{ min}^{-1}}\right]$		Background (cpm)	
Detector	239-Pu	241-Am	239-Pu	241-Am
Phoswich 1	$8.8\ 10^{-2}$	$8.9\ 10^{-2}$	2.4	3.5
Phoswich 2	$8.6\ 10^{-2}$	$8.7\ 10^{-2}$	2.8	3.1
INa (Tl)	$11.1\ 10^{-2}$	$8.9\ 10^{-2}$	157.5	12.1
Proportional Counter	$8.9\ 10^{-2}$	$3.4\ 10^{-3}$	16.2	4.0
Proportional Counter + PSD	$7.5\ 10^{-2}$	$2.4\ 10^{-3}$	7.2	1.0

CALIBRATION METHODOLOGY

Assessment of the Calibration Factors

239-Pu produces 13, 17 and 20 Kev photons in a $4.6\ 10^{-2}$ proportion for each desintegration (uranium x rays). Besides, since the mean thickness of the soft tissue for 17 Kev is 6 mm and transmission at the skeleton is approximately 50%, only 1.6% of the x rays produced may be detected, for an individual with 3 cm chest soft tissue.

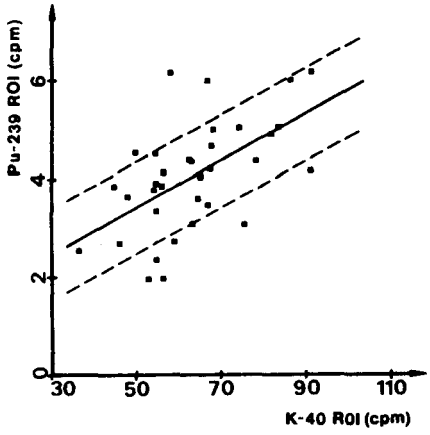
The measurement of known mixtures of 239-Pu and 241-Am is simpler, since the 60 Kev photons produced by 241-Am, with 36% efficiency by desintegration, are more easily detectable (mean thickness in soft tissue: 3.5 cm).

In order to assess the calibration, lungs of equivalent tissue were made with sources of 239-Pu and 241-Am uniformly distributed. These were adapted to a Remab hybrid phantom containing a human skeleton, with chest soft tissue approximately 3 cm thick (7),(8),(9).

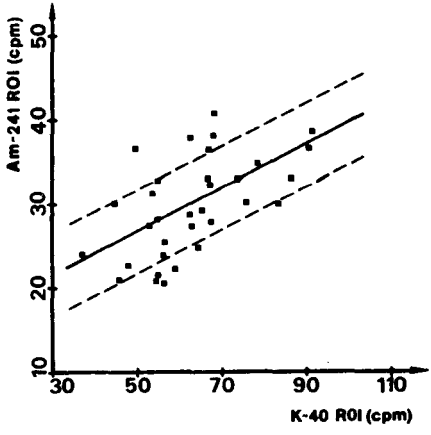
Prediction of the background

The presence of high-energy gamma-emitting nuclides, such as 40-K, produces background alterations in the various areas of interest (Roi's). For instance, for an individual with a mean amount of this nuclide, the background in the ROI for 239-Pu increases from 5.3 to 10.7 cpm and that in the ROI for 241-Am increases from 6.6 to 40.1 cpm.

Due to the above, the need arises to assess the background for each individual. Thus, a correlation was made between the 40-K content and the background variations at the ROIs for 239-Pu and for 241-Am, based on background measurements obtained from several persons (Graphs 1 and 2).



Graph 1



Graph 2

Corrections for Variations in the Thickness of the Soft Tissue

The calibration factors must be corrected in order to make them fit the anatomic features of each individual, since they were calculated for a soft tissue thickness that is usually different (Remab phantom).

In order to evaluate such difference, the methodology adopted was based on the measurement of the transmission resulting from placing a 600 cm^2 241-Am source under each individual's trunk (5).

These measurements, as well as a similar one performed on the above phantom, were related with a given thickness of equivalent tissue by means of experimental curves.

The coefficient utilized for correcting the calibration factor is that corresponding with one half the difference existing between the phantom and the individual and is also obtained by means of curves that were calculated by adding the phantom various thicknesses of equivalent tissue.

RESULTS

In order to evaluate the reliability of the method introduced herewith, a Remab phantom and a lung with 239-Pu and 241-Am uniformly distributed (Pu/Am = 54.7) were used, while the thickness of the chest equivalent tissue was increased in 1 cm.

Under these conditions, a series of measurements was performed and the activity of 241-Am was calculated with a maximum 12% error. For the direct evaluation of 239-Pu, the maximum error was 27%.

In this case, the minimum detectable activities (MDA) (10) were 52 nCi for 239-Pu and 0.12 nCI for 241-Am, while the ones for the reference phantom were 21 and 0.08 nCi respectively. The former were calculated with the calibration factors derived from the transmission resulting from the extense source and errors were 22 and 12% for 239-Pu and 241-Am respectively.

The MDA value obtained for the direct measurement of 239-Pu is high if compared with 16 nCi, which is the maximum allowed burden. However, if the calculation is made by using the Pu/Am ratio, that value results to be 7 nCi.

The importance of the errors made in the calculation of the activity, when the background is ignored because of variations in the 40-K content, is demonstrated by the calculation of the MDA. For an individual with an average 40-K content and a thickness of the soft tissue equal to that of the phantom, errors of 7% for 239-Pu and of 8% for 241-Am were found, the MDAs being 25 and 0.12 nCi respectively.

CONCLUSIONS

The figures shown above indicate that, although the detection levels for 239-Pu are generally higher than the lung burden derived from the recommended limits, their indirect measurement by means of 241-Am, plus the accuracy, versatility and fast applicability of the measuring method introduced herewith, make it particularly adequate for the routine control of professionally exposed workers, as well as for emergencies.

BIBLIOGRAPHY

1. Shoffner, B.M. IEEE Trans.Nucl.Sci.(USA) vol. NS-19: 502-511, 1972, No. 1.

2. The Harshaw Chemical Co. "The NC-25 pulse shape analyser".

3. The Harshaw Chemical Co. "Instruction Manual NC-25".

4. Morucci, J.P. 1966. CEA-R-3027. 87 p.

5. Daburon, M.L. and Jeanmaire, L. 1974. CEA-R-4580 34 p.

6. Bonino, A. and Righetti, M.A. X Reunion Cientifica de la Asociacion Argentina de Tecnologia Nuclear. Bahia Blanca, Argentina, Nov. 2-6, 1981.

7. Swinth, K.L. (ed.) 1976. BNWL-2088. 247 p.

8. Umbarger, C.J. and Jett, J.H. 1976. LA-UR-76-1718. 18 p.

9. Altshuler, B. and Pasternack, B. Health Phys. (U.K.) vol. 9: 293-298, 1963, No. 3.

FROM BODY BURDEN TO EFFECTIVE DOSE EQUIVALENT

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In §104 of ICRP-26 the concept of "effective dose equivalent" in introduced. It is the sum of organ or tissue dose equivalents H_T weighted with appropiate relative risk factors w_T: $H_E = \Sigma_T w_T H_T$. This concept enables a direct comparison with uniform whole body irradiation as far as risk is concerned. The effective dose equivalent is a powerful tool in the assessment of internal contamination of humans with radioactive material.

In case of an internal contamination with radionuclides, that emit penetrating radiation , the whole body counter is usually used to determine the magnitude of the contamination. The problem is to relate the value of the measured body burden to the subsequent effective dose equivalent.

Two extreme cases may be considered at this point: single and continuous intake. For single intake the ICRP gives a complete model for the translocation of material in the human body. This model is based on a linear chain of compartments. A system of coupled differential eqations describes the mathematics. The solution in terms of organ or tissue burden q_i as a function of time after intake of unit activity is given by

$$q_i(t) = \left(\prod_{k=1}^{i-1} \lambda_{k,k+1} \right) \sum_{k=1}^{i} \left[\left(\prod_{\substack{p=1 \\ p \neq k}}^{i} (\lambda_p - \lambda_k)^{-1} \right) \exp(-\lambda_k t) \right] . \qquad (1)$$

Here $\lambda_{k,k+1}$ is the rate of transfer from compartment k to k+1 and λ_k is the rate of loss from compartment k, including radioactive decay. The sum of organ and tissue burdens gives the whole body burden. In a computer programme (BODYBUR) the expression (1) is evaluated for 18 points in time after a single intake. The result may be used to estimate the intake from a measured body burden.

For continuous intake the values of U_S as given in the ICRP-30 supplements are used. These are expressed in Bq.s per Bq intake. It can be shown that they may also be considered as factors expressing Bq(organ burden) per Bq/s intake rate. For short lived nuclides the values arrived at following this procedure will represent the equilibrium body burden as the result of a continuous intake during a sufficiently long period of time. For long lived nuclides a continuous intake during 50 years is assumed, which yields the final body burden.

Finally, the conversion factors giving the effective committed dose equivalent per unit intake, as derived from ICRP-data, are presented. Analogously the conversion factors relating the effective dose-equivalent rate to a steady body burden are given.

The information described above has been tabulated for 400 nuclides, with a physical halflife exceeding 12 hours. An example of the output of the programme is given on the next page.

A limited number of copies of the final report will be available to interested persons at the conference.

CO-60 HALFLIFE= 1.924E+03 DAYS
AMAD= 1.0 MICROMETER
TTC= .50 DAYS

		A	T(DAYS)
LIVER	F2= .050	.600	6.000
		.200	60.000
		.200	800.000
REST	F2= .450	.600	6.000
		.200	60.000
		.200	800.000

COMPOUND	INGESTION	INHALATION CLASS D	W	Y
ORG+(INORG WITH CARRIER)	F1= 3.E-1			
(HYDR)OXIDE,INORG TRACER	F1= 5.E-2			
(HYDR)OXIDE,HALIDE,NITRATE	F1= 5.E-2			5.E-2
OTHER COMPOUNDS	F1= 5.E-2		5.E-2	

BODY BURDEN AFTER SINGLE INTAKE(FRACTION OF INTAKE)

TIME (DAYS)	INGESTION F1= 5.E-2	INGESTION F1= 3.E-1	INHALATION CLASS W	Y
.25	9.86E-01	9.69E-01	6.19E-01	6.29E-01
.50	9.23E-01	8.86E-01	6.06E-01	6.22E-01
1.00	7.11E-01	6.80E-01	5.60E-01	5.76E-01
2.00	3.35E-01	3.69E-01	4.22E-01	4.22E-01
3.00	1.46E-01	2.20E-01	3.09E-01	2.97E-01
5.00	3.68E-02	1.27E-01	2.08E-01	1.91E-01
7.00	1.95E-02	1.04E-01	1.79E-01	1.66E-01
14.00	1.25E-02	7.49E-02	1.53E-01	1.55E-01
30.00	8.38E-03	5.32E-02	1.23E-01	1.49E-01
60.00	7.14E-03	4.28E-02	8.79E-02	1.42E-01
90.00	6.21E-03	3.73E-02	6.40E-02	1.36E-01
180.00	4.60E-03	2.76E-02	2.83E-02	1.20E-01
365.00	3.26E-03	1.96E-02	1.12E-02	9.37E-02
730.00	2.04E-03	1.23E-02	6.14E-03	5.85E-02
1050.00	1.38E-03	8.28E-03	4.14E-03	3.94E-02
1825.00	5.33E-04	3.20E-03	1.60E-03	1.62E-02
3650.00	5.69E-05	3.41E-04	1.70E-04	2.86E-03
18250.00	9.48E-13	5.69E-12	2.84E-12	5.23E-06

EFF.COMM.DOSE EQUIV.(SV) PER BQ INTAKE

INGESTION F1= 5.E-2	INGESTION F1= 3.E-1	INHALATION CLASS W	Y
1.70E-09	7.00E-09	8.00E-09	4.10E-08

CONTINUOUS INTAKE

FINAL OR EQUILIBRIUM BODY BURDEN
(BQ BODY BURDEN PER BQ/DAY INTAKE)

6.20E+00	2.90E+01	2.50E+01	1.20E+02

EFF. DOSE-EQUIV. RATE CONVERSION FACTOR
(SV/Y PER BQ BODY BURDEN)

1.00E-07	8.81E-08	1.17E-07	1.25E-07

Fig.1. Body burden programme output for ^{60}Co

THE ACCURACY OF DOSE ASSESSMENT FOR EXTERNAL β-γ-RADIATION

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INTRODUCTION

Doses* to the body and skin from β- and γ-radiations are generally estimated from readings of dosimeters carried by exposed persons. The relationship between dose and reading may be expressed:

$$\text{Dose} = \text{Dosimeter Reading} \times S \times F.$$

where $$S = \frac{\text{measurement quantity at the dosimeter}}{\text{dosimeter reading}}$$

where $$F = \frac{\text{Dose}}{\text{measurement quantity at the dosimeter}}$$

The measurement quantity may be exposure, air dose, air kerma-in-air, or, in the case of β-rays, tissue dose at defined depths below the surface. Thus S is a measure of the response of the dosimeter to a known amount of radiation delivered under laboratory conditions. F, on the other hand, is a relationship between two measurement quantities and is independent of the dosimeter properties.

The fractional error in the dose estimate is thus the sum of the fractional errors in S and F. In order to set limits of uncertainty in dosimeter response it is necessary to have a realistic estimate of the uncertainties in F when measurements are made in the field. Until now we have had only measurements in the laboratory[1] and calculations[2] which relate only to defined conditions to estimate uncertainties.

If a certain value of F is chosen based on laboratory measurements or calculations are made errors will arise in field measurements because inevitably the conditions of irradiation will differ from the ideal. Three questions should be asked and answered about these errors.

1. How large can these errors be?
2. How large are they likely to be?
3. What can be done to diminish them?

This paper deals with these three questions.

LABORATORY DATA

The ratio F of the dose to the wearer of a dosimeter to the measurement quantity at the dosimeter depends upon the geometries[1,5] of radiation source, dosimeter and its wearer and the energy of the radiation (β or γ).

*Dose is used here generally for the terms absorbed dose, dose equivalent or effective dose equivalent[3,4] as appropriate.

In the laboratory these factors can be varied independently and the consequence measured[1]. Thus, it is possible to measure the extent of error possible assuming these factors are unknown.

For example, if an effective dose equivalent (H_E) of 1 sievert is assigned to the wearer of a dosimeter exposed to 1 gray (kerma in air) of γ-rays the following has been noted[1].

- If the dosimeter is worn at the waist front the error in F will be less than 10% if the radiation is from all around or from in front and the energy is above 5 keV.
- If the radiation is from behind H_E is underestimated by a factor of four at 50 keV and a factor of two at 600 keV.

It is plausible that radiation from in front or from all around, or a combination of these source geometries is much more probable than radiation from behind. A worker generally faces his work and, if it is radioactive, it will irradiate him from in front, directly, and from around by scattering. However, laboratory experiments cannot teach anything about the probability of radiation direction in the working situation.

It is therefore necessary to make tests in the working environment since the laboratory experiments show that while errors in the 'plausible' situations are small ($\sim$10%), errors in a possible situation are quite large ($\sim$100-300%) and result in serious underestimates of dose.

In the case of β-radiation the errors can be much worse. If the source is behind the worker with a dosimeter at the front the failure to detect β-radiation is effectively total.

FIELD EXPERIMENTS

Two kinds of experiments can be performed in the working environment. By equipping workers with dosimeters placed on different body surfaces, data can be obtained about the direction of the radiation and by loading a realistic phantom with dosimeters at specific tissue sites and placing it at typical working positions and orientations in the working environment the ratio of dose to dosimeter measurement quantity (F) can be measured directly.

Workers in the lower header room of the NRX reactor at the Chalk River Laboratories were equipped with TL dosimeters at the front and back of their waists on 80 occasions. On 48 of these, when the doses were appreciable ($\geq$0.5 mSv), the dosimeters at the back always received less than those at the front, the mean ratio of their readings being 0.64 with a standard deviation of 0.18. This showed that the radiation was primarily from in front.

In the same working area a realistic phantom was loaded with dosimeters, internally and externally and the dosimeters read after exposure. The dosimeter reading at the back of the waist was 58% of that at the front which was typical of the ratios found on workers in that area. Using the dosimeter at the waist front, the effective dose was estimated to be 6 mSv which was 18% higher than the effective dose observed with dosimeters sited throughout the phantom. [It is the practice at CRNL to assign an effective dose of 1 Sv for

1 Gy (kerma in air) to the dosimeter site].

The phantom experiment was next located at a place where the radiation source, a set of contaminated effluent filters, was behind the phantom and where the rear dosimeter reading was 2.7 greater than that at the waist front. In this case, the effective dose was underestimated by the dosimeter worn in the normal position by 47%. Had the direction of irradiation been known and the dosimeter been worn at the rear H_E would have been over-estimated by 49%.

Measurements reported by Walsh and Johns[6] on a phantom, set at eleven places in Ontario Hydro Nuclear Power Stations showed that H_E was over-estimated by 14 and 23%, for males and females respectively, if a mean, weighted for dose at each place (in the Nuclear Power Station) was calculated. At no place was the dose under-estimated by more than 17% or over-estimated by more than 26%.

THE USE OF TWO DOSIMETERS

The figure shows the relationship between H_E and the mean air kerma to two dosimeters, one at the waist front and one at the waist[1] rear. These measurements were made in the laboratory experiments referred to before. In this case, the ratio of the readings of the dosimeter could be used to determine the principal direction of the radiation. When this is done an appropriate factor can be chosen for converting the dosimeter kerma to H_E. Above 50 keV the conversion factor varies less than 15% regardless of energy and direction.

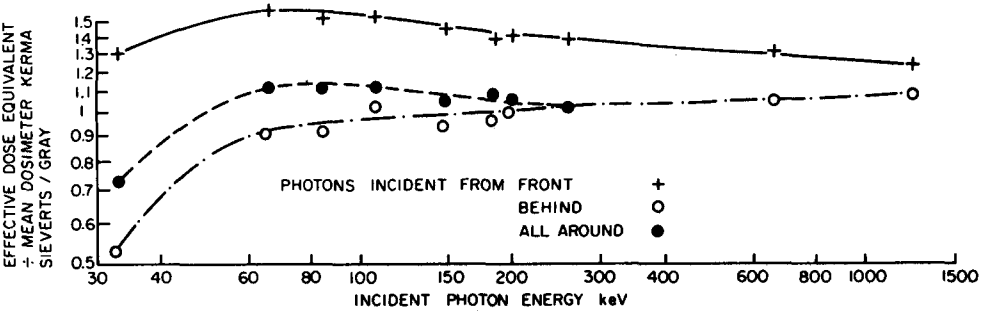

Figure 1. The dependence of energy and direction of the dose calculated from the mean of two dosimeter readings.

In the field measurements made at CRNL it was found that the error in estimating the effective dose, using two dosimeters, was only 6 and 3% when the radiation was from in front and behind respectively.

The approach to dosimetry illustrated in the figure can be taken without knowledge of the irradiation direction and as can

be seen in the figure extreme errors can be eliminated.

However, it is recommended that two dosimeters be worn in situations where appreciable exposure can be anticipated and where the direction of irradiation is not known. A comparison of the two readings can be made to identify the small proportion of exposures where the dominant component is from behind. Then, in only those cases an appropriate correction need be made. In the case of skin dose, the larger of the two estimates should be used because the body is opaque to β-rays.

ACKNOWLEDGEMENTS

I wish to thank J.A. Bond for organizing the multiple badging of exposed radiation workers and Mr. A. Ohno for his help in the measurement of dosimeters carried on phantoms.

REFERENCES

[1] Jones, A.R. and Marsolais, R. External Gamma Dosimetry in the Light of ICRP-26. Transactions of the American Nuclear Society 32, 659-660 (1979).

[2] Jones, T.D., Auxier, J.A. and Snyder, W.S. Dose to Standard Reference Man from External Sources of Monoenergetic Photons. Health Physics 24, 241-255 (1973).

[3] Jacobi, W. The Concept of Effective Dose - A Proposal for the Combination of Organ Doses. Rad. and Environ. Biophys. 12, 101-109 (1975).

[4] ICRP Publication 26. Recommendations of the International Commission on Radiological Protection, Pergamon Press, Oxford, (1977).

[5] Ambiger, T.Y. Phantom Measurements to Determine the Most Suitable Site for the Personal Dosimeter. Atomic Energy of Canada Limited, Chalk River, Ontario, AECL-3379 (1969).

[6] Walsh, M.L. and Johns, H.E. Measurement of the Effective Dose Equivalent in a Nuclear Reactor Environment with a Personal Thermoluminescent Dosimeter. Health Phys. (to be published).

DEVELOPMENTS IN THIN FILM DOSIMETRY

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A thin film thermoluminescent dosimeter (TLD), formed by incorporating the TL phosphor into a plastic matrix of polyethersulphone (PES) (1) has been developed, (2). The advantages of a thin film are reduction of the correction for cavity effects and self-shielding, and reduction in the perturbation of the radiation flux especially at interfaces.

PES is available as pellets or powder, and is resistant to attack by most chemicals. The material dissolves slowly in ketones, esters and some halogenated carbons. In its use as a matrix for the phosphor it is spread into a thin film, incorporating the TL phosphor (usually LiF) at a much higher loading than in PTFE, without any serious loss in sensitivity. A further advantage of these TLDs is that the PES can be dissolved in the solvent using a washing and filtering technique and thus the phosphor may be reclaimed.

The thin film TLD originally developed (the PES/LiF TLD), used the novel method of solvent evaporation to manufacture the film. This has been modified to ensure more control over the product with the aim of achieving the following specifications:-

1. A phosphor grain size of 30μm ± 10μm (LiF loses sensitivity rapidly below 20μm grain size (3).
2. The LiF and PES should be homogeneously mixed.
3. Phosphor loading as high as possible.
4. Uniform film thickness within ± 10%.
5. Physical toughness in order to withstand normal working conditions.

For the work reported in this paper TLD 700 was obtained as a powder which was processed by repeated ball milling and sieving. Grains of phosphor between the 40μm and 20μm mesh sieves were used for dosimeter manufacture and the yield of usable phosphor was approximately 30% of the original weight. A 60% loading of phosphor was achieved with air dried PES dissolved in a constantly stirred 1:1 by volume mixture of Xylene and n-methyl-2-pyrrolidone.

The mixture is spread into a thin film using the Sheen Applicator, a hand tool used in the paint industry for spreading films of paint as test pieces. The thickness of the film is set by the metal shim thickness. All surfaces of the applicator are cleaned with acetone before use. The TLD mixture is deposited on one end of the glass plate and spread in one movement with a straight edged blade. The film is left to dry in air at room temperature for twentyfour hours, after which it is easily peeled off the glass. The dosimeter is cut to the desired size using a metal punch. Each film provides approximately 70 dosimeters of 90μm ± 10% thickness and with an average weight of 6 mg ± 10% at a diameter of 1 cm.

ANNEALING

A common difficulty with thin film dosimeters is the damage which can occur during annealing. The matrix may distort or even melt in the high temperature required for annealing.

The standard anneal cycle for LiF is 1 hour at 400°C followed by 16 to 24 hours at 80°C. Unfortunately the highest temperature that the PES plastic is

claimed to withstand without distortion or discolouration (1) is 200°C. Slightly higher temperatures can be used when the PES is used as a matrix for the dosimeter. Alternative annealing regimes were studied as follows:-

Regime 1 Anneal completed dosimeter for 1 hour at 255°C, the highest temperature at which the TLDs would just survive for 1 hour, followed by 16 to 24 hours at 80°C.
Regime 2 Anneal LiF powder for 1 hour at 400°C followed by 16 to 24 hours at 80°C, before film manufacture. The TLD is used with no further annealing.
Regime 3 Anneal LiF powder for 1 hour at 400°C followed by 16 to 24 hours at 80°C. After film manufacture anneal the TLDs for 1 hour at 200°C.

TLDs from each film and regime were given a calibration irradiation of 410R by a Co^{60} source. During irradiation the TLDs were placed in a perspex TLD holder, 6mm thick, and then covered by a 50μm sheet of black plastic.

The TLDs were read out in a Toledo 654 TLD reader, utilising the standard read cycle of:

16s preheat at 135°C
16s read at 240°C
16s anneal at 300°C

The reproducibility of each regime was compared, for fresh and reclaimed powder, so that a recommended annealing technique for the PES/LiF TLD could be established.

The criteria for the readout of background and irradiated TLDs are as follows:-

1. The standard deviation (σ) of readout of the calibrated TLDs must be less than 10% of the average value of readout.
2. Background readout should be less than or equal to 5% of the average value of readout of the irradiated TLDs.
3. The lower detection limit, equal to 3σ of the background should be less than 1% of the average readout of the irradiated TLDs.

Table 1 gives the background and calibration readout expressed in counts per gramme of TLD.

Regime 3 met these requirements for both fresh and reclaimed powder separately and also as a mixed batch so the recommended method of use for PES/LiF TLDs is:-

1. Anneal LiF in powder form → 2. Film manufacture → 3. Cut out TLDs and anneal (200°C)
↑ ↓
6. Dissolve in solvent and reclaim LiF ← 5. Readout and weigh ← 4. Irradiate

Possible improvements include:-

1. A longer anneal after manufacture (ie 80°C for 16 to 24 hours).
2. Reject TLDs exceeding 100μm in thickness or improve the shim method

of thickness control.

GAMMA MEASUREMENTS IN A REACTOR MOCK-UP

It was necessary to establish the thermal neutron response of the PES/LiF TLD (using TLD 700) before gamma measurements could be made in a reactor environment. The thermal neutron responses were determined in a graphite thermal column (cave D at the NESTOR reactor, AEE Winfrith). The gamma dose was subtracted by using measurements in a lithium carbonate ($^6Li_2CO_3$) pot which was installed in the graphite reflector and contained a graphite shell around the dosimeter. This gave a thermal neutron calibration of 1.3 R Co^{60} per 10^{10}n cm^{-2}. The calibration was repeated at CEN/SCK, Mol in the Cavity Standard Field of the BR1 reactor. In this case a lithium fluoride (^6LiF) pot was used and the calibration yielded 2.9 R Co^{60} per 10^{10} n cm^{-2}. This discrepancy was not important in this application but is being investigated.

The dosimeter was then compared with Beryllium Oxide dosimeters in the NESDIP (Nestor Dosimetry Improvement Programme) facility of the NESTOR reactor where the Pool Critical Assembly - a mock-up of a civil Pressurised Water Reactor shield was installed.

The results showed good agreement between the two dosimeters throughout a series of measurements in the water layer between the thermal shield and pressure vessel and through the steel pressure vessel. The maximum discrepancy between dosimeters was within 2σ in spite of the absence of cavity correction. BeO TLD (4) have been developed for use in graphite, to which they are well matched. In steel and water the mismatch in Z number will invalidate the calibrations of the TLDs which are performed in a Co^{60} field built up in graphite. For this reason the results must be regarded as interim while a fuller energy dependent calibration in different build up materials is being made and understood.

REFERENCES

(1) PES or Victrex - Manufactured by Polyethersulphone Plastics Group, ICI Plastics Division, Welwyn Garden City, AL7 1HD

(2) "A New Thin Film Dosimeter"
Health Physics 37, pp 417-419, 1979
D Lowe, J R A Lakey and A V Yorke

(3) "Effect of Particle Size on the Thermoluminescence of Lithium Fluoride"
Nature 221, pp 1047-1048, 1969
T E Burlin

(4) Private Communication. Dr T A Lewis
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Berkeley, Gloucestershire, GL13 9PB

TABLE 1 - COMPARISON OF TLDs AGAINST THE MANUFACTURING METHOD
(Exposure 410 roentgens, Co^{60})

Regime	1			2			3		
Source of LiF	Fresh	Reclaimed	Mixture	Fresh	Reclaimed	Mixture	Fresh	Reclaimed	Mixture
Standard Deviation (σ) (% Signal)	5.01	7.66	11.7	8.57	15.82	21.8	4.73	6.15	9.91
Background (% Signal)	2.5	0.93	1.58	0.93	1.5	1.26	0.72	0.41	0.55
Standard Deviation of Background (% Signal)	0.39	0.07	0.07	0.29	1.36	1.36	0.24	0.02	0.02

INTERCOMPARISON OF SOME PERSONNEL DOSIMETERS IN THE MIXED GAMMA AND NEUTRON FIELDS

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INTRODUCTION

The personnel dosimeters such as film badges and thermoluminescence dosimeters (TLDs) are commonly used for radiation protection purposes as to penetrating radiation, gamma-rays and neutrons. However, gamma-ray and neutron doses obtained with these dosimeters are evaluated by each institute and/or dosimeter vendor using their original calibration factors and procedures. Therefore, personnel dosimetry intercomparison is very important not only for the improvement of dosimeter performances but also for the standardization of dose evaluation procedures. Much problems remain in neutron dosimetry rather than in gamma-ray dosimetry. Especially, neutron energy spectrum and mixed gamma-rays can affect neutron dose measurement and its evaluation procedure. In order to estimate the effects of these factors on measurements of neutron dose equivalent, an intercomparison of dose equivalent responses to thermal neutron for some kinds of personnel dosimeters was carried out under a variety of mixed field spectral conditions at Kyoto University Reactor (KUR) and Kinki University Reactor (UTR-KINKI). In this paper, some problems related to the precision and accuracy in the measurements of thermal neutron dose equivalent were investigated from the practical point of view.

EXPERIMENTAL

Five different mixed gamma and neutron fields at KUR and UTR-KINKI were used. These fields have different cadmium ratios and n/γ ratios, respectively. The specifications of the mixed fields are as folloes:

(1) mixed field I with dimensions of 300 x 300 x 300 cm at KUR heavy water facility,
(2) mixed field II with dimensions of 50 x 50 x 50 cm at KUR horizontal exposure tube No 2,
(3) mixed field III, the same as mixed field II, but different cadmium ratio and n/γ ratio from mixed field II,
(4) mixed field IV with dimensions of 20 x 20 x 10 cm, composed of bismuth scatterers and LiF tiles in the graphite reflector region at UTR-KINKI,
(5) mixed field V, the same as mixed field IV, but different cadmium ratio and n/γ ratio from mixed field IV.

The radiation characteristics of the mixed fields are shown in Table 1. The basic measurements made during this study were gamma-ray exposure rate, thermal neutron flux, cadmium ratio, n/γ ratio ($cm^{-2} \cdot mR^{-1}$) and n/γ ratio based on dose equivalent. Gamma-ray exposure rate was measured by TLD, and thermal neutron flux and cadmium ratio by gold foil activation method. The exposure dose is equal to the dose equivalent for gamma-ray, but the thermal neutron dose equivalent is assessed using a reference value of 936000

fluence per 1 mrem recommended by ICRP (1).

Table 1 Radiation characteristics of the mixed gamma and neutron fields at KUR and UTR-KINKI

mixed field	gamma-ray exposure rate (R/hr)	thermal neutron flux ($n/cm^2 \cdot sec$)	Cd ratio	n/γ ratio ($cm^{-2} \cdot mR^{-1}$)	(based on dose equivalent)
I	$1.75 \times 10^{-2} \sim 3.40$	$3 \times 10^{4} \sim 7 \times 10^{6}$	5000	6.91×10^{6}	7.41
II	6.22	2.59×10^{6}	491	1.50×10^{6}	1.60
III	6.35	3.88×10^{5}	252	2.20×10^{5}	0.235
IV	$4.11 \sim 4.70 \times 10^{-2}$	$2.57 \sim 3.13 \times 10^{5}$	5.9	2.37×10^{7}	25.3
V	$9.84 \sim 10.9 \times 10^{-2}$	$9.38 \sim 11.4 \times 10^{4}$	5.3	3.54×10^{6}	3.79

Dosimeters used for thermal neutron measurements are three film badges (hereafter denoted as dosimeter A, B, and C, respectively), and two TLDs (hereafter denoted as dosimeter D and E, respectively), commonly used in Japan. Each dosimeter was exposed in the every mixed fields under identical conditions. The evaluation of thermal neutron dose equivalent with each dosimeter was carried out by dosimeter vendors, respectively.

RESULTS AND DISCUSSION

As seen in Table 1, in these mixed fields cadmium ratio and n/γ ratio based on dose equivalent vary from 5000 (mixed field I) to 5.3 (mixed field V) and from 25.3 (mixed field IV) to 0.235 (mixed field III), respectively. Hence, we studied the effects of factors such as cadmium ratio and mixed gamma-ray dose on neutron dose equivalent measurement with each dosimeter.

Effect of cadmium ratio and n/γ ratio on thermal neutron dose equivalent measurement: when thermal neutron dose equivalent response (merm/fluence) of dosimeter was defined as a ratio of thermal neutron dose equivalent assessed with dosimeter to thermal neutron fluence measured by gold foil activation, its own trend in the response of dosimeter was obtained, although no systematic trend was found between film badges and TLDs.

Dosimeter A showed no definite trend in the responses with changes of cadmium ratio and n/γ ratio in the irradiation fields, and the responses vary from 1.83×10^{-6} (mixed field I and II) to 1.22×10^{-6} (mixed field III).

Dosimeter B showed no definite trend in the responses with changes of cadmium ratio and n/γ ratio in the irradiation fields, and the responses vary from 8.80×10^{-7} (mixed field II) to 5.04×10^{-7} (mixed field IV). It should be noted that in mixed field III, which is of the highest gamma-ray contamination, no significant neutron dose equivalent could be obtained with dosimeter B.

Dosimeter C showed no definite trend in the responses with changes of cadmium ratio and n/γ ratio in the irradiation fields, and the responses vary from 1.21×10^{-6} (mixed field I) to 9.74×10^{-7} (mixed field IV).

Dosimeter D showed that the response decreased from 2.10×10^{-6} (mixed field I) to 1.24×10^{-6} (mixed field III) as decrease in cadmium ratio and increase in gamma-ray contamination, although no

results were obtained in the mixed field IV and V, because of no participation.

Dosimeter E showed no definite trend in the responses with changes of cadmium ratio and n/γ ratio in the irradiation fields, and the responses vary from 1.38 x 10^{-6} (mixed field I) to 1.08 x 10^{-6} (mixed field III).

From above mentioned results, it may be concluded that each personnel dosimeter used in the experiments have fairly good performances for radiation protection purposes. The thermal neutron dose equivalent responses of the dosimeters have no large dependency on cadmium ratio (5000∿5.3) and gamma-ray contamination (n/γ ratio 25.3∿0.235 based on dose equivalent) except for dosimeter D, and the changing ratio of the measured values by individual dosimeters were within 1.7 (see in Fig.1 and 2).

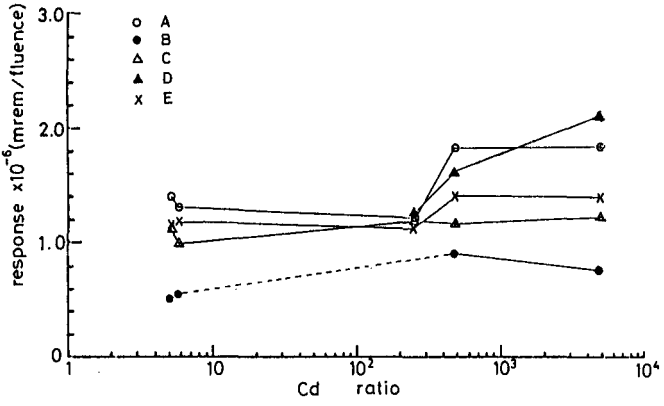

Fig.1 Comparison of thermal neutron dose equivalent response as a function of cadmium ratio

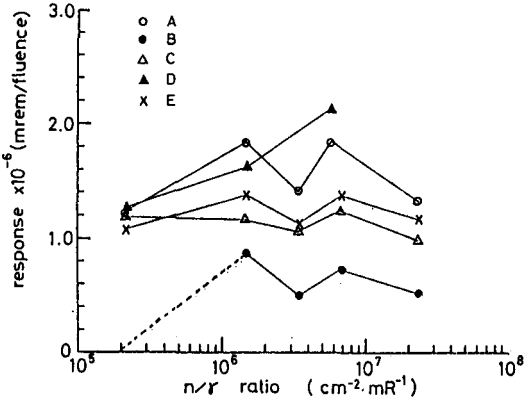

Fig.2 Comparison of thermal neutron dose equivalent response as a function of n/γ ratio

Table 2 shows a summary of the relative thermal neutron dose equivalent of individual personnel dosimeters normalized to ICRP recommended value as a reference value (1). The normalized thermal neutron dose provides a measure of accuracy of the measured value relative to the reference value. The percent standard deviations from the mean are measures of precision which describe agreement among individual measurements. As seen in Table 2, no consistent relationship between dosimeter asscracy or precision and incident cadmium ratio or gamma-ray contamination is indicated as reported by Sims et al. (2). The Nuclear Regulatory Commission (NRC) guidelines for personnel neutron dosimeters (3) suggest that dosimeters used in the dose equivalent range in mrem order should be accurate to within ±50 % and that the standard deviation (a measure of precision) should be within ±30 % of the mean. From the results shown in Table 2, dosimeter A, B (film badges) and dosimeter D (TLD) fail to meet the NRC accuracy guidelines in two of the five mixed fields. These results indicated that there is a difference of about 4 times in the measured values relative to the reference dose equivalent. However, all the dosimeters meet the precision of ±30 % standard deviation of the mean. The large difference of about 4 times in measured values relative to the reference value recommended by ICRP may be caused by different calibrations among the dosimeter vendors, since there is no large dependency on cadmium ratio and n/γ ratio in dose equivalent responses.

From the results obtained, it is suggested that the standardization of calibration of dosimeters is required from the practical point of radiation protection and safety.

Table 2 Comparison of relative thermal neutron dose equivalent and percent standard deviations of personnel dosimeters in various kinds of mixed gamma and neutron fields

personnel dosimeter	mixed field I	mixed field II	mixed field III	mixed field IV	mixed field V
A	1.71[1] (15.2)[2]	1.71 (10.5)	1.14 (17.5)	1.24 (15.3)	1.32 (15.2)
B	0.686 (19.7)	0.824 (10.3)	3)	0.487 (7.8)	0.472 (13.6)
C	1.13 (25.9)	1.08 (23.1)	1.11 (25.2)	0.912 (20.5)	0.983 (21.9)
D	1.97 (11.2)	1.50 (13.3)	1.16 (11.2)	4)	4)
E	1.29 (17.1)	1.28 (13.3)	1.01 (14.9)	1.10 (16.4)	1.06 (9.4)

1) relative thermal neutron dose equivalent
2) percent standard deviation
3) less than detectable limit
4) not participated

ACKNOWLEDGEMENT

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REFERENCES

(1) ICRP Publication 21 (1971): Data for Protection against Ionizing Radiation from External Sources.
(2) C.S.Sims and R.E.Swaja (1982): Personnel Dosimetry Intercomparison at the Health Physics Research Reactor: a Summary (1974-80), Health Physics,42,3.
(3) Nuclear Regulatory Commission (1977): Personnel Neutron Dosimetry

METHODOLOGIE DU CONTROLE DOSIMETRIQUE INDIVIDUEL DE L'IRRADIATION EXTERNE

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1. INTRODUCTION

Dans le cadre de l'homologation des services chargés en Suisse de la dosimétrie individuelle, une méthode uniforme pour la calibration des systèmes dosimétriques et l'évaluation des doses s'est avérée nécessaire et un groupe de travail a été chargé d'élaborer des recommandations dans ce domaine.
Les grandeurs à respecter, sur lesquelles sont fixées les limites de l'irradiation des personnes exposées aux radiations dans l'exercice de leur profession ("grandeurs limitées"), retenues dans le cadre de cette recommandation sont celles proposées par la CIPR dans sa publication 26 (1), à savoir :

- les équivalents de dose aux divers organes (H_T) pour la protection contre les effets de nature non stochastique
- l'équivalent de dose effectif (H_E) en vue de maintenir le taux des effets stochastiques à un niveau acceptable.

Pour leur évaluation, la personne porte, durant son activité professionnelle, un ou plusieurs dosimètres dont la mesure est effectuée périodiquement, en général chaque mois. La détermination des grandeurs limitées se rapportant à la personne, à partir de la mesure de son ou de ses dosimètres, pose un problème relativement complexe pour les raisons suivantes :

- le dosimètre mesure une dose à l'endroit où il est placé et ne peut donner qu'une valeur "représentative" des doses aux divers organes
- les conditions d'irradiation sont très variables : nature, énergie, variation spatiale et temporelle et directionalité du champ de radiation
- la morphologie est différente d'une personne à l'autre.

2. METHODOLOGIE

La valeur indiquée par le dosimètre ne permet de calculer la valeur des grandeurs limitées que lorsque toutes les conditions décrites ci-dessus sont précisées. Ceci n'est pas le cas dans la pratique. Afin de combler cette lacune et de définir une méthodologie dosimétrique cohérente, les trois démarches suivantes s'avèrent nécessaires :

- définition de grandeurs intermédiaires. Ces grandeurs font la liaison entre les grandeurs limitées et les grandeurs mesurées et peuvent être qualifiées comme suit :
  - . d'une part, elles sont représentatives des grandeurs limitées. Les grandeurs limitées peuvent en être déduites par une interprétation, certes entachée d'erreur, mais simple. Cette erreur, généralement conservative, peut être chiffrée pour toute situation d'irradiation,
  - . d'autre part, elles sont l'objectif de la mesure dosimétrique. Ainsi, elles servent de valeurs de référence lors de la calibration des dosimètres et le rapport entre grandeurs mesurées et grandeurs intermédiaires doit être le plus indépendant possible des conditions d'irradiation.
- définition de conditions d'irradiation. Ces conditions doivent être reproductibles et représenter un "standard" des conditions générales d'irradiation. Dans le cadre de cette démarche, le rapport entre les grandeurs mesurées et les grandeurs intermédiaires doit être constant et représenter une valeur moyenne acceptable pour l'ensemble des conditions réelles d'irradiation.
- standardisation des paramètres liés à la personne contrôlée. Il est évident qu'une dosimétrie individualisée, dans le sens qu'elle tienne compte de la morphologie de la personne contrôlée, n'est pas possible. La standardisation des paramètres liés à la personne consistera à définir un fantôme d'irradiation suffisamment représentatif de toute personne dans des conditions d'irradiation standard.

L'interprétation de la mesure sera alors effectuée de deux manières, selon le niveau de la grandeur intermédiaire déterminée :

- aux faibles valeurs (inférieures aux limites), la valeur de la grandeur intermédiaire, calculée à partir des grandeurs mesurées et à l'aide d'un algorithme défini lors de la procédure de calibration, est choisie comme valeur de la grandeur limitée.
- aux valeurs élevées (à partir des limites), une enquête est effectuée qui a pour but de déterminer les conditions particulières d'irradiation. La détermination des grandeurs limitées à partir de la grandeur intermédiaire est alors effectuée de manière spécifique.

3. CHOIX DES GRANDEURS INTERMEDIAIRES

Parmi les grandeurs intermédiaires, celles qui caractérisent le champ de radiation (kerma, exposition, etc.), ainsi que les indices d'équivalent de dose (2) n'ont donc pas pu être retenues. Par contre, le concept des équivalents de dose à différentes profondeurs satisfait à nos conditions. En effet, le rapport entre grandeurs mesurées et grandeurs intermédiaires est largement indépendant les conditions d'irradiation. D'autre part, ces grandeurs sont bien représentatives des grandeurs limitées.

L'évaluation directe des grandeurs limitées à partir des mesures n'a pas été retenue. L'inconvénient de cette méthode est l'absence d'un objectif concret pour la mesure et la nécessité de recourir, lors de la calibration des systèmes, à des méthodes de Monte Carlo dont l'accès est complexe et les résultats trop dépendants des codes de calcul utilisés.

Les grandeurs intermédiaires retenues et l'attribution des doses aux divers organes à partir de la mesure sont données au tableau 1. Dans le cas de l'irradiation des extrémités, la méthodologie proposée, essentiellement pragmatique, consiste dans le choix judicieux du type de dosimètre (mesure en surface ou en profondeur) et de la manière de le porter (bague, bracelet, etc.) en fonction des conditions particulières d'irradiation. En général, la surveillance de la dose au cristallin n'est pas nécessaire, car la valeur limite pour cet organe ne peut être atteinte sans que la limite de dose au corps entier ou celle de la peau ne soient dépassées (3).
Dans le cas d'une irradaition partielle de l'organisme, une mesure complémentaire sur l'organe particulièrement exposé peut s'avérer nécessaire. On utilisera donc comme grandeur intermédiaire l'équivalent de dose à une profondeur massique de 1 $g.cm^{-2}$ au niveau de l'organe en question.

Tableau 1. Grandeurs intermédiaires et valeurs des grandeurs limitées associées

Nature de l'irradiation	Grandeurs intermédiaires	Relation au grandeurs limitées
Irradiation du corps entier	H_p: équivalent de dose à 1 $g.cm^{-2}$ au niveau de la poitrine H_s: équivalent de dose à 7 $mg.cm^{-2}$ au niveau de la poitrine	$H_T = H_p$, pour chaque organe T, à l'exception de la peau $H_{peau} = H_s$ $H_E = H_p$
Irradiation des extrémités	$H_{extr,p}$: équivalent de dose à 300 $mg.cm^{-2}$ au au niveau de la partie irradiée de l'extrémité $H_{extr,s}$: équivalent de dose à 7 $mg.cm^{-2}$ au niveau de la partie irradiée de l'extrémité	$H_{extrémités}$ = max ($H_{extr,p}$, $H_{extr,s}$)
Irradiation du cristallin	H_i: équivalent de dose à 300 $mg.cm^{-2}$ au niveau de la poitrine	$H_{crist} = H_i$
Irradiation partielle de l'organe T	$H_{T,p}$: équivalent de dose au nivau de l'organe	$H_T = H_{T,p}$

4. CONDITIONS STANDARD D'IRRADIATION

La calibration des systèmes dosimétriques est basée sur les conditions standar d'irradiation : nature, énergie, variation spatiale, directionalité du champ de radiation et fantôme. L'utilisation d'un standard unique de radiation n'est pas admissible et le système dosimétrique doit être testé pour l'ensemble des rayonnements intervenant dans l'application en question et répondre dans chaque cas aux performances exigées.
La calibration sera alignée sur une radiation de référence, dans notre cas le rayonnement γ du ^{137}Cs.

Le champ d'irradiation standard est étendu (c'est-à-dire qu'il a la même radiance et le même spectre énergétique en chaque point de l'espace), monodirectionnel et dirigé en incidence frontale sur le fantôme.

La spécification détaillée du fantôme n'est pas critique, puisqu'il s'agit d'une incidence frontale. On choisira ainsi le fantôme le plus simple à réaliser et représentant pour le rayonnement en question une équivalence au tissu suffisante. Dans le cas du rayonnement X ou γ, un bloc parallélipipèdique de plexiglas s'avère adéquat.

Les équivalents de dose seront mesurés dans le fantôme par ionométrie selon les procédures recommandées par la CIUR.

5. CONCLUSIONS

En l'absence de directives de la CIUR, la méthodologie proposée permet de standardiser la procédure de calibration des systèmes dosimétriques. Au cours de l'élaboration de cette recommandation, le groupe de travail a eu le souci constant d'une part de proposer une méthode allant dans le sens des démarches effectuées sur le plan international, d'autre part de rester les pieds sur terre en choisissant des concepts pragmatiques.

BIBLIOGRAPHIE

1. International Commission on Radiological Protection. Publication No 26, Pergamon Press, Oxford (1977).

2. International Commission on Radiation Quantities and Units. Publication No 25, Washington, D.C. (1976).

3. Commission électrotechnique internationale. Mesures d'équivalent de dose et de débit équivalent de dose β, χ, γ, utilisables en radioprotection (projet de février 1982).

IMPLICATIONS OF DEPTH DOSE EQUIVALENT CONCEPT IN PERSONNEL DOSIMETRY

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ABSTRACT

Current regulations in the United States require measurement of dose equivalent for protection at tissue depths of 1.0 g/cm^2 and 7 mg/cm^2 as prescribed in ANSI N13.11-1982 standard. This paper discusses a few implications of the depth dose equivalent requirements on the design of the personnel dosimeters.

INTRODUCTION

A standard[1] has been developed by the Health Physics Society Standards Committee and adopted by the American National Standards Institute as ANSI N13.11-1982. The purpose of the standard was to develop a uniform procedure to test the performance of personnel dosimetry processors throughout the United States. In this standard for personnel protection dosimetry, radiation dose depths of 1.0 g/cm^2 and 7 mg/cm^2 in tissue were adopted for deep and shallow dose equivalent estimations, respectively. Since it is not practical to determine the dose equivalent quantities each time a dosimetry measurement and calibration are performed, a set of conversion factors ($C_{x,d}$, $C_{x,s}$) was specified in the standard for converting from photon air exposure values (R) in which radiation dosimetry instruments are calibrated to the appropriate deep- and shallow-dose equivalent (rem) quantities. Dose equivalent conversion factors are not necessary in the case of beta particles if a source calibrated in terms of absorbed dose in a tissue equivalent phantom at a depth of 7 mg/cm^2 is used.

At the request of the U. S. Nuclear Regulatory Commission, the National Bureau of Standards under the National Voluntary Laboratory Accreditation Program (NVLAP) is organizing an accreditation testing of processors of personnel radiation dosimeters. The proficiency testing will be conducted at the University of Michigan. All participating organizations are required to demonstrate satisfactory performance in accordance with the ANSI N13.11-1982 standard as one condition of accreditation. As a result of this program, impending uncertainties and problems are receiving close scrutiny.

Two element personnel dosimeters, having two thick (0.9 mm or 240 mg/cm^2) TLD ribbons, are widely used for radiation dosimetry. In these dosimeters, one of the elements is covered with a few hundred mg/cm^2 (up to 1 g/cm^2 in some dosimeters) filtering material the TL data from which is used to estimate the deep-dose equivalent (DDE) due to penetrating photon radiation. The other element in the dosimeter is behind a thin filter (normally less than 50 mg/cm^2) and provides the TL data necessary to determine the shallow-dose equivalent (SDE) or whole body skin dose due to beta and photon radiations. With the introduction of the depth dose estimation requirements, it was speculated by many dosimetry processors that, in order to obtain correct DDE and SDE values, the active dosimetry

elements should be kept behind 1.0 g/cm^2 and 7 mg/cm^2 tissue equivalent material, respectively. This may not be necessarily true due to energy dependent response of these dosimeters to beta-gamma radiations, and, in case of beta particle dose determination, due to a large thickness of TLD ribbons. A propagation of systematic error therefore occurs whenever the difference in workplace and calibration radiation energies exists.

The formulae correlating the dosimeter responses and DE conversion factors at different energies are discussed in this paper by taking into account the guidelines of ANSI N13.11-1982. To clarify the implications, examples of two dosimeters which use different amount of filtering materials are also presented. For detailed derivation of the formulae, the reader is referred to the pertinent literature[2,3].

PROCEDURE AND FORMULATION

Personnel dosimetry calibration measurements are performed in air using Cs-137 and Sr/Y-90 point sources, then converted to the dose equivalent quantities by multiplying by a proper conversion factor. When all the exposures are performed on proper tissue equivalent phantoms[1], to obtain correct DDE and SDE values, the formulae correlating the dosimeter response and dose equivalent conversion factors at different photon energies are as follows:

$$C_{x,d}(E) = 1.03 \cdot \frac{T_d(E)}{T_d(Cs)} \quad \text{and} \quad C_{x,s}(E) = 1.03 \cdot f \cdot \frac{T_s(E)}{T_s(Cs)}$$

where C's are the dose conversion factors for photons of energy E and T's are the TL readings of the ribbons behind thick filter 'd', or behind thin-filter 's' of the field dosimeter when equal amount of photon exposures are given at energy E or exposed to Cs-137 photons. Factor $f = F_s(Sr)/F_s(Cs)$ correlates beta calibration factor, $F_s(Sr)$, and photon calibration factor, $F_s(Cs)$, for the ribbon behind the thin-filter. It mainly depends upon the thickness of thin filter. Numerical factor corresponds to the values of $C_{x,d}(Cs) = C_{x,s}(Cs) = 1.03$.

Rewriting the above equations, we have

$$C'_{x,d}(E) = T'_d(E) \quad \text{and} \quad C'_{x,s}(E) = f \cdot T'_s(E)$$

where C' and T' are normalized values of C and T by their corresponding Cs-137 values. Hence, for a correct estimation of the DDE and SDE for photons, these equations should be valid for all photon energies to which the dosimeter is going to be exposed in the field. Deviations larger than 10% indicate energy regions of poor dosimeter response.

Example of two dosimeters are presented. First dosimeter had about 630 mg/cm^2 (540 mg/cm^2 aluminum filter and 90 mg/cm^2 thick security photograph) and 5 mg/cm^2 (mylar) filters and second dosimeter had 1 g/cm^2 (30% boron nitride loaded plastic) and 10 mg/cm^2 (aluminized mylar) filters over TLD ribbons. Both dosimeters used 240 mg/cm^2 thick Harshaw TLD-700 ribbons. After exposing these dosimeters to well characterized photons (K x-rays and Cs-137) and beta particles (Pm-147, Tl-204, Sr/Y-90) of different energies on

proper phantoms, all ribbons were read on TLD reader and net TL values were determined by subtracting the background TL readings from each of the averaged TL values.

In Figure 1, the values of $C'_{x,d}(E)$, $C'_{x,s}(E)$, $T'_d(E)$ and $T''_s(E)[= f \cdot T'_s(E)]$ are plotted for $f = 0.9$ as a function of photon energy. I and II subscriptions indicate the two dosimeters. Deviations in the T' or T" plots from the corresponding C' plots will show the dosimeter response energy region where the dosimeter responses are not too good. In Figure 2, the variation of the thin-shielded ribbon response to beta radiation at 7 mg/cm^2 depth in tissue for the first dosimeter is given.

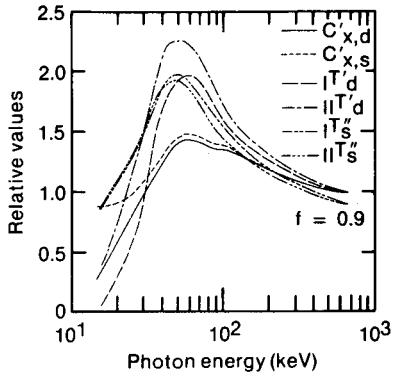

Figure 1. Plots of C's, T's and T"s as a function of photon energy.

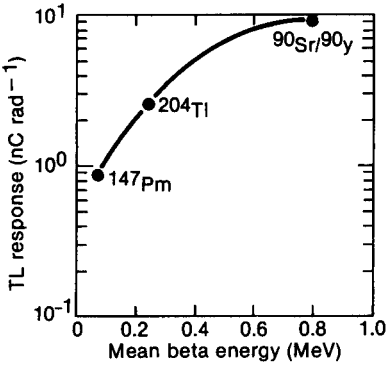

Figure 2. Beta response at 7 mg/cm^2 depth as a function of mean beta energy.

RESULTS AND DISCUSSION

It has been customary, in the case of photons, simply to equate the numerical values of exposure (R) and dose equivalent (rem) for personnel dosimetry. Yet, ICRP recommendations[4] for conversion factors have existed for some time. With the introduction of the requirement for depth dose in tissue, it was speculated by many dosimetry processors that, in order to obtain correct dose equivalent values, the TLD ribbons should be placed behind 1000 and 7 mg/cm^2 tissue equivalent plastic. This requirement cannot be satisfied simply by using those plastic thicknesses, since plastics are not necessarily tissue equivalent at energies below 100 keV. In addition, LiF TLDs do not respond linearly with energy below 200 keV. The DDE or SDE estimation requirements are fulfilled by using a properly designed dosimeter for which the normalized response $T'_d(E)$ or $T''_s(E)$ coincides with the $C'_{x,d}(E)$ or $C'_{x,s}(E)$, respectively for all the energies of interest.

In the example of two dosimeters presented here, after comparing the curve $C'_{x,d}(E)$ with the curve $_{I}T'_d(E)$ or $_{II}T'_d(E)$, it is

evident that dosimeter I with 630 mg/cm^2 filter would overestimate the DDE by as much as 37% in the range from 32 to 662 keV and underestimate up to about 80% in the range from 30 down to 16 keV. On the other hand, dosimeter II with 1 g/cm^2 filter would overestimate the DDE over the whole experimental range with a maximum value of about 68% near 45 keV. On comparing the curve $C'_{x,s}(E)$ with the $T''_s(E)$ curves, it is clear that the estimation of SDE by these dosimeters would be underestimated by a maximum of 10% over the range from 662 keV to about 200 keV and would be overestimated by a maximum of about 42% in the range from 17 to 175 keV. The SDE response of the first dosimeter to beta rays, which has a 5 mg/cm^2 thin window, is shown in Figure 2. The dosimeter response drops rapidly as the mean beta particle energy decreases. Therefore, the dosimeter introduces large systematic errors in the SDE esimations in the radiation fields which are markedly different from the penetrability of the beta radiation from the Sr/Y-90 source used for calibration. Other similar dosimeter designs with thicker windows, as is more common, would show even larger systematic errors.

In practice, a worker may receive beta-gamma doses in radiation fields generated by one or more beta-ray sources or photons of single or multiple energies or a mixture of beta and photon sources. An ideal dosimeter will always give correct dose values for all beta-gamma exposures independent of their energies. However, with practical thick-element dosimeters, there are several problems intrinsic to the estimation of DDE and SDE values. Depending upon dosimeter design, processing, and readout techniques, substantial errors could be made when the dosimeters are exposed to unknown fields, especially if it is softer than the calibration field.

ACKNOWLEDGEMENT

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REFERENCES

1. American National Standards Institute, "Criteria for Testing Personnel Dosimetry Performance", ANSI N13.11-1982, New York, 1982.

2. V. P. Gupta, "Rem Response of the Idaho National Engineering Laboratory Personnel Dosimeter to Photons," Health Physics, 42, 1982, p. 833.

3. V. P. Gupta, "Estimation of Shallow Dose Equivalent Using a Two-Element Dosimeter," Health Physics, 44, 1983, p. 373.

4. ICRP, Data for Protection Against Ionizing Radiation from External Sources, Supplement to ICRP Publication 15. Publication 21, Pergamon Press, ISBN 0 08 016872, 8, 1969.

DETERMINATION OF DOSE EQUIVALENT QUANTITIES IN A PHANTOM FOR CALIBRATION PURPOSES

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For the introduction of the new measurement quantity in radiation protection by external exposure which has been under discussion for some years [1], no primary standard exists at present. Conversion factors have therefore been determined, allowing the new quantities to be linked to the primary standards for air kerma or absorbed dose in water. Conversion factors presented in the literature are based on Monte Carlo calculations and only a few experimental results.

Measurements of the distribution of dose equivalent in the ICRU sphere were therefore performed, using a 0.8 cm^3 spherical ionisation chamber which was developed at the PTB. A PMMA hollow sphere, 30 cm in diameter and filled with water, served as a phantom.

For determining the energy dependence of the conversion factors in the X-ray range, heavily-filtered radiation was used. The small volume of the ionisation chamber and the filtering caused very small signals, while the leakage current was of the same order of magnitude. In order to separate the signal from the leakage current the X-ray beam was chopped and the pulsed ionisation current was measured using the lock-in technique.

With this arrangement depth dose curves in the phantom were obtained in the range from 30 kV up to 300 kV tube voltage and with ^{137}Cs and ^{60}Co γ-radiations. The ionisation chamber was calibrated to indicate the air kerma free in air. For calculating the dose equivalent, the different spectra in air and at the various depths in the phantom had to be considered.

The spectra in the phantom were obtained by segment-by-segment multiplication of the real X-ray spectra [2] with the interpolated spectra known from Monte Carlo calculations for mono-energetic radiation [3]. Two examples of heavily-filtered radiation for 30 kV and 300 kV tube voltage are shown in figures 1 and 2. In each case the points represent the spectrum of the incident X-ray beam, while the lines describe the spectra in the phantom with the depth as a parameter. For low energies the spectra in the phantom are similar in shape to that of the primary beam, but as a consequence of the photon scattering the mean energy at the surface is displaced to smaller values. The spectra decrease and flatten with increasing depth. At higher energies (fig. 2) the Compton effect becomes important, indicated by the second peak which appears at nearly half-energy of the first peak.

Using these spectra it is possible to calculate the dose equivalent from the response of the ionisation chamber in the phantom and the calibration free in air.

The following relation between the measured value M_i and the value m(E) which would have been caused by a monoenergetic spectral fluence giving rise to the same air kerma as the fluence φ^i can be considered:

$$M_i = \int_0^{E_{max,i}} \varphi_E^i(E)\ m(E)\ dE$$

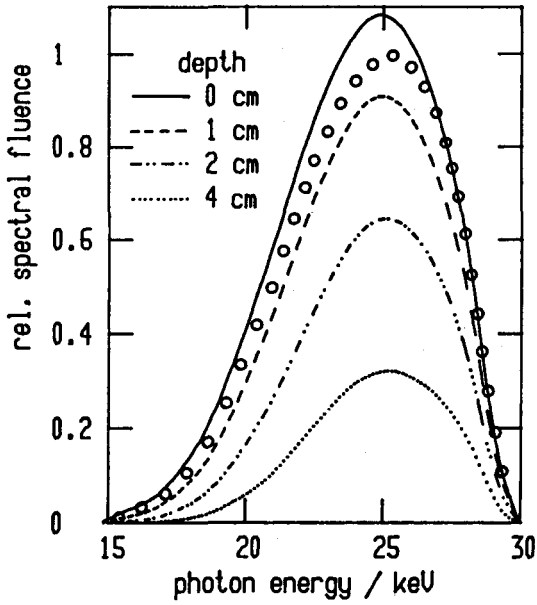

Fig. 1 X-ray spectra at various depths in the ICRU-sphere for a heavily-filtered X-ray beam with 30 kV tube voltage. Points indicate primary spectrum.

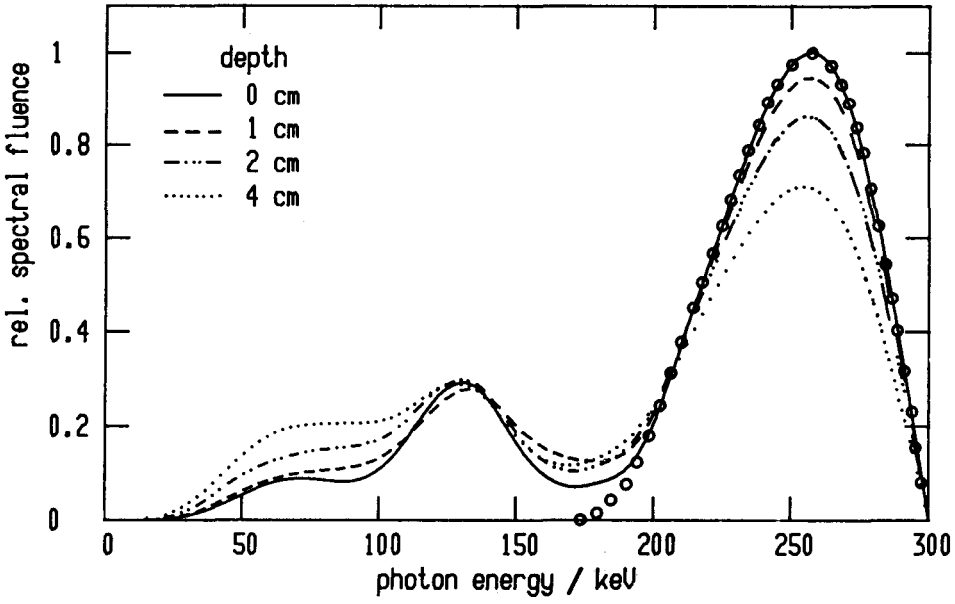

Fig. 2 X-ray spectra at various depths in the ICRU sphere for a heavily-filtered X-ray beam with 300 kV tube voltage. Points indicate primary spectrum.

where $\varphi_E^i(E)$ is the spectrum of the X-ray beam (i) with the maximum energy $E_{max,i}$. This integral equation can be solved for m(E) by iteration. Applying this method also to the values determined from calibration free in air, one obtains a calibration function k(E) corrected by the primary spectra.

The dose equivalent is now given by

$$H(E_o) = Q\int_0^{E_o} m(E) \cdot k(E) \cdot \mu_{a\to t}(E) \cdot k_D(E) \cdot \tilde{\varphi}_E(E)\, dE$$

The quality factor Q is unity for photon radiation, $\mu_{a\to t}$ is defined as

$$\mu_{a\to t} = \frac{(\mu_{en}/\varrho)_{tissue}}{(\mu_{en}/\varrho)_{air}}$$

and $k_D(E)$ is a correction factor for the displacement effect. $\tilde{\varphi}_E(E)$ represents the spectrum in the phantom caused by monoenergetic primary radiation with the energy E_o.

The correction $k_D(E)$ which depends on photon energy, depth in the phantom and the radius of the ionisation chamber was determined theoretically by Monte Carlo calculations and a passage to the limit R→0 for the chamber radius R.

The values of the displacement correction vary between 0.87 at 30 keV and 0.97 at 1250 keV for 1 cm depth and a chamber radius of R = 6 mm.

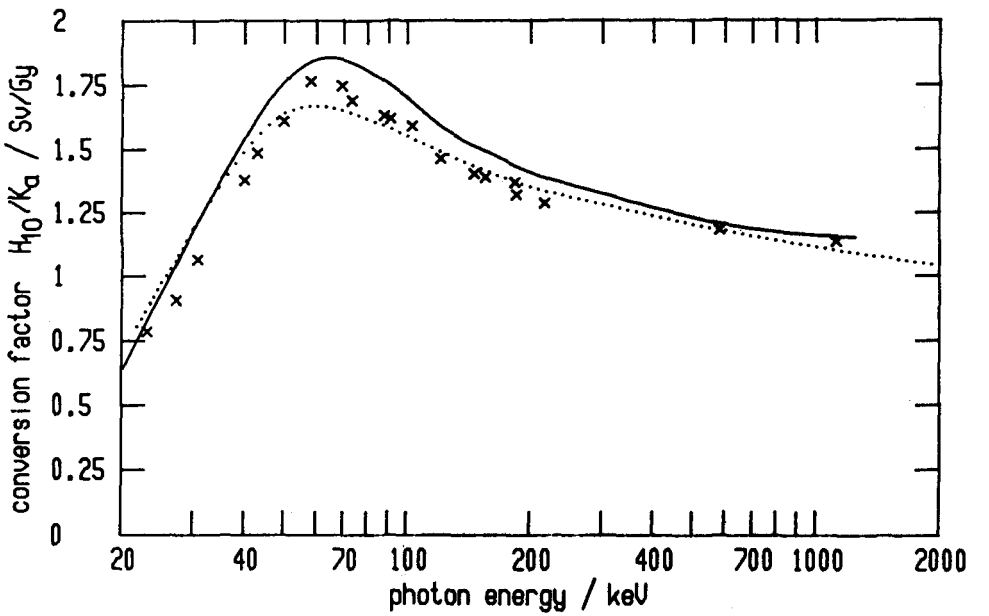

Fig. 3 Energy dependence of the conversion factor for the dose equivalent H_{10} at a 10 mm depth in the ICRU sphere.
solid line: Monte Carlo calculation [3]
dotted line: IEC Recommendation [5]
points: own measurements

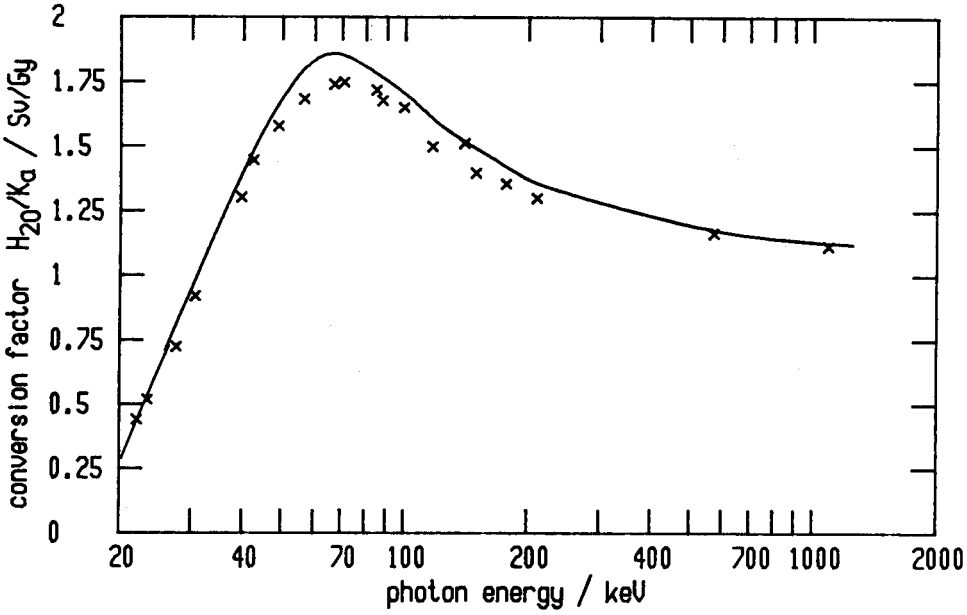

Fig. 4 Energy dependence of the conversion factor for the dose equivalent H_{20} at a 20 mm depth in the ICRU sphere
solid line: Monte Carlo calculations
points: own measurements

Dividing the measured and corrected dose equivalent at a 10 mm depth by the air kerma at the point of measurement yields the conversion factor H_{10}/K_a in the units Sv/Gy.

The results of these investigations are shown in fig. 3. The experimental values are in good agreement with theoretical Monte Carlo calculations [4], but on the average they are by about 6 % smaller. The curve specified by the IEC [5] shows a smaller maximum (about 5 %) and is flatter in shape. The conversion factor for the dose equivalent at a 20 mm depth shown in fig. 4 also confirms the Monte Carlo calculations with even smaller deviations. Altogether, the agreement between the experimental results and the theoretical data is very satisfactory.

1 ICRU Report 25, Conceptual Basis for the Determination of Dose Equivalent ICRU Washington, D.C. (1976)

2 W. W. Seelentag
Phys. Med. Biol. 24 (1979), 767

3 B. Großwendt
Medizinische Physik 1979 (Hrsg. H. Reich) 85-89, Tagung der Deutschen Gesellschaft für Medizinische Physik, Braunschweig (1979); Hüthig Verlag, Heidelberg

4 K. Hohlfeld, B. Großwendt
Radiation Protection Dosimetry 1 (1981), S. 277

5 IEC/45B (Central Office)43, Febr. 1982: Beta, X and Gamma Radiation Dose Equivalent and Dose Equivalent Rate Meters for Use in Radiation Protection

SEA LEVEL COSMIC RAY IONIZATION INTENSITY AT MIDDLE GEOMAGNETIC LATITUDE (25°N) IN JAPAN

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1. INTRODUCTION

The environmental radiation field consists of very complex compositions having relatively low intensities. In addition, they have the characteristics such as the energy and angular distributions, and the time variation, etc. On the other hand, an instrument for measuring the environmental radiation has also the characteristics such as the sensitivity, the energy and directional dependence, and the self-contamination, etc. It seems that the difficulty of the in situ measurements of the environmental radiation and the inconsistency among the obtained values can be attributed to the correlation of both complex characteristics. The situation is same with the case of the measurements of the cosmic-ray ionization intensities at the earth's surface that have been carried out over last 50 years.

On the other hand, a very few experimental determinations have been carried out in Japan.

The present work represents a measurement of the cosmic-ray ionization intensities at sea level concurrently using an air-filled, air-like walled ionization chamber, a pressurized argon-filled, stainless steel walled one, and a NaI(Tl) scintillation spectrometer, performed in September 1980, in the course of the precise measurement program of the environmental radiations in Japan.

In this paper, the term "roentgen" is used as the unit of free air ionization resulting from irradiation by charged particles and photons of high energy, though not strictly in accordance with its definition.

2. METHOD OF MEASUREMENT

Two ionization chambers were used to measure the cosmic-ray ionization, the brief specifications of which are listed in Table 1.

Table 1. Specifications of the ionization chambers

Chamber	Nesco	Aloka
Shape	Cylindrical	Spherical
Wall material	Phenolic resin	Stainless steel
Wall thickness*	5 mm (0.7g/cm^2)	3.2 mm (2.6g/cm^2)**
Active volume	21.3 liter	14.0 liter
Filling gas	Air	Argon
Pressure	1.0 atm.	8.0 atm.

* Perpendicular wall thickness
** Including the iron cover 1.6 mm (1.3g/cm^2) in thickness

These chambers which are of air-tight type have been developed to

measure the low level environmental radiation dose in the vicinity of nuclear facilities. The Nesco chamber is a large volume, normal pressure air-filled ionization chamber with phenolic resin air-like walls. The characteristics of this chamber has been discussed in detail by Kobayashi et al (1). The Aloka chamber is similar to those discussed in detail by DeCampo et al (2). This chamber is constructed from two cold rolled type 304 stainless steel hemispheres of 1.6 mm in thickness, welded together, and filled with argon gas under 8.0 atmospheres at standard temperature and pressure. The inner diameter of this chamber is 30 cm and its active volume is 14.0 liter. The collecting electrode is a spherical aluminum of 6 cm in diameter,and is completely isolated from outer shell with the aid of an alumina ceramic insulator, and guard electrode is connected to the ground to prevent the leakage current. The chamber is contained in iron cover of 1.6 mm in thickness, and its total effective wall thickness is 3.1g/cm^2. The electrometer to measure the ionization current consists of a MOSFET, directly connected to the collecting electrode.

A portable scintillation spectrometer equipped with a spherical NaI(Tl) scintillator of 3-inch in diameter was used to measure the gamma-ray from environmental radionuclides. The spectrometer consists of the spherical NaI(Tl) scintillation detector, an interface, and a commercial stereo tape recorder (3).

It is necessary to evaluate the responses of the ionization chamber to environmental gamma-ray and cosmic-ray secondaries to determine the free air ionization intensity. As the Nesco chamber is an air-filled ionization chamber which has an air-like thin walls of 0.7g/cm^2 in thickness, the response to cosmic-ray secondaries is considered to be very closely equal to that to environmental gamma-ray. Therefore, in a measurement using the Nesco chamber, the ionization intensity due to cosmic-ray secondaries is determined by subtracting that due to environmental gamma-ray from the total ionization intensity. On the other hand, as the Aloka chamber is a pressurized argon-filled ionization chamber which has a stainless steel walls of 3.1g/cm^2 in effective thickness, the treatments of the wall effects become to be very important. In this paper, the response to environmental gamma-ray was determined using a Ra-226 standard source which had been calibrated by ETL, and the response to cosmic-ray secondaries at sea level was evaluated using the method of HASL (2) from data on a transition effect at wall due to electromagnetic shower of electron and photon components obtained by Beck (4) and the data on fluxes of muon and electron at sea level at 50°N geomagnetic latitude summarized by Lowder and O'Brien (5). The value of the ratio of the cosmic-ray response to the environmental gamma-ray one was evaluated to be 1.07 at sea level.

3. MEASUREMENT

Measurements were carried out using fishing boats and rubber boats over the sea of the bay of Wakasa at middle geomagnetic latitude (25°1'N, 202°5'E) in Japan. The observation point was located at a distance of about 2 km from shore to avoid the terrestrial gamma-ray. The Nesco chamber and the portable NaI(Tl) scintillation spectrometer were on two rubber boats, respectively, and measured with floating away about 20 m from the fishing boats to avoid the gamma-ray from boats and human bodies. The Aloka chamber was held at about 1 m over the deck of the fishing boat. The total ionization intensity was

measured continuously by these ionization chambers, and the pulse height distribution due to environmental gamma-ray which was observed by the NaI(Tl) spectrometer was accumulated and recorded every 45 minutes in a cassete tape recorder. The concentration of Rn-222 and its associated daughters in the atmosphere was also measured using a filter pack method. In addition, the concentrations of U-series, Th-series, and K-40 in the sea water were also measured. The atmospheric condition all through the measurements was fair weather with the atmospheric pressure of (760 ± 2) mmHg and the temperature of (22 ± 2) °C.

4. RESULTS AND DISCUSSION

In order to determine the cosmic-ray ionization intensity in free air using an ionization chamber, it is necessary to correct the cosmic-ray response for the contribution of cosmic-ray after subtracting the components of environmental gamma-ray, alpha particles due to self-contamination in the wall material, and leakage current, from the total ionization intensity. Here, the leakage currents of these chamber were considered to be negligibly small compared to other components. The self-contamination of the Nesco chamber was corrected in the readings of the recorder chart, and that of the Aloka chamber was determined to be 0.026 μR/h using a method of charge accumulation in the condition of lowered pressure of filling gas. The environmental gamma-ray was evaluated using the gamma-ray energy spectrum which was transformed after subtracting the contributions of cosmic-ray and K-40 within the detector from the measured pulse height distribution by the iterative method using the response matrix of the NaI(Tl) detector (6). The contributions of cosmic-ray secondaries and K-40 within the detector were determined by an extrapolation from the measurements in the upper atmosphere to it at ground level (7), and by the measurements in the iron room with walls of 20 cm in thickness, respectively. The exposure rates due to environmental gamma-ray which were determined in this manner were in 0.21 ∿ 0.32 μR/h.

On the other hand, the gamma-ray exposure rates over the sea were calculated from the concentrations of environmental radionuclides which were determined experimentally, to compare those measured by the NaI(Tl) scintillation spectrometer. We took U-series, Th-series, and K-40 in the sea water, and Rn-222 and its associated daughters in the atmosphere, as the sources of environmental gamma-ray. The calculations were performed with a dose build up factor under the assumptions of uniform distribution of radioactivities in the objective environment, and radioactive equilibrium between parent and its associated daughters. Results of the calculations are summarized in Table 2. The calculated values were somewhat larger than those measured by the NaI(Tl) scintillation spectrometer when the wind velocity was under few meters per second and the concentration of Rn-222 and its associated daughters was relatively high, because of the high concentration in the lower atmosphere near the sea level. But both values were in very good agreement with each other when the wind velocity was several meters per second. In this paper, the values measured by the NaI(Tl) scintillation spectrometer were employed as the environmental gamma-ray exposure rates.

Table 2. Ionization intensities due to environmental radionuclides

	Nuclide	Concentration(pCi/cm^3)	Ionization(μR/h)
Sea water	K-40	3.23×10^{-1}	0.057
	U-series	8.7×10^{-4}	0.0017
	Th-series	6.2×10^{-4}	0.0015
Atmosphere	Rn-222	$(8.6 \sim 20.0) \times 10^{-5}$	0.15 ∿ 0.35

The average contributions of cosmic-ray secondaries at sea level which were obtained after subtracting the environmental gamma-rays from total ionization intensities were (3.34 ± 0.02) μR/h for the Nesco chamber, and (3.52 ± 0.05) μR/h for the Aloka chamber, respectively. For the Nesco chamber, this value is considered to be equal to the cosmic-ray secondaries ionization intensity. The value of the Aloka chamber is transformed to (3.29 ± 0.05) μR/h in considering the correction factor 1.07 of the cosmic-ray response. As we used the data at 50°N for a lack of the data on the fluxes of cosmic-ray secondaries at sea level at middle or low geomagnetic latitude, this correction factor may be altered a little in future. But these values are considered to be in very good agreement with each other within the errors in spite of the uses of the different types of the ionization chambers. These values are not corrected the variation due to the pressure and temperature dependence of the cosmic-ray secondaries.

References

(1). H. Kobayashi et al., Natural Radiation Environment III, Symposium Series DOE 51, vol. 2, 1004 (1980).
(2). J. A. DeCampo et al., HASL-260 (1972).
(3). M. Okano et al., Natural Radiation Environment III, Symposium Series DOE 51, vol. 2, 896 (1980).
(4). H. L. Beck, Nucl. Instr. and Methods, 91, 525 (1971).
(5). W. M. Lowder and K. O'Brien, HASL-254 (1972).
(6). S. Minato and M. Kawano, J. Nucl. Sci. Technol., 7, 401 (1970).
(7). I. Urabe and K. Katsurayama, Radiation Protection Dosimetry, 1, 183 (1981).

A METHOD OF DETERMINING COSMIC-RAY DOSE RATE BY A 3"ø SPHERICAL NaI(Tl) SCINTILLATION COUNTER IN THE INDOOR ENVIRONMENT

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Although indoor cosmic-rays have been considered to be significant in estimating population dose, very little has been reported on the method of measuring the dose rate separately from indoor gamma-rays. In this contribution, a 3"ø spherical NaI(Tl) scintillator is used with techniques of integral count rate for the pulse-height regions greater than 3 and 10 MeV, where there are no gamma-ray contributions from natural environmental isotopes and nuclear reactors, to provide cosmic-ray dose rate with high accuracy.

Figure 1 gives a distribution of count rates for cosmic-rays passing through a lead block, which was observed in a normal concrete building. Figure 2 gives the result of observations inside and outside a 6 storeyed building.

The outline of the method of converting the count rate into the dose rate is roughly as follows. The variation of flux spectrum and dose rate with concrete depth for cosmic-ray muons, photons and electrons is calculated from a numerical approach to the Boltzmann transport equation(1). The input data on cosmic-rays at sea level were taken from the work of O'Brien(2,3). The count rate of a 3"ø spherical NaI(Tl) scintillator is obtained through Monte Carlo calculations for the flux spectrum at each depth in concrete evaluated above(3). The calculated result is given in Fig.3, in which $\dot{X}_0$ and C_0 are, respectively, the dose rate and the integrated count rate measured in the outdoor environment at sea level where there is no influence of buildings. At Nagoya, the values of $\dot{X}_0$ and C_0 amount to about 3.33 μR/h and 1.46 cps, respectively, in 1980. Using Fig.3, we can estimate indoor cosmic-ray dose rate for concrete buildings by measuring the integrated count rate for the region greater than a discrimination level(4).

In order to assess the validity of the method, the comparison of calculated response with the experimental measurement made on the roof of the building is given in Fig.4. The result shows a good agreement between them.

REFERENCES

1. Minato,S.,(1982) Rep. Gov. Ind. Res. Inst. Nagoya, 31, 77.
2. O'Brien,K.,(1979) Trans. Amer. Nucl. Soc., 33, 671.
3. Minato,S., Takamori,T., and Ikebe,Y., (1983) Rep. Gov. Ind. Res. Inst. Nagoya, 32, 14.
4. Minato,S., and Minakuchi,S.,(1983) Health Phys., in printing.

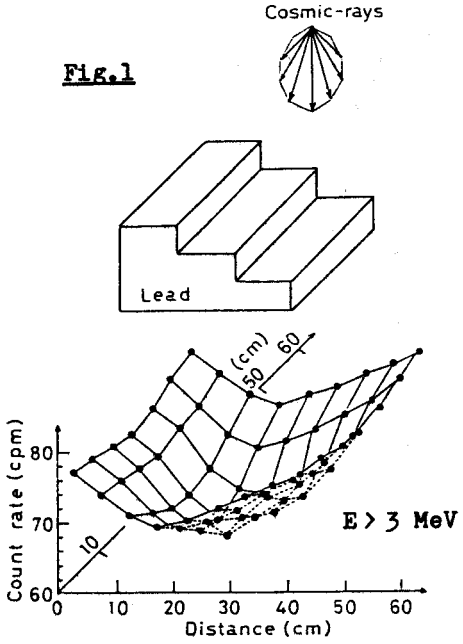

Fig.1

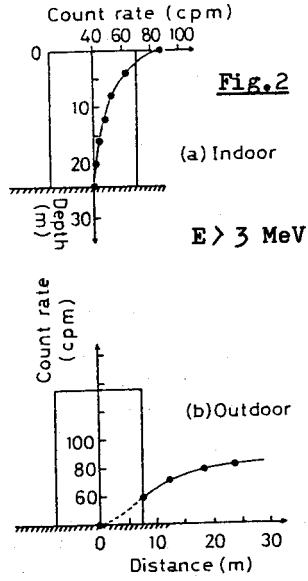

Fig.2

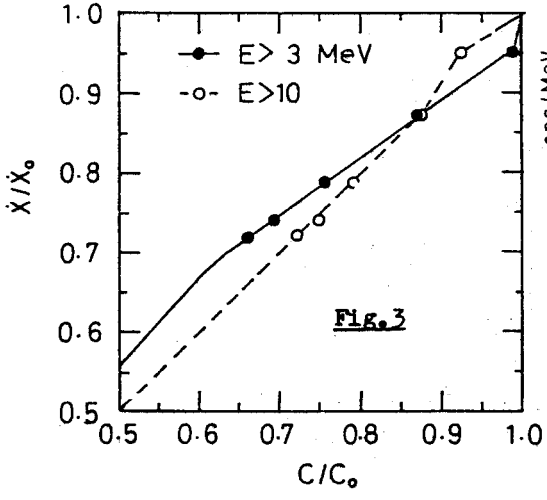

Fig.3

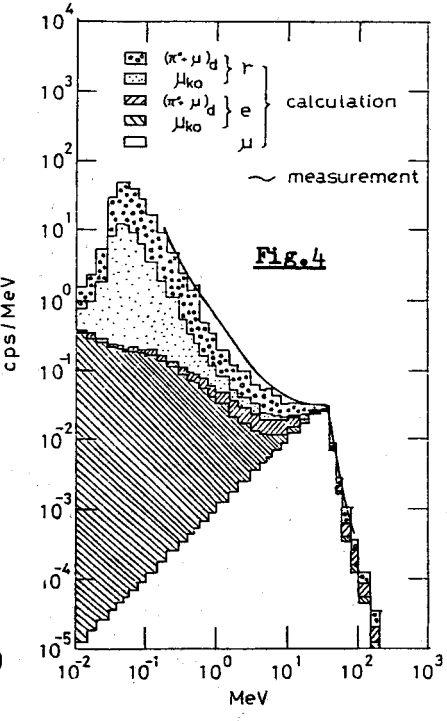

Fig.4

PERSONNEL DOSIMETRY USING PELLETS OF $CaSO_4$:Dy BOUND IN ALKALY HALIDES

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Thermoluminescent dosemeters (TLD) are widely used for routine dose measurements in personnel monitoring due to their advantages over film dosemeters (1,2). Among TL materials used in this field the most accepted has been LiF:Mg,Ti which is obtained commercially. However, Teflon embedded $CaSO_4$:Dy is being increasingly utilized in many countries for this purpose (3-6).

In our laboratory, to replace LiF commercial TLD's in personnel monitoring pellets of $CaSO_4$:Dy bound in alkaly halides were developed.

Before deciding on the suitability of introducing $CaSO_4$:Dy bound in alkaly halides pellets in routine personnel dose measurements, dosimetric characteristics of $CaSO_4$:Dy+KBr, $CaSO_4$:Dy+KCl and $CaSO_4$:Dy+NaCl pellets were determined and tests of comparative effectiveness of $CaSO_4$:Dy+KBr pellets in the field were carried out for a period of one year.

Our test design was done encasing $CaSO_4$:Dy+KBr pellets together with TLD-100* ribbons in black plastic cassettes provided with appropriate compensating filters. The cassettes were then distributed to selected groups of occupationally exposed personnel in the Nuclear Center of Mexico.

Results obtained during this test period showed good agreement between the dosimetric effectiveness of $CaSO_4$:Dy+KBr pellets and the Harshaw TLD-100 ribbons.

EXPERIMENTAL

$CaSO_4$:Dy phosphor powder was prepared using the acid evaporation method proposed by Yamashita (7) and improved by us (8,9) doping $CaSO_4.2H_2O$ with 0.1 mol% of Dy_2O_3 at 285°C and drying in an oven at 600°C during one hour in a platinum crucible. The dried material was powdered and sieved and crystalline powder of grain size between 80 and 200 μm was selected to be used as TL phosphor.

To make pellets of $CaSO_4$:Dy+KBr, $CaSO_4$:Dy+KCl and $CaSO_4$:Dy+NaCl the phosphor powder was mixed with the respective alkaly halide which was used as binder in a ratio of 1:3 applying a pressure of 980 Pa for each 100 ± 3 mg of mixture. This technique yielded pellets of 0,95 ± 0.05 mm thickness.

Before determining their dosimetric characteristics the pellets were treated by standard thermal annealing procedure consisting in heating them at 400°C during one hour.

To select homogeneous batches all the pellets were irradiated at a dose of 5 mGy of ^{60}Co gammas and those pellets which TL response deviated ± 5% from the average were considered within an homogeneous batch.

The dosimetric characteristics of the pellets were determined irradiating them with ^{60}Co gammas from an encapsulated source of 18 GBq. All irradiations were efectuated under electronic equilibrium conditions.

Readings were made in a Harshaw 2000 A/B TL system connected to an x-y recorder to obtain glow curves. TL signal was integrated between room temperature

* Trade name of Harshaw Chemical Co. Solon, Ohio USA.

(∿20°C) and 300°C during 60 seconds using a linear heating rate of 6°C/s.

TL response in function of dose was determined irradiating the pellets with ^{60}Co gammas at different known doses and plotting TL intensity against dose.

Fading of TL response was investigated irradiating 20 pellets of each kind at 5 mGy of ^{60}Co gammas and storing them, protected from the light, along with 5 other unirradiated pellets of each kind. Readings were made immediately after irradiation and at different post-irradiation times ranging from 1 to 30 days.

Reusability was determined irradiating 10 pellets of each kind at 5 mGy and recording the glow curves and the TL response for each pellet. After readings, pellets were annealed and again submitted to the same absorbed dose under the same conditions and their TL response and glow curves were recorded. This process was repeated 10 times and the average TL response for each pellet of the three groups was determined.

For personnel monitoring, in the field, we decided to use pellets of $CaSO_4$:Dy +KBr. For this purpose we selected, from a large number of pellets, a batch of 500 pellets to be distributed to 30 occupationally exposed individuals working at the Department of Radioactive Materials in the Nuclear Center of Mexico.

Each dosemeter distributed consisted of two $CaSO_4$:Dy+KBr pellets and two TLD-100 ribbons placed into black plastic cassettes consisting of two plates of dimensions 4x5x0.5 cm hinged toghether at the center by means of a screw, sealed with an O'ring rubber seal and provided with a clip for wearing. As shown in figure 1 both two plates have four circular depressions: two of 1.0 mm and two of 2.0 mm depth of diameter 6 mm. At the locations 3 and 4 and 3' and 4' lead filters of 0.8 mm thick are placed with silicon glue. When the cassette is closed pairs of filters face each with other.

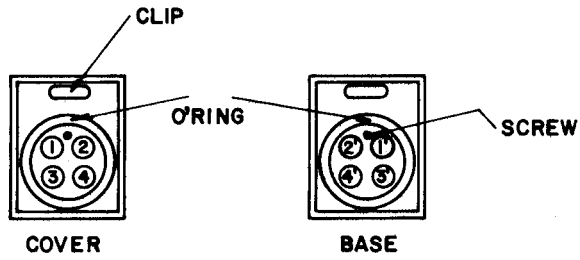

Figure 1.- Black plastic cassettes in which $CaSO_4$:Dy+KBr pellets and TLD-100 ribbons were placed for personnel monitoring.

Two CaSO4:Dy+KBr pellets were placed in locations 1 and 3 and two TLD-100 ribbons in locations 2 and 4 respectively.

Two personnel dosemeters, constituted by two pellets and two ribbons, were asigned to each individual. Three of such dosemeters were supplied to the users to be kept as control dosimeters.

During the processes of thermal annealing, placing in plastic cassettes and

read out, the pellets and the ribbons were handled in safe light conditions.

After one month of use in field the dosemeters were sent back to the laboratory and substituted by other set assigned.

Immediately arriving the dosemeters to laboratory readings were taken. To determine the unknown doses a calibration curve was constructed using 100 pellets having TL characteristics very similar to those of the pellets used in the field.

Pellets were reused by the same individual before thermal annealing. This annealing consisted in reading each pellet twice.

During the 12 months of this study a record of date of thermal annealing, date of dispatch, period of use, name of users, date of receipt and date of TL reading were maintained.

RESULTS AND CONCLUSIONS

Pellets of $CaSO_4$:Dy+KBr, $CaSO_4$:Dy+KCl and $CaSO_4$:Dy+NaCl presented similar glow curves as shown in figure 2. For the three kinds of pellets, TL response in function of dose fitted to a straight line in log-log scale from 1 mGy to 1 Gy (see figure 3). Also they exhibited a fading rate of about 5% per year. Both characteristics are desirable for personnel monitoring.

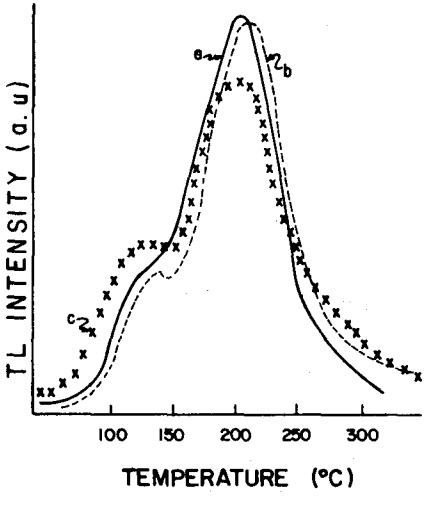

Fig. 2. TL glow curves of $CaSO_4$:Dy bound in alkaly halides obtained irradiating with 5 mGy of ^{60}Co gammas a) $CaSO_4$:Dy+KBr, b) $CaSO_4$:Dy+KCl, c) $CaSO_4$:Dy+NaCl

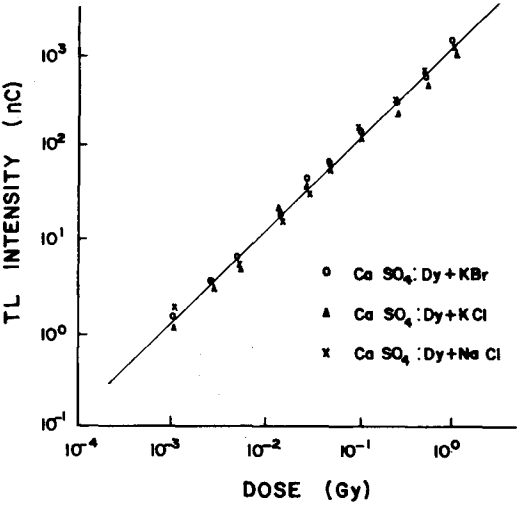

Fig. 3. TL response in function of dose for $CaSO_4$:Dy bound in alkaly halides.

Taking the average TL response of 10 pellets, submitted to repeated irradiation and reading cycles by 10 times, the standard error found was 1.2% for $CaSO_4$:Dy+KBr, 1.5% for $CaSO_4$:Dy+KCl and 1.4% for $CaSO_4$:Dy+NaCl. Therefore, all pellets can be used repeatedily without any recalibration.

The standard error we obtained for $CaSO_4$:Dy+KBr pellets (the smallest of the three) decided us to use them to test in the field.

Results of using $CaSO_4$:Dy+KBr pellets in the field during 12 months were quite satisfactory. Absorbed doses measured with these pellets deviated from those measured with TLD-100 placed in the same cassette by ± 3% which we considered as a good grade of agreement.

Based on these results we concluded that pellets of $CaSO_4$:Dy+KBr provided with appropriate filtration to make their TL response energy independent and placed in plastic cassettes furnish a practical, sensitive and reliable dosemeter for personnel monitoring.

Therefore, this dosemeter can be produced locally on mass scale and commercialized at low cost.

Acknowledgements.

The authors acknowledge the cooperation of Luis E. Guzmán R. in the experiments and express their sincere appreciation to the staff of the Radiactive Materials Department who participated in this study wearing the dosemeters at the same time of their routine personnel dosemeters.

REFERENCES

1.- Becker, K. Status and Trends in Personnel Monitoring A Worldwide Survey Kerntechnik 18 (1/9) 345-351 (1976).
2.- Madhuanath, V., Patel, P.H., Dhond, R.V., Shenoy, K.S. and Kadman, A.B. Film vs TLD: A Choice for a Countrywide Personnel Monitoring Programme Health Phys. 30 (1) 119-124 (1976).
3.- Pradhan A.S., Dere A. and Popli K. Annealing and Repeated Readout of TLD Cards Based on $CaSO_4$:Dy Teflon Discs. Int. J. Appl. Radiat. Isot. 30 317 (1979).
4.- Lakshmanan A.R., Bhuwan Chandra, Pradhan A.S., Kher, R.K. and Bhatt R.C. The Development of Thin $CaSO_4$:Dy Teflon Dosimeters for Beta Dosimetry in Personnel Monitoring Int. J. Appl. Radiat. Isot. 31 107 (1980).
5.- Mattheus R.J. and Stoebe T.G. Precision TLD Using $CaSO_4$:Dy Tefon Dosimeters Nucl. Instrum. Methods 175 171 (1980).
6.- Boas J.F., Murray A. and Young J.G. Testing of $CaSO_4$:Dy in Teflon Discs as a TLD Material for Personnel Monitoring of Uranium Mine and Mill Workers Aust. Radiat. Lab. Melbourne ARL/TR-031 (1981).
7.- Yamashita T., Nada N., Onishi H. and Kitamura S. $CaSO_4$ Activated by Thulium or Dysprosium for TLD Health Phys. 21 295 (1971).
8.- Azorín J., Salvi R. and Moreno A. Improvement in Preparation of $CaSO_4$:Dy as TL Dosimeter Nucl. Instrum. Methods 175 81 (1980).
9.- Azorín J., González G., Gutiérrez A. and Salvi R. Preparation and Dosimetric Properties of Highly Sensitive $CaSO_4$:Dy TL Dosimeter Health Phys. 1983 (in press).

DEVELOPMENT OF PERSONNEL DOSIMETER USING $Li_2B_4O_7$(Cu) ELEMENTS AND AUTOMATIC TLD READER

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1. INTRODUCTION

New personnel dosimeter capable of evaluating gamma, beta and neutron doses was developed using new TLD elements ($Li_2B_4O_7$:Cu) at PNC Tokai Works. The developed dosimeter is measured by the automatic TLD reader connected with the computer system. The thermoluminescence of each element is measured by a photon counting method in a very short time.

Gamma dose is evaluated as dose equivalent index at 1000 mg/cm^2 depth and beta dose is evaluated by energy information obtained from ratio of two β-ray luminescences. The principle of neutron dose evaluation is based on the albedo method and three neutron doses are separately evaluated for thermal, epithermal and fast neutrons.

At present, more than 3000 personnel are routinely monitored by this new TLD badge at PNC Tokai nuclear fuel facilities.

2. CONSTRUCTION OF THE TLD BADGE

Figure 1 shows the overall construction of new personnel dosimeter (TLD badge). This badge consists of two dosimeters (one for gamma and beta radiation and the other for neutron radiation) and badge case. The case is designed to hold the two dosimeters in two vertical rows.

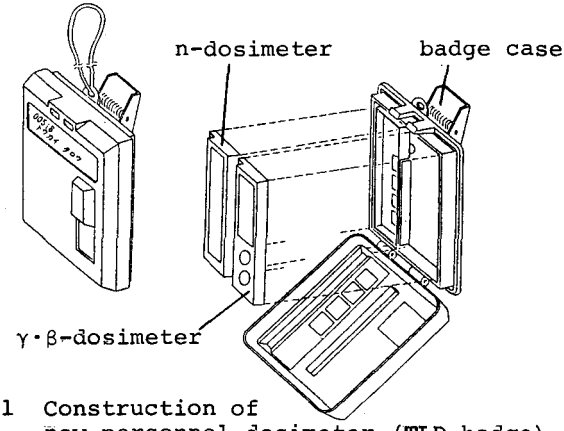

Fig. 1 Construction of new personnel dosimeter (TLD badge)

Each dosimeter contains four TLD elements and 40 bit punched indentification codes for optical reading.

The phosphors of TLD elements are $Li_2B_4O_7$(Cu) with an effective atomic number of 7.26 which is the bio-equivalent of soft tissue and $CaSO_4$(Tm) with an effective atomic number of 15.3 which is not

bio-equivalent. The construction of TLD elements is shown in Fig.2.

One $Li_2B_4O_7$ and one $CaSO_4$ for gamma ray are shielded to provide interpretation of 1000 mg/cm^2 tissue depth dose and two $Li_2B_4O_7$ elements are used for beta absorbed dose interpretation. Three $^6Li_2{}^{10}B_4O_7$ and one $^7Li_2{}^{11}B_4O_7$ elements are used for thermal, epithermal and fast neutrons by means of albedo method. Figure 3 shows the construction of TLD elements and filter system in neutron dosimeter.

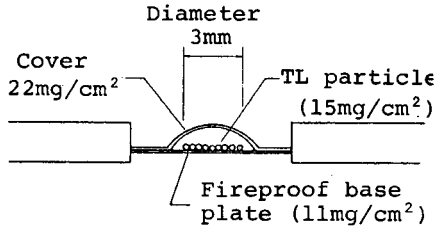

Fig. 2 Construction of TLD element

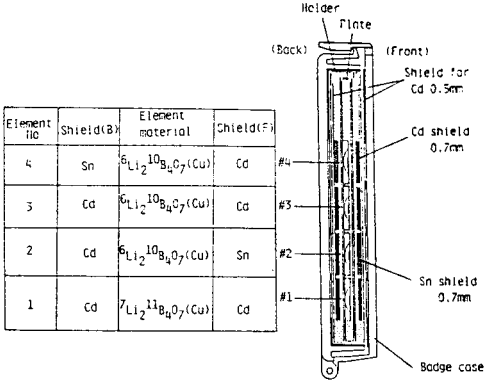

Fig. 3 TLD badge for neutron dosimetry

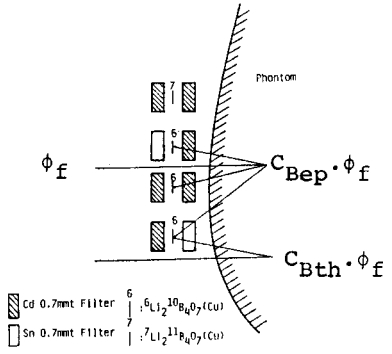

Fig. 4 Conceptual figure of TL-element and incident fast neutron

The specification for 8 TLD elements installed in the two dosimeters are presented in Table 1.

Table 1. Constitution of TLD element and filter

Dosimeter	Object	Element	TLD element and filter
γ·β dosimeter	γ - ray	T_4	$^7Li_2{}^{11}B_4O_7$(Cu) + 1000 mg/cm² plastic filter
		T_3	$CaSO_4$(Tm) + 1000 mg/cm² plastic filter
	β - ray	T_1	$^7Li_2{}^{11}B_4O_7$(Cu) + 17 mg/cm² plastic filter
		T_2	$^7Li_2{}^{11}B_4O_7$(Cu) + 69 mg/cm² plastic filter
n dosimeter	neutron	T_1	(front) Cd/$^7Li_2{}^{11}B_4O_7$(Cu)/Cd (rear)
		T_2	Sn/$^6Li_2{}^{10}B_4O_7$(Cu)/Cd
		T_3	Cd/$^6Li_2{}^{10}B_4O_7$(Cu)/Cd
		T_4	Cd/$^6Li_2{}^{10}B_4O_7$(Cu)/Sn

3. BASIC EQUATIONS FOR DOSE EVALUATION

The relationships between the luminescences (Tn) of TLD elements in the mixed beta gamma field and the gamma dose equivalent (Dγ) and the beta dose equivalent (Dβ) are expressed by the following equations:

$$T_1 = b_1(E_\beta)D_\beta + g_1(E\gamma)D\gamma + C_{BG}$$
$$T_2 = b_2(E_\beta)D_\beta + g_2(E\gamma)D\gamma + C_{BG}$$
$$T_4 = g_4(E\gamma)D\gamma + C_{BG}$$

bn,gn: sensitivity of TLD-n to beta/ gamma radiation.
From equations, we derive the following equations:

$$Dr = Q_4/g_4(E\gamma)$$

$$D_\beta = \frac{Q_1 - g_1(E\gamma)D\gamma}{b_1(E_\beta)} = \frac{Q_1 - \frac{g_1(E\gamma)}{g_4(E\gamma)}Q_4}{b_1(E_\beta)}$$

$$= \frac{Q_2 - g_2(E\gamma)D\gamma}{b_2(E_\beta)} = \frac{Q_2 - \frac{g_2(E\gamma)}{g_4(E\gamma)}Q_4}{b_2(E_\beta)}$$

where $Q_n = T_n - C_{BG}$

Similar relationships are expressed in the mixed neutron-gamma field. (Fig. 4)

$$T_1 = G_\gamma\phi_\gamma + G_{\gamma\cdot c}\phi_{th}$$
$$T_2 = G_{th}\phi_{th} + (1+b_{Bth})G_{ep}\phi_{ep} + C_{Bth}G_f\phi_f + G_\gamma\phi_\gamma + G_{\gamma\cdot c}\phi_{th}$$
$$T_3 = (1+b_{Bep})G_{ep}\phi_{ep} + C_{Bth}G_f\phi_f + G_\gamma\phi_\gamma + G_{\gamma\cdot c}\phi_{th}$$
$$T_4 = a_{Bth}G_{th}\phi_{th} + (1+b_{Bth}+b_{Bep})G_{ep}\phi_{ep} + (C_{Bth}+C_{Bep})G_f\phi_f + G_\gamma\phi_\gamma + G_{\gamma\cdot c}\phi_{th}$$

From equations, we derive the neutron doses.

$$D_{th} = \Gamma_{th}G_{th}\phi_{th} = \Gamma_{th}(T_2 - T_3)$$

$$D_{ep} = \Gamma_{ep}\left[\left\{\frac{1}{1+b_{Bep}} + \frac{b_{Bth}\cdot C_{Bep}}{1+b_{Bep}}\cdot\frac{1}{C_{Bth}+b_{Bep}\cdot C_{Bth}-b_{Bth}C_{Bep}}\right\}\right.$$
$$\chi\ (T_3 - T_1) - \frac{C_{Bep}}{C_{Bth}+b_{Bep}\cdot C_{Bth}-b_{Bth}\cdot C_{Bep}}$$
$$\left.\chi\left\{(T_4 - T_3) - a_{Bth}(T_2 - T_3)\right\}\right]$$

$$D_f = \frac{\Gamma_f}{K_f}\left\{(T_4 - T_3) - a_{Bth}(T_2 - T_3) - \frac{b_{Bth}}{1+b_{Bep}}(T_3 - T_1)\right\}$$

where

Dn: thermal, epithermal, fast neutron dose equivalent

Gn: conversion coefficient from thermal, epithermal, fast neutron fluence to luminescence (mR eq ^{60}Co - γ/n/cm^2)

a_{Bi}, b_{Bi}, c_{Bi}: fraction of backscatterd thermal (i=th), epithermal (i=ep), fast (i=f) neutron to incident thermal (a), epithermal (b), fast (c) neutron

K_f: $(c_{Bth} + b_{Bep}\cdot c_{Bth} - b_{Bth}\cdot c_{Bep}) / (1 + b_{Bep})$

Fn: conversion coefficient relating the luminescence to each neutron dose equivalent. (mrem/mReq^{60}Co-γ)

4. DOSIMETRIC CHARACTERISTICS

Figure 5 shows the gamma energy response for $Li_2B_4O_7$ (Cu) element with resin filters having thickness of 1000 mg/cm^2. The distinct difference were found between the data obtained with phantom installed and those obtained in free air.

The minimum detectable amount for gamma and thermal neutron dose were 2 mrem and 0.02 mrem respectively.

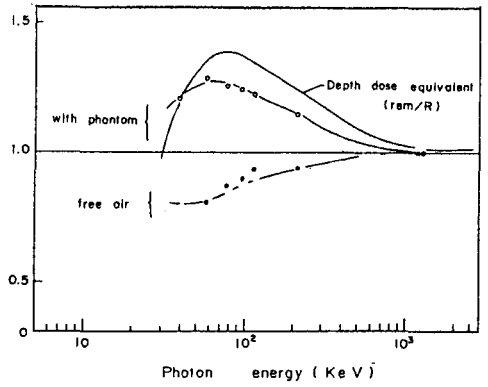

Fig. 5 Energy response of $Li_2B_4O_7$

The capability to evaluate each dose in the mixed beta and gamma field was studied by the standard sources. The irradiation results are shown in Tabel 2. The beta dose equivalent can be evaluated at 100 mrem ± 20 % (20 mrem ± 100 %) in the mixed beta and gamma field.

Table 2. Beta and gamma mixed dose comparison

No.	β-Dose given (mrad)		γ-Dose given (mR)		Total β-Dose given (mrad)	β/γ Ratio	Energy index (EI)	Effective β-ray maximum energy by EI (MeV)	β-Dose evaluated	β-Dose evaluated / β-Dose given
	^{90}Sr-^{90}Y	^{204}Tl	^{60}Co or X-Ray							
1	456	—	^{60}Co	191	456	2.4	0.80	2.0	466	1.02
2	—	525	^{60}Co	191	525	2.7	0.35	0.8	582	1.11
3	482	—	X-ray 60 keV	170	482	2.8	0.84	2.3	541	1.12
4	—	484	X-ray 60 keV	170	484	2.8	0.41	0.8	504	1.04
5	456	567	^{60}Co	192	1,022	5.3	0.60	1.3	887	0.87
6	379	452	^{60}Co	80	831	10.4	0.65	1.4	806	0.97
7	455	141	^{60}Co	80	596	7.5	0.77	1.8	635	1.06
8	10	—	^{60}Co	59	10	0.17	0.64	1.4	22	2.20
9	50	—	^{60}Co	59	50	0.85	0.81	2.0	47	0.93

Similar examination was studied in the mixed neutron and gamma field. The results are shown in Table 3. According to those results, the gamma dose and the fast-neutron dose are evaluated with a mean error of about 20 percent relative to the irradiated doses.

Table 3. Neutron and gamma mixed dose comparison

Irradiated dose (mrem)		evaluated dose (mrem)		Hf/Df	Hr/Dr
Df	Dr	Hf	Hr		
10	20	11 ± 19 %	20 ± 5 %	1.10	1.00
30	58	31 ± 12 %	60 ± 4 %	1.05	1.03
40	59	39 ± 12 %	63 ± 5 %	0.97	1.05
50	64	49 ± 15 %	67 ± 3 %	0.97	1.05

ENERGY DEPENDENCE AND FILTER COMPENSATION OF SELF-FABRICATED TLD-$CaSO_4$:Dy/TEFLON DISCS FOR BETA-GAMMA PERSONNEL DOSIMETER

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Thermoluminescent dosimeter-$CaSO_4$/Teflon discs were prepared by the authors. Series of experiments were applied to investigate the photon energy dependence of these TLD discs. Calculations of Bassi et. al.[1] had shown that the photon energy dependence of high effective atomic number (Z_{eff}) TL materials such as CaF_2, $CaSO_4$:Dy, etc. could be reduced by embedding them into lower Z_{eff} materials such as teflon and silicone rubber. Pradhan et.al.[2] had pointed out that the photon energy dependence is independent of the proportion of TL phosphor embedded. Our results are in good agreement with the statements of Pradhan et.al. The reasons are discussed thoroughly. The effects of various filters on photon energy dependence of TLD discs were also studied and a choice of filter combination for personnel β-γ dosimeter was suggested.

MATERIALS AND EXPERIMENTS

Self-fabricated $CaSO_4$:Dy powder (0.11 mol% Dy) was prepared following the method derived from Yamashita et.al.[3] Two ranges of grain size from 0 to 74μ and 74 to 177μ were divided by sieving them through sieves with equivalent mesh numbers 200 and 80 respectively. A mixture of phosphor and teflon (Du Pont 800J, grain size 330) was pressed under 7000 psi into small discs. After being properly sinterred, TLD discs (0.4mm thick and 9mm diameter) of different weight percentages (15%, 25%, and 30%) of $CaSO_4$:Dy were thus produced[4,5]. The weight of each disc was 70mg and the density was 2.75 g/cm^3 approximately.

Irradiations were carried out by using a ^{60}Co source, a ^{137}Cs source, and an X-ray machine which provided 42, 64, 86, and 110 keV X-rays. A batch of 10 TLD discs kept adjacent to each other and sandwiched between two sheets of 1.5mm plastic was attached to a phantom and exposed to 2.58×10^{-3} C/kg (10R) each time. These TLD discs were immediately read out in TLD reader and the data were averaged to minimize the statistical errors. TLD discs were annealed at 290°C for an hour before reused.

Square plastic and metal plates including aluminum, copper, cadmium, and stannum of 2 cm^2 area and various thickness were used in the experiments for filter compensation. Whenever Cu, Cd, and Sn were used, a sheet of 0.4mm Al was added before them to cut off beta particles from reaching these high Z materials and thereby eliminating the production of bremsstrahlung radiations.

RESULTS AND DISCUSSION

Bassi's calculations[1] based on 1). considering the mixture as a homogeneous compound, 2). assuming the electronic equilibrium,

3). not taking the self-absorption and the effect of grain size into account, and 4). assuming the light yield per rad independent of the LET. They claimed that the energy dependence of TLD discs could be obviously reduced by decreasing the weight percentage of TL phosphor. In this study (curves 1,2 and 3 in Fig.1) we found that the energy dependence of $CaSO_4$:Dy/teflon discs is almost independent of the proportion of $CaSO_4$:Dy.

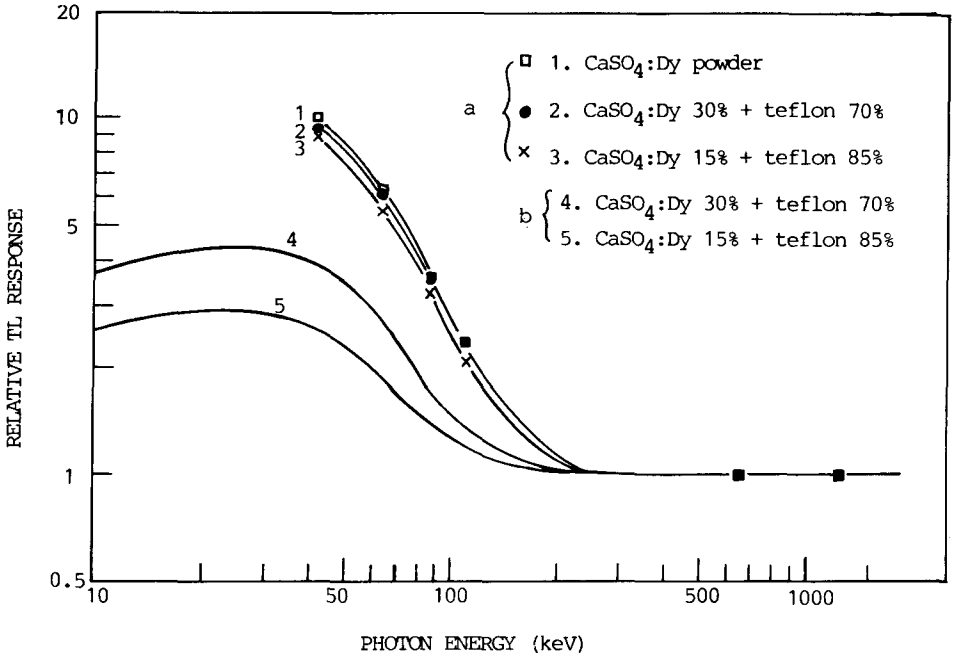

Fig.1. Experimental and calculated photon energy dependence of TLD-$CaSO_4$:Dy/teflon discs.
(a. results of present work. b. calculations of Bassi et al.[2])

The calculations took only the "mass energy absorption coefficients" into account. Therefore the values obtianed could merely represent the "relative mass energy absorbed by TLD discs" but not the "relative TL response of TLD discs".

The energy of secondary electrons (including Auger electrons) originated in both $CaSO_4$:Dy and teflon when exposed to low energy photons (<60 keV) is less than 60 keV. According to the "Katz and Penford Formula"[6],

$$R(mg/cm^2)=412\cdot E_{max}^{1.265-0.0954\ \ln E_{max}}, \quad 0.01<E_{max}<2.5MeV \quad ...(a)$$

the ranges of these secondary electrons in TLD discs are smaller than 20μ. Most of the photon energy transferred to both $CaSO_4$:Dy

and teflon particles is completely absorbed by themselves respectively when the grain size of $CaSO_4$:Dy is larger than 40μ. The energy absorbed in teflon gives no significant contribution to the TL response. Only the energy absorbed in TL phosphor is responsible for TL response. This is the reason why the relative photon energy dependence of TLD discs does not change obviously while the proportion of TL phosphor varies. The matrix of teflon particles may act only as an attenuator to low energy photon and therefore reduce the TL response per unit weight of TL phosphor.

When the grain size of $CaSO_4$:Dy in TLD disc is smaller than 40μ the secondary electrons originated in TL grains may escape and deposit their residual energy to the surrounding teflon particles. The probability of energy transfer from TL phosphor to teflon increases as the grain size or proportion of $CaSO_4$:Dy decreases. Therefore the photon energy dependence of TLD discs can be significantly reduced by using small-grain-sized $CaSO_4$:Dy particles. It had been stated[7] that not only the handling procedures of such fine particles are difficult but also the sensitivity may decrease to a factor of 8. Hence the attempt to produce energy independent TLD discs by using small-grain-sized TL phosphor is not practical.

The problem of energy dependence of TLD discs can be solved by applying several kinds of filters. Fig.2 shows how the filters affect the TL response of 25% $CaSO_4$:Dy/teflon discs. Curves 5 and 6 in Fig.2 show that an energy independent response beyond 70 keV can be achieved by using a filter combination of 1.0mm Sn + 0.4mm Al or 0.85mm Cd + 0.4mm Al. The decrease in the response under these combined filters below 70 keV and the increase in the response of bare or 1.0mm Al + 1.5mm plastic filtered TLD discs appear to be complementing each other. It had been found that the energy independent response (I) can be evaluated by the following equation[7],

$$I = 0.9A + 0.1B \quad \text{.....(b)}$$

Where A is the reading of 1.0mm Sn + 0.4mm Al or 0.85mm Cd + 0.4mm Al filtered TLD discs, and B is the reading of bare or 1.0mm Al + 1.5mm plastic filtered TLD discs. The data of our experiments are listed in Table 1. Maximum error of the dose estimated by this method is 10%.

Table 1. Calculated Energy Independence of $CaSO_4$:Dy/ Teflon Discs Using Two Sets of Filters.

RELATIVE RESPONSE / E_{eff} (keV) / FILTERS	42	64	86	110	662	1250
A:1.0mm Sn + 0.4mm Al	0.19	0.67	0.81	0.95	1.00	1.00
B:1.0mm Al + 1.5mm Plastic	8.33	4.95	3.26	2.36	1.02	1.00
I = 0.9A + 0.1B	1.00	1.10	1.06	1.09	1.01	1.00

It is suitable to use three TLD discs with bare (B_o), 1.00mm Al + 1.5mm plastic (B), and 1.00mm Sn + 0.4mm Al (A) filters in a badge for a personnel β-γ dosimeter. The difference between the readings of B_o and B gives the information of effect of beta in β-γ mixed fields. The readings of A and B can be used to determine γ(or x) dose by applying eq.(b). The "effective energy" of the photons can be determined by the ratios of these two sets of readings.

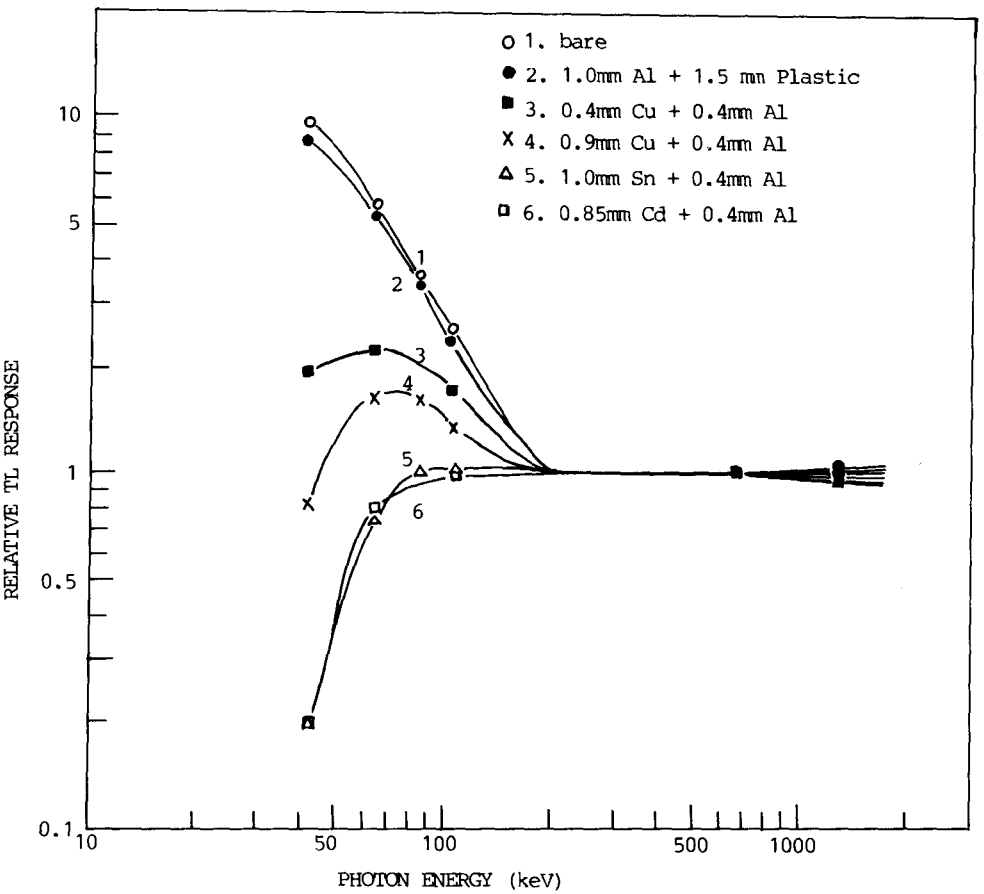

Fig.2. Effects of different filter combinations on the photon-energy-dependent TL response of 25% TLD-$CaSO_4$:Dy/teflon discs.

REFERENCES

1. P. Bassi and G. Busuoli, Intern. J. Appl. Radiation Isotopes, 27, 291 (1976).
2. A.S. Pradhan, R.K. Kher, A. Dere and R.C. Bhatt, Intern. J. Appl. Radiation Isotopes, 29, 243 (1978).
3. T. Yamashita, N. Nada, H. Onishi and S. Kitamura, in Proc. 2nd Int. Conf. on Luminescence Dosimetry, Gatlinburg (1968).
4. C.S. Wuu, C.F. Wu and S.J. Su, Nucl. Sci. J., 19, 108 (1982).
5. C.F. Wu, C.S. Wuu and S.J. Su, Hoken Butsuri, 18, 129 (1983).
6. L. Katz and A.S. Penford, Rev. Mod. Phys., 24, 28 (1952).
7. A.S. Pradhan and R.C. Bhatt, Nucl. Instr. and Meth., 16, 233 (1979).

INFLUENCE OF CHEMICALS ON UNIRRADIATED LIF THERMOLUMINESCENCE DOSEMETER READING

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In determining the partial body dose of a large number of persons using thermoluminescence dosemeters (TLD's), contact of the TLD with detergents or other chemicals during monitoring cannot definitely be ruled out. Studies were carried out into whether a dose can be simulated by these means in TLD evaluation and how a simulated dose can be recognized in routine evaluation (1).

Method

Unirradiated, regenerated TLD's of LiF were studied both as TLD chips (Harshaw, TLD 100, 1/8 x 1/8 x 0.035") and as small Teflon disks with embedded LiF (Teledyne Isotopes, D-LiF 7-0.4). In order to determine the influence of liquids on the TLD reading, the dosemeters were immersed in the liquid for a few minutes and subsequently kept in the dark to dry for 20 h before evaluation. In some of the tests, the TLD's were rinsed with distilled water before drying.
The TLD's were stored over concentrated solutions of HNO_3, HCl and NH_3 for one to two hours in order to study the influence of aggressive vapours.

The TLD's were evaluated in a reader of the Harshaw 2000 A/B type. During the evaluation the multiplier current in the reader was only integrated for planchet temperatures of 160 °C to 240 °C. Light emitted from the TLD at lower temperatures was not included in the evaluation. The mean background of unirradiated TLD's not influenced by chemicals is subtracted from the readings. The glow curve was recorded during evaluation. Immediately after evaluation the TLD was evaluated a second time and the glow curve recorded once again (Fig. 1). The glow curves were plotted semi-logarithmically because of the large dynamic range of the emitted light intensity. The chemical influence was generally studied on 2 TLD's. If a major influence on the TLD reading was found then the measurements were repeated with 10 TLD's.

Results and Discussion

The increases in the TLD readings caused by non-radiation-induced light emission after the influence of various chemicals are compiled in Table 1. In many cases the apparent dose induced by chemicals is smaller than the fluctuations in background of the unirradiated TLD's. The influence of the chemicals is larger with TLD's of LiF in Teflon than with TLD chips. Non-radiation-induced light emission is particularly large after treatment with $NaPO_3$, NaOCl, NaOH and certain detergents. The fluctuations of the measured values about the mean are unusually large. Lingertat (2) has already pointed out this property of non-radiation-induced light emission. With the aid of the glow curve from the first or second TLD evaluation it can always be recognized whether an apparent dose generated by chemical influence is being measured during TLD evaluation. A comparison of the curves in Figs. 2-4 with those in Fig. 1 shows this for several examples. Whereas after treatment with water, NaOCl and the detergents Sunil and Fakt a clear difference in the glow curves from those of the irradiated dosemeters is established, this is not the case after treatment with NaOH. With NaOH, however, the glow curve clearly differs from that of irradiated TLD's when repeating the evaluation.

Teflon disks without LiF treated with detergents, NaOCl or NaOH displayed similar glow curves during evaluation in the TLD reader and approximately the

same dose increases as TLD's of LiF in Teflon after chemical influence. A non-radiation-induced light emission is obtained with the detergent "Sunil" if a few grains of this substance are heated in the TLD reader. If the TLD's of LiF in Teflon are rinsed with water after being kept in the Sunil solution then the value of the non-radiation-induced light emission drops by more than one order of magnitude.

Summary

The influence of chemicals on TLD's can bring about readings during evaluation which correspond to doses of less than 1 mSv in the case of LiF chips and to doses of less than approx. 20 mSv with TLD's of LiF in Teflon. The chemical influence in the tests described here is many times larger than could unintentionally occur in radiation monitoring with TLD's. Apparent doses caused by chemical influence can be recognized in TLD evaluation and corresponding errors can be excluded.

References

1. Schumacher, R.
Untersuchungen zur Fehlererkennung bei der Routineauswertung von Thermolumineszenzdosimetern
Internal Report: ASS-No. 0366, Jülich 1982

2. Lingertat, J.
Untersuchungen zur Personendosimetrie mit Lithiumfluorid
Dissertation, Dresden 1967

Table 1: Apparent Dose after Influence of Chemicals on Unirradiated TLD's

Group	Chemical	Concentration**	Exposition time	apparent dose ***	
				LiF chips	LiF in Teflon
		%	min	mSv	mSv
	H_2O	-	30	0.03	0.36
acids	CH_3COOH	10	5	0.02	0.07
	H_2SO_4	10	5	0.03	0.06
	HCl	10	5	0.02	0.00
	HNO_3	10	5	-	-0.02
	H_3PO_4	10	5	0.03	0.22
lye	NaOH	10	5	0.58	7.6 ± 6.5
dissolved salts	NaCl	10	5	0.03	0.15
	NaI	10	5	0.04	0.12
	Na_2SO_3	10	5	0.03	0.29
	Na_3PO_4	10	5	0.21	0.79 ± 0.32
oxidizing agents	$KMnO_4$*	5	5	-	0.77 ± 0.51
	$KMnO_4$*	5	30	0.02	-
	H_2O_2	10	5	0.05	0.70 ± 0.39
	NaOCl	10	5	0.22	2.7 ± 2.4
aggressive vapours	NH_3	-	120	0.01	0.22
	HCl	-	120	0.02	0.04
	HNO_3	-	60	0.02	0.05
organic substances	methanol	-	5	0.03	0.55
	ethanol	-	5	0.04	0.30
	acetone	-	5	0.03	0.49
	methyl-acetate	-	5	0.03	0.73
detergents and washing agents (trade names)	soap solution	0.1	30	0.03	1.4 ± 0.2
	Risol	10	30	0.02	0.33
	Decopan 85	-	30	0.10	0.75
	Decopan 85*	-	30	0.01	-
	RBS 50	10	30	0.18	1.1
	RBS 50	-	30	0.20	1.8 ± 0.7
	RBS 50*	-	30	-	2.5 ± 1.2
	Sanso	1	30	-	0.84 ± 0.40
	Calgon	1	30	-	1.7 ± 0.6
	Fakt	1	30	0.07	1.2 ± 0.8
	Mustang	1	30	0.04	1.1 ± 0.2
	Dash	1	30	0.11	1.6 ± 0.3
	Persil	1	30	0.05	5.0 ± 2.5
	Sunil	1	30	0.22	5.3 ± 2.6
	Quanto	1	30	-	7.5

* After chemical treatment the TLD's were rinsed with distilled water before drying.

** If no concentration is given the substances were used in the initial concentration.

*** If measurments with 10 TLD's were carried out then the standard deviation is given in addition to the mean.

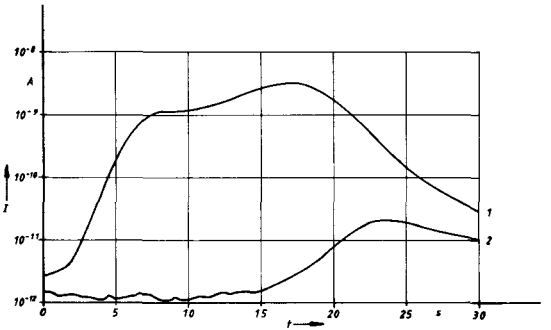

Fig. 1: Glow curve of TLD's of LiF in Teflon irradiated at a dose of 0.03 Sv (1) and glow curve of the subsequent repetition of the TLD evaluation (2)

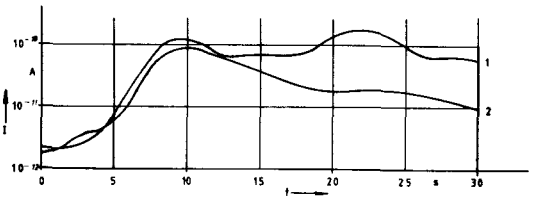

Fig. 2: Glow curve of TLD's of LiF in Teflon after influence of NaOCl solution (1) and distilled water (2)

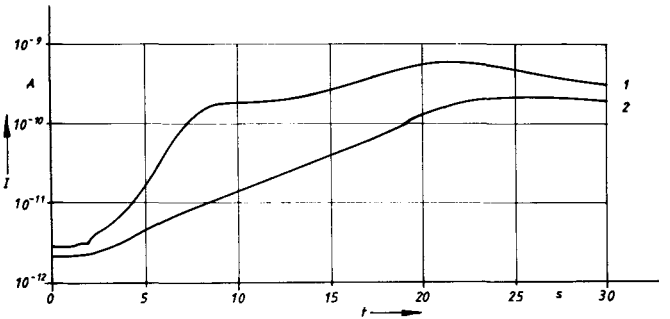

Fig. 3: Glow curve of TLD's of LiF in Teflon after influence of 10 % NaOH solution (1) and glow curve of the subsequent repetition of TLD evaluation (2)

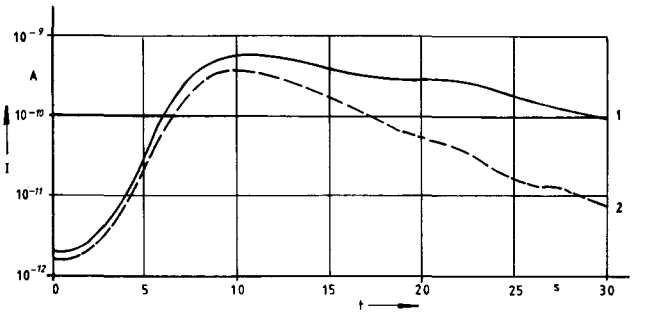

Fig. 4: Glow curve of TLD's of LiF in Teflon after influence of 1 % solutions of the detergents Sunil (1) and Fakt (2)

FIVE YEARS OF EXPERIENCE WITH TL-SYSTEMS FOR AUTOMATICALLY MEASURING THE RADIATION DOSE OF OCCUPATIONALLY EXPOSED PERSONS

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Seit dem Jahre 1973 ist das Thermolumineszenzdosimeter in Österreich im Sinne des §24 der Österreichischen Strahlenschutzverordnung als Personendosimeter für beruflich strahlenexponierte Personen zugelassen. Seit Juni 1978 ist das Institut für Radiochemie als autorisierte Anstalt mit der Personendosimetrie betraut. In den letzten fünf Jahren konnten dadurch wertvolle Erfahrungen mit den im Handel erhältlichen TL-Auswertesystemen erhalten werden. Die statistische Auswertung der erhaltenen Daten ermöglicht ausserdem die Erkennung gewisser Tendenzen der Dosimeterträger.

Zum gegenwärtigen Zeitpunkt werden von uns 2500 Kunden in den Bereichen Personen-, Fingerring-, Neutronen-, Orts-, und Therapiedosimetrie betreut. Dazu stehen uns mehrere Auswertsysteme zur Verfügung:

- Referenzstrahler (Harshaw 2271SC)	
- Auswertgerät (2 Harshaw TL-Reader 2271)	Personen-, Orts-, Therapie-
- Datenspeicher (Teletype)	und Neutronendosimeter
- Minicomputer (Texas Instruments 990/10)	
- Referenzstrahler (Studsvik 6527B)	
- Auswertgerät (Harshaw 2000D)	
- Datenspeicher (Teletype)	Fingerringdosimetrie
- Minicomputer (Texas Instruments 990/10)	
- Referenzstrahler (Studsvik 6527B)	
- Auswertgerät (Teledyne TLD-7300 + CCD)	für die Analyse der
- Datenspeicher (CBM-8050)	Glowcurve
- Mikrocomputer (CBM-8032)	

Praktische Erfahrungen führten zur Aufstellung zweier verschiedener Auswertabläufe für die Dosimeterkarten der Personendosimeter bzw. für die Einzeldosimeterkristalle der Neutronendosimeter. Im folgenden werden die Abläufe der Routineauswertungen für die beiden Dosimeterarten beschrieben.

a) Auswertungsablauf für Personen-,Orts-,Therapie-,und Neutronendosimeter:

- Rücknahme der Dosimeter (TLD-100,600,700) und Ordnung nach Trägernummer
  (Ausgabe - Rücknahme im 2 Monatszyklus; Dosimeterhalterungen zur leichten Identifizierung in 4 verschiedenen Farben)
- Eichung des Auswertsystems an Hand von 20 beliebigen Dosimetern aus 4 verschiedenen Dosimeterchargen
  (Bestrahlung der Dosimeter mit 1, 2, 4, 8 mSv mittels Eichstrahler; nach Auswertung dieser Dosimeter berechnet ein Programm den Leerwert sowie den Kalibrierungsfaktor für alle Dosimeterkarten dieses Monats)
- Auswertung der Dosimeter
  (Vollautomatischer Harshaw TL-Card-Reader 2271)
- Datensicherung
  (TTY - Lochstreifenstanzer + Datenfiles auf EDV - Harddisk)
- Kontrolle der Datenfiles
  (Vergleich der Ausgabe bzw. Rückgabedaten auf fehlende Dosimeter etc.)
- Computerauswertung
  (Berechnung der Monats-, Quartals-, und Lebensaltersdosis bzw. deren Überschreitungen; Erstellung der neuen Trägerlisten; Printen der Auswertungsbögen für die Strahlenschutzbeauftragten; Printen der Adresslisten für die Rücksendung der Dosimeter)
- Rücksendung der Dosimeter

b) Auswertungsablauf für Fingerringdosimeter:

- Rücknahme der Einzelkristalldosimeter (TLD-100,200)
  (Chips oder Rods)
- Auswertung in Gruppen zu 50 Dosimetern
  (Vollautomatischer Harshaw Single - Crystal - Reader 2000D)
- Datensicherung
  (TTY - Lochstreifenstanzer + Datenfiles auf EDV - Harddisk)
- Eichung des Auswertsystems für jedes einzelne Dosimeter
  (Bestrahlung der jeweils 50 Dosimeter mit 1, 2, 4, 8 mSv mit Studsvik Reference Irradiator 6527B; daraus folgen wieder Leerwert und Kalibrierungsfaktor für jedes einzelne Dosimeter)
- Computerauswertung
  (wie oben)
- Rücksendung der Dosimeter

Nach Abschluss der Routineauswertung findet die statistische Verarbeitung der EDV-gespeicherten Daten in Hinsicht auf Häufigkeit der ausgewerteten Dosen in den Dosisbereichen: 0 - 0.1 mSv; 0.1 - 0.3 mSv; 0.3 - 1 mSv; 1 - 4 mSv und über 4 mSv statt, zusätzlich noch aufgeteilt nach Berufsgruppen der Dosimeterträger: "0" Universität, Behörden; "1" Radiologen; "2" Zahnärzte; "3" Ärzte; "4" Industrie; "5" Krankenhäuser. Die Verteilung der Dosimeterträger in diesen Berufsgruppen wird ebenfalls laufend ermittelt.

Tab.1) Verteilung der Dosimeterträger nach Berufsgruppen (1978-Juli 1982) in Prozent %

	"0"	"1"	"2"	"3"	"4"	"5"
1978	10.2	1.1	9.1	2.2	2.3	74.9
1979	6.9	2.5	19.1	6.3	2.2	63.0
1980	6.4	2.2	26.6	8.4	2.1	54.2
1981	8.1	2.3	22.3	9.3	2.5	55.0
1982	8.0	2.2	22.1	11.2	2.2	54.2

Tab.2) Verteilung der ausgewerteten Dosen (1978-Juli 1982) in Prozent %

	0 - 0.1	0.1 - 0.3	0.3 - 1	1 - 4	über 4 mSv
1978	87.5	9.7	2.2	0.4	0.2
1979	93.0	5.6	1.2	0.2	0.03
1980	85.1	12.1	2.2	0.5	0.6
1981	88.8	9.0	1.7	0.4	0.1
1982	97.6	1.7	0.4	0.2	0.04

Die Erfahrungen der letzten fünf Jahre wurden dazu genutzt, ein TL-Auswertsystem für die Routineüberwachung von beruflich strahlenexponierten Personen aufzubauen. Nach Vergleich von verschiedenen Systemen wurde, auch aus Gründen der Kompatibilität in Österreich, eine Kombination aus Harshaw - Auswertgeräten und Referrnzstrahlern, Studsvik - Referenzstrahler und einem Texas Instruments 990/10 Minicomputer gewählt. Die entwickelten Auswertabläufe für Dosimeterkarten und Einzelkristalldosimeter führen zu einer Dosisauswertung, welche mit maximal 10% Fehler behaftet ist (Bestimmung durch ein Vergleichsprogramm des Fachverbandes für Strahlenschutz der BRD). Abschliessend wäre zu erwähnen, dass die erhaltenen Erfahrungen zum Anlass genommen wurden, sich genauer mit der Glowcurveanalysis zu beschäftigen, was zu einer Optimierung der verwendeten Dosimeter sowie der Auswertgeräte führen soll.

Literatur

1 R.Früh, P.Brunner, O.Bobleter; Thermoluminescence Characteristics of Manganese-Doped KCl-KBr-Mixed-Crystal Systems; Radiochem.Radioanal. Letters 31 (3) 1977

2 R.Früh, P.Brunner, O.Bobleter; Automatic TLD-Readers-Comparison of Three Commercially Available Instruments; AIAU 78501, Februar 1978

3 J.Beinsteiner; Untersuchung über Alluminiumoxid als Thermolumineszenz-Material für die Personendosimetrie; Dissertation, Institut f. Radiochemie Universität Innsbruck, 1980

4 A.Gotwald; Energieabhängigkeit klinischer Dosimeter; Diplomarbeit, Inst. f. Radiochemie, Universität Innsbruck, 1982

5 P.Brunner, A.Gotwald; Strahlenschutzüberwachung durch VSK, Statistik 1978, 1979, 1980; Versuchsstelle f. Strahlenschutz und Kerntechnik, Institut f. Radiochemie, Universität Innsbruck, 1983

RESULTS OF TESTING AND EVALUATING A HEALTH PHYSICS INSTRUMENT PERFORMANCE STANDARD

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INTRODUCTION

Pacific Northwest Laboratory, operated by Battelle Memorial Institute, is currently performing a 3-year project evaluating a draft ANSI Standard (N42.17) on the performance specifications for Health Physics instrumentation. This paper presents data taken during testing that provides information for the evaluation of the applicability and practicality of the proposed standard and the determination of the degree of conformance of currently available North American commercial instruments to the proposed standard.

The draft ANSI Standard N42.17 on performance specifications for health physics instrumentation was written in 1981 by a task group that included both manufacturers and users of these instruments as well as representation from regulatory bodies; the second draft of this standard is being evaluated. The draft standard attempts to establish minimum acceptable performance criteria for health physics instrumentation for use in ionizing radiation fields. Included are testing methods to evaluate the adequacy of several types of instrumentation for use under specified conditions. The requirements and testing methods discussed in the draft standard are related to five major areas: general characteristics (e.g., zero set, alarm threshold, battery status, AC and DC power requirements), electronic and mechanical requirements (e.g., stability, geotropism, response time), radiation response (e.g., accuracy, precision, energy dependence, angular dependence), interfering responses (e.g., non-ionizing electromagnetic radiations, ionizing radiations), and environmental factors (e.g., temperature, humidity, shock, vibration, ambient pressure.)

The instruments tested fell into one of the following categories: ionization chambers, Geiger Mueller (G.M.) detectors, alpha survey meters, neutron dose equivalent survey meters, or air monitors. Instruments were procured for testing by direct purchase of new production units or by loan from others. All units were calibrated before testing.

METHODS

Instrument test and evaluation procedures were developed with emphasis on the requirements and testing guidance stated in the draft ANSI standard and with additional criteria from other draft and current ANSI and IEC standards. Each procedure was verified prior to implementation of the testing phase. Initial data obtained with the testing procedures were carefully analyzed to determine that the data

represented the responses of the instruments under the conditions required by the draft standard and that those conditions existed for all of the instruments evaluated.

Instruments that have been tested and evaluated were procured by three methods: 1) direct purchase of off-the-shelf units, 2) loan of new or recently manufactured units from manufacturers, and 3) loan of units from DOE contractors. Approximately 80 instruments have been procured by these methods.

Twenty readings were taken for each data point for all tests to assist in obtaining statistically reliable data.

RESULTS

Initial results from eight tests will be discussed for G.M. and ion chamber type instruments. These tests include mechanical and electronic tests (i.e., stability, geotropism, response time), tests of radiation response (i.e., precision, accuracy, energy dependence), and environmental tests (i.e., temperature, humidity).

Stability

The standard evaluates the drift at constant temperature and pressure for a 3 hour period following a 10 minute warmup period. During this period, the readings are expected to remain with ±3% of full scale or decade on the most sensitive scale. The test is performed by affixing a source to the instrument which produces a mid-scale reading on the most sensitive scale and then recording readings every 30 minutes for 3 hours. Based on forty instrument tested, nine of 30 G.M. instruments and none of the ion chambers can be considered to have failed based on a greater than ±3% deviation at any one time.

Geotropism

Geotropism is defined by the standard as "a change in instrument response with a change in instrument orientation as a result of gravitational effects." The standard requires that a change in reading due to spatial orientation shall not exceed ±3% of the full-scale reading on the most sensitive scale. The instrument is tested by rotating it through two perpendicular planes and recording readings at 90° increments. During the testing, a source is affixed to the instrument which will produce approximately a mid-scale reading on the most sensitive scale. Again 20 readings are recorded at each point and the mean and standard deviation are determined. Of the 24 instruments tested to date, six of the instruments can be considered to have failed based on a greater than ±3% deviation in the reading at any one angle. However, it must be recognized that the statistical fluctuations are large due to inadequate precision of the instruments on the low ranges thus making it impossible to provide an unqualified judgement.

Response Time

Response time requirements in the standard are different for various dose rate ranges. These range from maximum response times of 30 seconds to 1 second for ranges of <0.01 mR/h and >1000 mR/h, respectively. Most instruments were tested against the response time requirement of 5 seconds for ranges of >0.1 to 100 mR/h. Instruments are tested by exposing them to a radiation field such that a response between mid and full scale is established. After equilibration, the radiation

field is removed instantaneously and the time is recorded to reach the background value. After equilibration at background, the source is re-applied and the time required to reach 90% of the initial equilibrium value is recorded. This is repeated on all ranges.

Twenty G.M.-type instruments were tested on a total of 55 ranges. Four of these instruments had selectable time constants and it was found that invariably the longer time constants failed to meet the requirements. On the instruments tested, 11 instruments failed to meet the performance criteria typically on high or very low dose rate ranges.

Precision

The standard requires that the instruments have a relative standard deviation of ≤2.5% on all ranges. A total of 21 instruments were tested on 49 ranges; 8 of these instruments were ion chambers and 13 were G.M. detectors. One ion chamber and all of the G.M. detectors failed the test. For ion chambers, only one range out of 18 tested failed; only 6 out of 31 ranges passed for G.M. detectors. The relative standard deviations of approximately 15% on the lowest ranges of the G.M. detectors complicates the interpretation of tests such as those for geotropism and stability which require readings on the lowest range.

Accuracy

For ionizing electromagnetic radiation, the response of the instrument compared to the conventionally true value is required to be within ±15%. This measurement is made at the 95% confidence level with ^{137}Cs radiation. The measurement must be made at approximately 25% and 75% of each range. Sixteen instruments (i.e., 4 ion chambers, 12 G.M. detectors) were tested on a total of 39 ranges. One ion chamber and 6 G.M. detectors failed the standard. Most of the failures were observed on the higher ranges.

Energy Dependence

For energy dependence, the standard states that the useful range shall be from 40 keV to 3 MeV. Over this range, the response of the instrument shall be within ±20% of the response for the reference energy. This permits selection of the energy to optimize the range of uniform response. To date, the energy dependence measurements have been limited to the low energy end of the range (ie., 23.7 to 248 keV) Sixteen instruments were tested with only five instruments (three from the same vendor) meeting the criteria in the standard. All of the seven G.M. type instruments failed.

Temperature

The standard states that instruments shall remain operational over the temperature range 0 to 40°C and should be operational over the range of -20 to 50°C. Instruments are exposed in an environmental chamber to an appropriate reference radiation source of sufficient strength to give a midscale response on any scale or decade. The temperature is raised at a rate of 10°C/h until the maximum test temperature is reached, and the results recorded at least at 10° increments. The temperature is reduced at a rate of 10°C/h until the minimum test temperature is reached, recording test item readings at 10 C intervals. Data was obtained at -20, -10, 0, 10, 20, 30, 40 and 50°C after an equilibration time at each level for all

instruments. Non-operational is interpreted as having mean instrument readings greater than 15% from the mean reading obtained at 20°C. Corrections for air density changes are made where necessary.

Thirty four instruments were tested (i.e., 15 ion chambers and 19 G.M. detectors). All G.M. detectors passed the test while 9 of the ion chambers were non-operational (i.e. failed). Five of the 9 ion chambers that failed were marginal failures; that is a mean instrument reading fell outside the ±15% criterion, but just barely.

Humidity

The standard states that instruments shall remain operational over the relative humidity range of 40-95% referenced to 0 and 40°C. The instrument is exposed in an environmental chamber to an appropriate reference radiation source of sufficient strength to give a midscale reading on any scale or decade. The instrument is allowed to equilibrate for 24 hours at 40 ± 2°C and 40 ± 5% relative humidity, and the reading noted. The relative humidity is raised to 95 ± 5% and held for 24 hours, and the instrument scale reading recorded. The relative humidity is lowered to 40 ± 5% while maintaining the temperature at 40°C. After 24 hours, the instrument reading is recorded. The above test is then repeated identically in all respects except that the temperature is 0 ± 2°C.

Twenty-two instruments were tested (i.e., 5 ion chambers and 17 G.M. detectors). One of the ion chambers and 4 of the G.M. detectors failed the test. Most of the failures were observed after the elevated-humidity phase of the exposure.

CONCLUSIONS

For the instruments tested to date, the G.M. detectors and ion chamber instruments fall into two distinct categories. Ion chamber instruments generally lack the sensitivity of the GM detectors but can meet the requirements of the standard. The G.M. detectors seldom meet the test of radiation response and electronic requirements of the standard, and their poor precision makes it difficult to make definitive statements concerning their performance on some tests.

Recommendations to the ANSI working group will include comments on: 1) obtaining statistically reliable data, 2) the precision requirement of the relative standard deviation of ≤2.5% on all ranges, 3) equilibration periods for the environmental tests, and 4) the need for quality assurance information in the standard.

MEASUREMENT OF RADIATION DOSES DUE TO NUCLEAR CRITICALITY ACCIDENTS

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Perfomance characteristics of biological and external dosimetry systems used to measure acute, high-level radiaton doses associated with nuclear criticality accidents have been determined based on results of twenty nuclear accident dosimetry intercomparison studies conducted annually at Oak Ridge National Laboratory since 1964. During these studies, more than 60 different agencies measured neutron and gamma radiation doses greater than 0.1 Gy at area monitoring stations and on anthropomorphic phantoms following simulated criticality accidents produced by operating the Health Physics Research Reactor in the pulse mode. Neutron doses were measured for a variety of energy spectra using foil activation, thermoluminescent, simulated blood sodium activation, and human hair activation methods. Thermoluminescent, radiophotoluminescent, and film dosimeters were used to measure associated gamma doses.

Performance standards for criticality accident dosimetry systems suggest that preliminary radiation doses should be determined with an accuracy of + 50% within 24 hours after exposure. Composite results of recent intercomparisons indicate that about 80% of the neutron measurements and about 70% of the gamma measurements satisfied this criteria relative to reference values. Performance standards for final dose estimates suggest accuracies of + 25% for neutron dose and + 20% for gamma dose. Analysis of the final reported experimental results shows that approximately half of the neutron and gamma dose measurements met the subject criteria. The greatest difficulties in measuring accident doses occurred for radiation fields with a large fraction of low energy neutrons and a high gamma component. Results of these studies emphasize the importance of considering measurement location, neutron energy distribution, and dosimeter sensitivity in the evaluaton of accident doses.

SELECTION CRITERIA FOR DETECTORS FOR ENVIRONMENTAL DOSERATE MEASUREMENTS AROUND NUCLEAR INSTALLATIONS

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Introduction

The external exposures to the population in the vicinity of nuclear facilities by airborne radioactive effluents can be assessed, with a detection limit of some 10 µSv per year above background, by long-term measurements using TLD dosimeters (1). This is done today routinely around most nuclear power plants in the western world.

The information gained from such measurements, however, is not readily or momentarily attainable. Furthermore, it will come much too late in case of abnormal or accidental emissions, to derive any recommendations or even measures for preventive dose reduction.

Consequently, many countries exploiting nuclear power have issued regulations for doserate measuring stations to be operated around nuclear reactor sites(2,3). The stations may be set up at the site boundaries, or, preferably, in the midst of settlements in the neighbourhood of the site.

The main point is that the measured values are transferred directly to a central control station where the data are available at any time. The control station doesn't necessarily need to be located at the reactor itself. It will be, in many cases, operated by the local public safety authorities.

In establishing performance criteria for such environmental doserate measuring networks, many aspects have to be taken into consideration. The most incisive decision, however, is the choice of a suitable detector because, in spite of all progress achieved with modern electronics and microprocessors, it is still the detector that determines the essential characteristics of the system.

It is the aim of the following contribution to specify some important requirements, and to suggest detectors meeting them.

Detector Requirements

The requirements influencing the selection of detectors cover: doserate range and accuracy of lower doserate measurements; energy range; directional dependence; temperature dependence; long-term stability and detector lifetime; economy. Several standards or guidelines relating to environmental gamma ray doserate measurements have been published in the last years (4,5,6,7). However, their specifications, on the one hand, differ widely, and on the other hand do not take into account all of the relevant points enumerated above. A comparison has been given in (8), and the major conclusions will be discussed here.

The external doserate caused by the natural terrestrial, atmospheric and cosmic background radiation, is assumed to be about 0.1 $\mu Sv h^{-1}$. There are, however, many regions where it is considerably less. The requirements, with respect to the lower measuring range, vary between 0.1 and 0.01 $\mu Sv h^{-1}$, however in most cases without specifying what the statistical accuracy of the measurement shall be, and what measuring times are acceptable to achieve this limits.

By combining and averaging the various requirements, we arrive at 0.04 $\mu Sv\ h^{-1}$ to be assessed within $\pm$ 10%, single statistical standard error, and for 60 s measuring time. Less stringent requirements of 0.1 μSvh^{-1} within 3600 s are applied for measurements used only for accident monitoring.

The upper limit of the measuring range is not clearly defined in the specifications either. In practice a value between 1 and 10 Svh^{-1} has been adopted bearing in mind incident monitoring. Two detectors are then necessary and sufficient for covering the entire range.

The next important feature is the energy range of the detector. Most standards state 50 keV as the lower edge, with a tolerable deviation between 25% and 40% from the indication at 662 keV. 50 keV seems to be a compromise between the technical possibilities of usual detector systems and the necessities of the measurement. It has been shown that in case of Xe-133 emission, which is a comparatively likely event, approx. 60% of the exposure rate from the cloud is due to energies lower than 50 keV (9). Therefore, detectors would be preferable with an energy range down to about 30 keV - 25 keV.

The requirements for the upper energy range limit vary between 1.3 MeV and 3 MeV. This, however, is no great issue because most detectors are oversensitive in this range anyway.

With regard to directional dependence, an average sensitivity is asked for over the full sphere that is not lower than 70% of that in the most sensitive direction, again referring to 662 keV.

Primarily in the English-speaking countries, there are also requirements for detector arrangements which are screened against the lower hemisphere, i.e. have an extremely asymmetrical response distribution in favour of the upper hemisphere. This lowers the background value and improves the detection limit for cloud radiation. However, this arrangement seems unrealistic for general assessment of human exposure dose. Man is not screened against the lower hemisphere either.

The range of temperature independence requirements begins at -25°C and ends at +50°C, with an acceptable deviation of $\pm$ 50%.

When a good long-term accuracy must be obtained, a low temperature dependence is an absolute must. An error span of $\pm$ 50% seems totally unacceptable. It is not technologically determined either since there are better detectors.

For the upper temperature limit 50°C is often not enough since temperatures up to 70°C can be reached in the probe when the detector is mounted outside and the sun shines.

However, when determining the temperature dependence, it should be taken into account, that all components mounted out of doors, i.e. preamplifiers etc. must be included in the test. They may constitute the factor determining the deviation.

Detector Performance

All detector types used so far for environmental monitoring - ionisation chamber, scintillation counter and counter tube - have their strengths and weaknesses. Judged by the requirements discussed in this paper and by the current state of the art, however proportional counter tubes of modern design are particularly well suited.

To obtain the lower detection limits referred to above, with a pulse-generating detector, a response of the probe is required of at least 42 s^{-1} (sensitive measurement) or 0.35 s^{-1} (accident measurement) per 1 μSh^{-1}. The statistical error of the signal from high-pressure ionisation chambers as used today for environmental monitoring corresponds roughly to that of a counter tube with 40 s^{-1} per 1 μSvh^{-1}. There are, however, proportional counter tubes being applied already for high-precision low-level doserate measurements which produce a countrate of more than 200 s^{-1} per 1 μSvh^{-1} (10).

Another problem nevertheless arises when natural background doserates are to be measured with a high degree of accuracy. Part of the natural background is cosmic radiation, the hard component of it amounting to about 0.03 μSvh^{-1} at sea level and more at higher elevations. Detectors, however, are calibrated at much higher doserates and usually with 662 keV gamma radiation. They may show a very different doserate response for the very high cosmic energies. It has, in fact, been observed that different detector systems may give background readings differing by a factor of two or more, even if they have been thoroughly calibrated in the upper dose range (12). Measures to be taken have been proposed in (11).

Since counter tubes produce a digital signal, they demand much less in terms of stability from the associated electronics than ionisation chambers or scintillation counters with analog signal outputs. Compared to GM counter tubes, proportional counter tubes have a considerably longer life.

Scintillation counters, on the other hand, suffer from a relatively high temperature dependence that has to be compensated for to achieve acceptable results.

It is possible to build proportional counter tubes where the influence of temperature is virtually zero between -20°C and +70°C.

In the following table, the performance figures of doserate and energy dependence are given for two newly developed detectors, covering together a doserate range from 10^{-8} Svh^{-1} to 5 x 10^{0} Svh^{-1}. HV supply and amplifier are integrated in the probes, and pulses may be transferred by suitable cables, over distances of several km.

	Proportional counter tubes		Remarks
	LB 6121 High-doserate probe	LB 6123 Low-doserate probe	
External dimensions	12 mm Ø x 96 mm	53 mm Ø x 400 mm	without HV/preampl.
Calibration factor	$4x10^{2}$ s^{-1} per 1 $mSvh^{-1}$	$4x10^{4}$ s^{-1} per 1 $mSvh^{-1}$	
Lower doserate range	4 μSvh^{-1}	0.04 μSvh^{-1}	within ± 10% 1σ, 60 s
Upper doserate range	1 Svh^{-1}	5 $mSvh^{-1}$	10% deviation from linearity
Energy range	50 keV - 2 MeV	40 keV - 2 MeV	within ± 40% from 662 keV
Temperature range	-20°C - +70°C	-20°C - +70°C	within ± 10%, including preampl.
Detector lifetime	> 5 years	> 5 years	

References

1. J. Czarnecki, M. Baggenstos, J. Schuler, H. Völkle; Eine Methode zur Auswertung von Messresultaten der Umgebungsüberwachung von Kernkraftwerken mit Thermolumineszensdosimetern.
An evaluation method for TLD-results from dose measurements around nuclear power plants. In: Radiologische Auswirkungen von Kernkraftwerken und anderen kerntechnischen Anlagen auf den Menschen und seine Umwelt. FS-82-27-T (1982).

2. Richtlinie zur Emissions-und Immissionsüberwachung kerntechnischer Anlagen, gemeinsames Ministerialblatt Ausgabe A, Nr. 32, vom 26.11.79 S. 668-683.

3. Compare NRC Regulatory Guide 1.97

4. Spiers, F.W., et al: A Guide to the Measurement of Environmental Gamma-Ray Dose Rate, BCRU 1981, (Report published by the national Physical Laboratory, Tedington, Middlesex TW 11 OLW, U.K.).

5. Portable or Installed X or Gamma Radiation Absorbed Dose Rate Meters for Measuring Environmental Dose Rate in Air, IEC-Dokument 45 B (Central Office) 52, Genf, März 1983

6. Rahmenempfehlung für die Fernüberwachung von Kernkraftwerken vom 11.9.80, BMI RS II 4-515 607/3

7. Anforderungen der PTB an ortsfeste Strahlenschutzmeßsysteme für die innerstaatliche Zulassung zur Eichung vom 15.11.1981, PTB-Mitteilung 91, 6/81, S. 454-458

8. Maushart, R., Kriterien zur Auswahl von Detektoren für Dosisleistungsmeßsysteme zur Umgebungsüberwachung kerntechnischer Anlagen, Atomenergie-Kerntechnik Vol. 43 (1983) No. 2 p. 91-94.

9. Beck, H.L.: Spectral Composition of the Gamma-Ray Exposure Rate due to Noble Gases Released during a Reactor Accident. Health Physics 43 (No. 3) (1982) 335-343

10. R. Maushart, J. Eppler, Dosisleistungsmessung zur Immissionsüberwachung in der Umgebung von Kernkraftwerken, Nuclex International Fair Conference, Session E Paper No. 5, Basel 1981

11. R. Maushart, Some Remarks Relating to Calibration of Environmental Doserate Measuring Systems, Radiation Protection Dosimetry, Vol. 3 No. 1 (1983).

12. Bøtter-Jensen, A Nordic Intercomparison for Background Radiation Monitoring Systems, Risø M 2266, April 1981.

A LARGE VOLUME IONIZATION CHAMBER SYSTEM FOR ENVIRONMENTAL MONITORING

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INTRODUCTION

In view of decreasing tolerance levels continuously increasing accuracy requirements for environmental monitoring have forced the development of such equipment to its physical and technological limits. Although ionization chambers inherently provide optimum dosimetric performance, GM-counters are most widely used for this application due to the much simpler electronic circuitry required. Steel-walled high-pressure ionization chambers show a rather poor energy response with distinct low energy cut-off.

In continuation of our long term research and development programme on secondary standard dosimetry systems[1] we designed and tested a large air-equivalent ionization chamber with 120 l sensitive volume, operated at normal atmospheric pressure, which can be used for environmental monitoring. As compared to available high pressure chambers and GM-counter instruments the large chamber provides considerably improved dosimetric performance.

IONIZATION CHAMBER

The ionization chamber consists of a cylindrical barrel made from polyethylene with the outside dimensions 510 mm diameter and 760 mm height. The wall thickness is 5 mm corresponding to an area mass of 0,45 g/cm^2 . The walls are coated with graphite on both sides. The wall thickness provides secondary electron equilibrium for photon energies up to several MeV. For compensation of wall absorption losses in the low-energy range the inner wall surface is coated with a thin layer of Al generating a higher photon electric yield as compared to air. For higher energies this compensating layer is transparent. By this method an energy response of $\pm$ 2 % in the range of 30 keV$_{eff}$ and ^{60}Co is achieved. For comparison present GM-counter instruments with energy compensating shields as well as high pressure chambers provide an energy response of about $\pm$ 25 % between 60 keV and ^{60}Co with a complete cut-off below 50 keV. .

Due to the cylindrical shape of walls and central electrode the directional response around the circumference is quite uniform. With an effective volume of 120 l the chamber calibration factor is 6,8 kg^{-1} (2,6 . 10^{4}R/C). With a chamber high voltage of 300 V the saturation loss is 10 % at 10 mSv/h (1 R/h). In order to extend the dynamic range to higher doserates with a saturation loss below 1 % at 0,1 Sv/h (10 R/h) the collecting electrode of the chamber is devided into two electrically separated parts deviding the chamber into two sensitive volumes in the ratio 1:100. The chamber is hermetically sealed. Due to the flexibility of the wall material the volume changes according to the outside atmos-

pheric pressure and the internal air mass, which determines the sensitivity, remains constant independent of barometric pressure and air temperature.

Fig. 1 is a photograph of the chamber as used for environmental monitoring in our Research Center.

ELECTRONIC CIRCUITRY

For the measurement of ionization currents in the range of 10 fA up to 100 nA for doserates from 10 nSv/h (1 µR/h) up to 0,1 Sv/h (10 R/h) an electrometer amplifier circuitry with analog/digital converter is used. In order to cope with a dynamic range over 7 decades two amplifiers are used simultaneously each connected to one part of the collecting electrode. For the amplifier an integrated Varactor-op-amp-modul (Teledyne-Philbrick Mod. 1702) is used which provides a sufficiently low off-set current of 5 fA. The amplifier output is fed into a voltage-to-frequency-converter (BURR BROWN VFC 62) with a resolution of 0,03 % full scale at 100 kHz and a dynamic range of 6 decades. Both amplifiers and the VFC are enclosed in a thermally isolated box containing a Peltier element for temperature stabilization. By changing power supply polarity it can be operated as a cooler or heater and the temperature controlled to $\pm$ 5°C. The complete electronic system is located inside the chamber below the top opening and therefore protected against climatic environmental conditions.

The output pulses of the VFC can be directly transmitted through cable or telephon lines or used for telemetric radio-transmission. The pulse frequency is proportional to the doserate. A remote microprocessor circuitry converts pulse frequency into units of dose or doserate and can be used to operate a network of up to 10 chambers simultaneously in a sampling mode.

PERFORMANCE

The ionization chamber has been calibrated by comparison with our low-level secondary standard chamber LS-10[2] with 10 l sensitive volume, which is traceable to the Austrian Primary Standards. The calibration is performed at a doserate in the range of 10 µSv/h with X-ray qualities and gamma radiation according to ISO 4037. In order to test the linearity of doserate measurement down to the limit of detection a series of ^{226}Ra standards of identical geometry and measured activity in the range of 10 µg to 10 mg has been used. The radium standards were positioned free-in-air in 5 m distance from the chamber. The linear-least-square fit of the doserate readings vs. radium weight resulted in an uncertainty of 1 nSv/h in the range of 0,01 - 10 µSv/h.

After calibration of the chamber the limit of detection has been determined be repeated measurements within the shielded enclosure of a whole-body-counter.
Fig. 2 shows a series of 25 consecutive readings measured with an integration time of 1000 s each in the whole-body-counter. With a mean value of 38,3 nSv/h (3,83 µR/h) the standard deviation is 0,77 nSv/h (0,077 µR/h).

The limit of detection, defined as three standard deviations of the background doserate is therefore 2,3 nSv/h (0,23 μR/h). This is comparable to a typical high pressure ionization chamber.

CONCLUSION

The large volume ionization chamber system described is particularly useful for continuous, accurate and remote measurement of environmental radiation around nuclear installations. As compared to instruments presently commercially available it provides considerably improved dosimetric performance. Extended networks can be operated with one central electronic data processing unit and telemetric data transmission is directly possible due to the digital form of information. The dynamic range extends over 7 orders of magnitude which makes the instrument suitable also for application in early warning systems.

REFERENCES

(1) DUFTSCHMID K.E., WITZANI J., HIZO J.: Automated Dosimetry System for Calibration of Radiation Protection Instruments. Proc. 3rd Int. Symp. of the Soc. for Rad. Prot., Inverness, (1982).

(2) DUFTSCHMID K.E., HIZO J.: Secondary Standard Ionization Chamber System for Environmental Dosimetry. Proc. 1st Symp. of IMEKO TC 8, Leningrad, (1981).

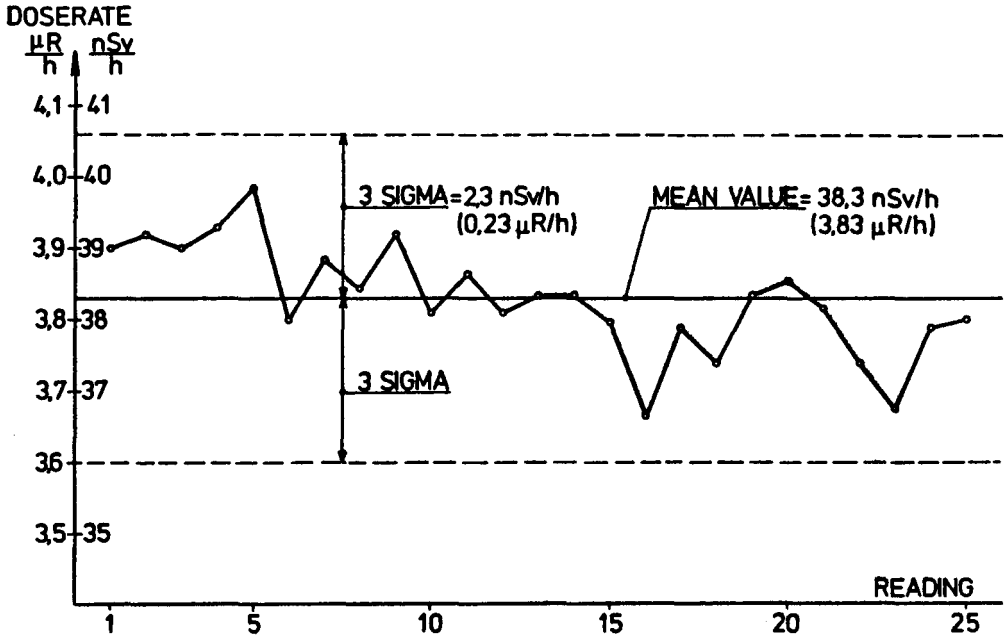

Fig. 2 Environmental Background Doserate measured inside the shielded enclosure of the whole-body-counter

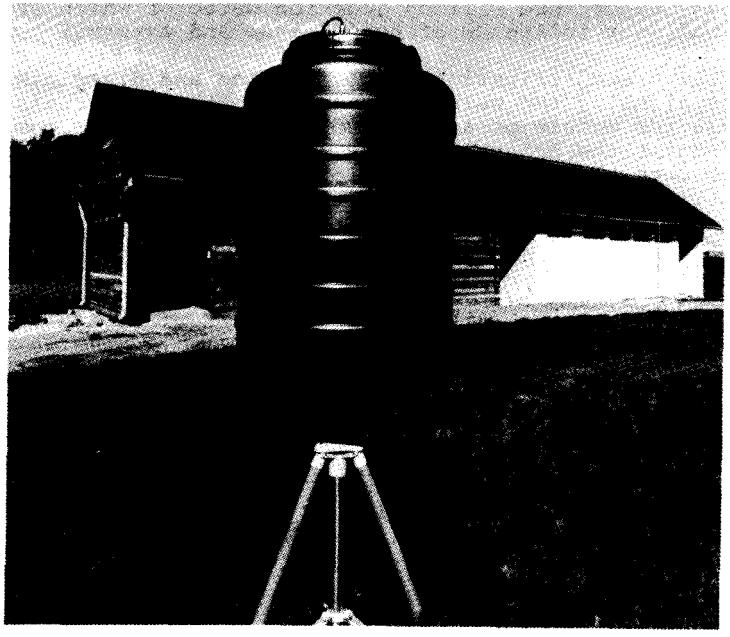

Fig. 1 Photograph of the large volume ionization chamber installed at the Research Center

"AN ULTRA-THIN FILM TRITIUM SOURCE STANDARD FOR INSTRUMENT CALIBRATION"

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1. INTRODUCTION

One requirement called for in consultative documents for revised national legislation on Radiological Protection, following EEC directive L246, is that measuring instruments must be calibrated annually against referable standards. The provision of standard sources for low energy beta emitters, of which tritium is the most familiar, presents special problems because self absorption both distorts the energy spectrum and reduces the number of particles emitted.

We have made a number of sources by the Langmuir-Blodgett (L-B) method and our tests presented here demonstrate that primary and working large-area standards having negligible self absorption for low energy emitters can now be produced which will be of interest to the International Standards Organisation (ISO). Previous investigations carried out elsewhere using radioactively labelled L-B films complement and support our conclusions[1].

2. LANGMUIR-BLODGETT FILMS

2.1 General principles: A monolayer of molecules is conveniently formed when a solution of surface-active material, dissolved in a water-insoluble, volatile organic solvent, is placed drop by drop onto a clean water surface. The drops spread and the solvent evaporates, leaving behind a monomolecular layer of the surface-active material which can then be compressed by moving barriers on the surface, as indicated in figure 1. When a monolayer is compressed into a tightly packed, coherent state it is possible, by passing a carefully prepared solid substrate through it, to transfer the monolayer intact from the liquid to the solid surface, figure 2; this is the L-B technique[2,3]. By repeated passages of the substrate through the monolayer, thicker (multilayer) films can be built up.

It is essential to maintain the monolayer at constant surface pressure throughout the deposition process. The original approach was to use a "piston oil"[1,2], but modern "L-B troughs" employ motor-driven barrier mechanisms linked to a surface pressure sensor via an electronic feedback system. Provided adequate care is taken the L-B technique allows remarkably pinhole-free, homogeneous films to be prepared with fine control of film thickness to within a few tenths of a nanometer. Furthermore it is possible to modify the molecular structure in order to achieve the required film characteristics, including the incorporation of radioactive nuclei.

2.2 Preparation of L-B tritium sources: Tritium-labelled stearic acid, dissolved in toluene (specific activity 3.6 mCi mM^{-1} and concentration 0.45 mg cm^{-3}), was obtained from Amersham International PLC. It had been prepared by the catalytic reduction of oleic acid $CH_3(CH_2)_7\ CH{=}CH(CH_2)_7COOH$ using tritium gas.

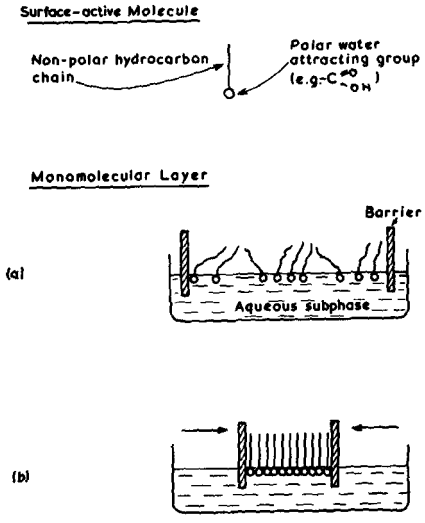

Fig. 1 Schematic showing a monomolecular layer of surface-active material, (a) under low compression and (b) under high compression with the molecules tightly packed.

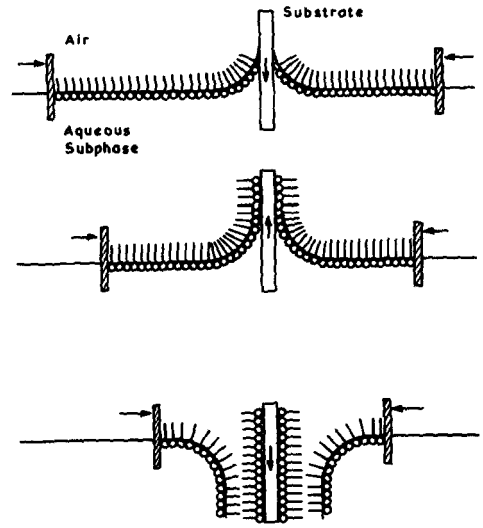

Fig. 2 Transfer of monomolecular layer to hydrophilic solid substrate.

Earthed electrically-conducting substrates are required for absolute emission rate counting. In order to achieve this, aluminium films of thickness 0.15 μm were deposited onto Corning 7059 glass plates 5 cm square by thermal evaporation. Substrates were dipped at 0.5 cm min^{-1} to a typical depth of 3.5 cm using a motor-driven micrometer and the monolayer surface pressure was maintained at 30 ± 2 mN m^{-1} during the deposition process using an electronic feedback system. The subphase was a 2.5 10^{-4} M $CdCl_2$ aqueous solution (pH = 6.05 ± 0.05, T = 20 ± 0.5°C); purified water (resistivity 18MΩ cm) was obtained from a Millipore "Milli-Q" filtration system. Under these conditions fatty acids are almost totally ionised and so the monolayers are converted to cadmium stearate.

3. MEASURING INSTRUMENTS

A Whitlock tritium meter WTM5000 was used for source emission measurements. It comprises a 100 cm^2 flat plastic scintillator detector viewed by two photomultipliers and coincidence pulses are digitally displayed. A rubber moulding excludes external light and acts as a distance gauge (0.50 mm ± 0.12 mm) when the instrument is used for measuring surfaces. A partial vacuum is created in the chamber bounded by the rubber moulding by means of a single-action, manually-operated pump, the reaction spring of which creates a highly reproducible pressure of approximately half an atmosphere.

Emission rates were also measured using a Hurfurth HGZII gas flow proportional detector by mounting and earthing the source on the inside of the mylar window. The activity of material removed on smears was measured using a Nuclear Enterprises LSC1 liquid scintillation counter.

4. SOURCE EMISSION CHARACTERISTICS

The activity cm^{-2} (a_A) of a monolayer source is determined by the specific activity (a_S) and the number of molecules cm^{-2} (n_M) which depends on surface pressure and subphase conditions. In this investigation, a_S = 3.6 mCi mM^{-1} and $n_M = 4.9 \times 10^{14}$ molecules of stearic acid cm^{-2} (equivalent to 8.1×10^{-7} mM cm^{-2}). Hence, $a_A = 2.9 \times 10^{-6}$ mCi cm^{-2} which should result in 107 dps cm^{-2} (4π geometry). Measurements with the HGZII support this value allowing for a value of 15% for backscatter enhancement, but as yet ignoring Bremstrahlung secondaries.

Measurement of activity for 1,5 and 9 layer sources using the WTM5000 gave 2.66, 2.86 and 2.77 cps cm^{-2} of monolayer transferred respectively. This demonstrates that, within experimental error ($\sigma \approx 5\%$), (a) self-absorption with respect to total activity is not significant for thicknesses $\leq$ 22.5 nm, which is in agreement with Beischer[1], (b) L-B films may be built-up in a controllable, uniform fashion , and (c) the WTM5000 has a 5% efficiency in 2π geometry.

5. PHYSICAL TESTS ON SOURCES

In order to assess the suitability of the L-B sources as standards and working sources they were subjected to the tests described below.

5.1 Thermal stability: Sources were held for one hour at various temperatures ($-20°C \leq T \leq 80°C$) in an environmental cabinet, then removed and restored to 25°C in air dried over silica gel. Emission rate measurements showed that the monolayer sources did not degrade at 70°C but a 20% permanent loss of activity was observed at 80°C. Multilayer sources showed a 10% permanent loss at 60°C increasing to 20% at 80°C.

5.2 Effect of relative humidity: Environmental cabinet tests using a range of temperatures below 60°C with relative humidity up to 100% resulted in negligible change of emission rate.

5.3 Effect of pressure: The sources were subjected to vacuum tests down to 0.18 mbar at 25°C for periods of one hour during which no change in emission rate was detected. This stability can be expected to be maintained over longer periods of time because cadmium stearate is much less volatile than the free acid which is known to degrade significantly[4]. Furthermore, emission rates were measured at a range of sub-atmospheric pressures and source-to-detector distances and representative results are shown in figure 3. Note the improved sensitivity that can be gained by measuring tritium sources at short range and reduced pressures.

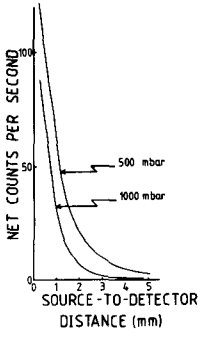

Fig. 3 Emission rate for the 9 layer tritium-labelled cadmium stearate source as a function of source-to-detector distance and air pressure.

5.4 Wipe tests: It is virtually inevitable that any source will be mishandled during its lifetime and so, to be viable, a source must be sufficiently robust to withstand such treatment. Due to the low emission rate from our monolayer sources we anticipated that any small reduction in emission rate resulting from an "average" manual wipe would be difficult to observe. Therefore, for this reason, and in an attempt to employ a quantifiable, more reproducible wipe procedure, we constructed a simple device for wiping sources with cloth wipes ("Nu Con" smears) at a pressure of 25 g cm^{-2} over a fixed stroke of 2.5 cm. A series of wipe tests were carried out on both 1 and 9 layer sources. Source activity was recorded after 5,10,50 and 100 wipes using the WTM5000 and the activity on the smears was measured using the LCS1.

The results of this rather severe abrasive test showed that at the pressure used here the loss of source activity was $\lesssim$ 2% per wipe for both the monolayer and multilayer sources; the WTM5000 and LSC1 data were self-consistent. A more quantitative analysis regarding the adhesive characteristics of the cadmium stearate layers could not be made due to the removal of some of the supporting aluminium film by the smear. Abrasive resistance could be improved by the use of polished solid aluminium plates.

6. CONCLUSIONS

We believe that, with only little further development, both primary and working standard sources of low energy emitters could be produced by the L-B method to allow calibration of measuring instruments, thereby satisfying the requirements of new legislation, the ISO (ISO/TC85/SC2/WG2) and anyone wishing to study unmoderated emission characteristics.

ACKNOWLEDGEMENTS

We thank Mr W L Pitt for experimental assistance.

REFERENCES

1. For example, D.E. Beischer, U.S. Naval School of Aviation Medicine, Naval Air Station, Pensacola, Florida, Project NM 001 059.16, Report Nos 3 (May 1950) and 9 (July 1953); Nucleonics, 11(12), 24 (1953).

2. K.B. Blodgett, J. Am. Chem. Soc., 56, 495 (1934).

3. For collections of papers on L-B films see Thin Solid Films, 68 (1980) and 99 (1983).

4. M. Mizuno, C. Mori and T. Watanabe, Oyo Butsuri, 45(6), 514 (1976).

A SENSITIVE, REAL-TIME TRITIUM WASTE WATER MONITOR*

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HISTORY

Since its beginning over 30 years ago, Lawrence Livermore National Laboratory (LLNL) has maintained a constant environmental-surveillance program to ensure that concentrations of potentially hazardous materials in LLNL effluents are well below current guidelines. One of these effluents is the combined flow of sanitary and industrial waste water. Our waste-water-monitoring capabilities now include on-line low- and high-energy-photon detectors, a pH monitor, a x-ray fluorescence analyzer (XRFA) for metal ions, and a composite sampler. We are improving this system by developing a sensitive, real-time tritium waste water monitor. This monitor will quickly detect and mitigate either chronic or acute tritium spills to the sewer. Our early investigations have been reported previously.[1] We will summarize here the present system design and test results, and indicate directions for future investigations.

Our design goals for the monitor required the system to be--

- sensitive enough to detect 1-20 pCi HTO/ml (0.037 Bq-0.74 Bq. HTO/ml);
- capable of responding within four hours from the time tritiated waste water first enters the monitor;
- economical to operate on a continuous basis;
- rugged enough to operate at a remote environmental sampling station;
- easy to maintain on a routine basis;
- simple enough for a casual user to operate;
- capable of being connected to the existing microcomputer alarm system; and
- capable of obtaining a representative and particulate-free liquid sample.

SYSTEM DESIGN

Figure 1 shows the overall sample flow and electronic structure of the monitor system. It includes a macerator, ultraviolet (UV) sterilizer, "cross-flow" filter disk, passive mixing cell, flow cell detector, and dedicated microcomputer. The macerator reduces all foreign matter to 100 μm diameter or less in size. Because one of the largest maintenance problems is bio-fouling within the tritium monitor tubing, we installed a Model-500 ultra-violet flow sterilizer[2]; the sterilizer has allowed us to achieve the 16,000-μW-s/cm^2 dose recommended by the U. S. Public Health Service for sterilization of drinking water supplies for a few hours each day, and we have observed a reduction in the particulate hang-up within the "cross-flow" filter disk unit and associated tubing. The passive mixing cell is T-shaped; it surrounds the aqueous sample stream with a sheath of liquid scintillator, greatly reducing the chance of viscous gel formation. The dedicated LSI-11 microcomputer[3] makes the necessary quench correction and sends an alarm to a central emergency dispatch center if preset levels are reached. We divided our investigations of the system into two separate tasks: sample collection and radiation detection.

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SAMPLE-COLLECTION INVESTIGATION

We developed a prototype sampling filter of "cross-flow" design. The fast-moving stream in the instrument flume continuously washes across the filter medium, removing debris that would otherwise clog the filter. This cleaning action greatly extends filter life, while providing a clean sample. The prototype is installed in the existing XRFA flume; it consists of a 49-mm diameter filter disk, supported by a perforated stainless-steel backing, sealed with Neoprene O-ring gaskets to the back of the flume and held in place with a screw-on retainer ring (shown in Fig. 2).

We tried a variety of filter disks with pore sizes ranging from 10 μm down to 0.5 μm. (For reference, 1.2 μm is considered general filtration and 0.2 μm is considered sterilization.) The most important quality seems to be the macroscopic surface texture. Smooth-surface filters have lasted 10 times longer than rough-surface filters of comparable pore size. Also, smaller pore size filters lasted longer than similar large pore size filters. For example, 1.0 μm pore size filters lasted 70% longer on the average than identical 3.0 μm pore size filters. A 1.0 μm Teflon[4] filter disk, wetted with ethanol, has lasted the longest, approximately 10 days, at a filtrate flow rate of 0.23 ml/min. Visual inspection showed the filtrate to be particulate-free, thus fulfilling one of our most important design goals.

RADIATION-DETECTION INVESTIGATIONS

Liquid scintillation counting with a continuous flow cell appeared to be an optimum compromise between sensitivity and simplicity. The sample stream is mixed with a precise amount of liquid scintillation cocktail before being pumped through a flow cell detector. We set a design goal to limit cocktail consumption to one gallon (3.8 liters) per week to reduce operating costs.

There are a number of simple liquid scintillation, flow cell detectors available commercially. Most monitor high-specific-activity effluents from high-performance liquid chromatograph (HPLC) columns. We acquired a Flo-One, Model HP[5] radioactive flow detector on a long-term loan for proof-of-concept testing. The detector element is a thin-wall Teflon[4] tube coiled flat and sandwiched between two matched photomultiplier (PM) tubes. The usable cell volume is 2 ml. We characterized the performance of the unit as follows:

- The filtered waste water background count rate was 12% higher than the background count rate of drinking water. This was due to chemiluminescence and bioluminescence.
- The maximum counting efficiency for tritium in waste water was 10.6% as compared to 22.6% in drinking water. This was due to chemical quenching in the waste water.
- The minimum detectable activity (MDA) for a 20-minute counting period was calculated using the method of Tritium Measurement Techniques[6]. It was 265 pCi/ml (9.8 Bq/ml) with "known" (stable) background and 374 pCi/ml (13.9 Bq/ml) with "unknown" (variable) background.
- Response time for an aliquot of tritiated water containing 14 nCi (518 Bq) was less than 20 minutes to reach the MDA level, and 50 minutes to reach maximum count rate. Since at least four hours pass before LLNL effluent reaches the Livermore Water Reclamation Plant, sufficient time is available to evaluate the simulated emergency, and, if necessary, divert the flow to emergency holding basins at the plant.

FUTURE INVESTIGATIONS

We have begun to construct a second-generation monitor. It will include the following features to improve its sensitivity:

- A flat, spiral-wound coil of thin-walled Teflon[4] tubing sandwiched between the two matched PM tubes, similar to the flow cell developed by Schram and Lombaert.[7] The light-transmission properties of the cell will be improved by filling all voids with water or silicon oil, coupling the cell to the PM tubes with optical grease, and placing reflectors around the perimeter of the cell. Two PM tubes will be used in the coincidence counting mode. The flow cell and PM tubes will be placed in a graded metal shield to further reduce the background count rate.
- An external gamma source will be used for quench correction. The source will be actuated automatically by a simple mechanical system.
- Sachan and Soman have shown it is possible to decontaminate hydrophobic liquid scintillators.[8] We will decontaminate and reuse the collected "waste" scintillator, hence greatly reducing our operating costs.

REFERENCES

1. C. J. Huntzinger, D. E. Hankins, J. L. Cate, Jr., M. Dreicer, and A. A. Odell, "A Real-Time Monitor for Tritium in Waste Water," Hazards Control Department Annual Technology Review 1982, LLNL, UCRL-50007-82 (1983).
2. Ultra Dynamics Corporation, Santa Monica, CA 90404.
3. Digital Equipment Co., Inc., Maynard, MA 10754
4. E.I. du Pont DeNemours & Co, Inc. Plastic Products and Resins Department, Wilmington, DE 19898.
5. Radiomatic Instruments & Chemical Co., Inc., Tampa, FL 33611.
6. National Council on Radiation Protection and Measurements, Tritium Measurement Techniques, NCRP No. 47 (1976).
7. E. Schram and R. Lombaert, Analytical Biochemistry, 3, 68 (A62).
8. S. R. Sachan and S. D. Soman, "Decontamination and Modification of Liquid Scintillators," Proceedings of the 5th International Congress of the IRPA, Jerusalem, Israel (1980).

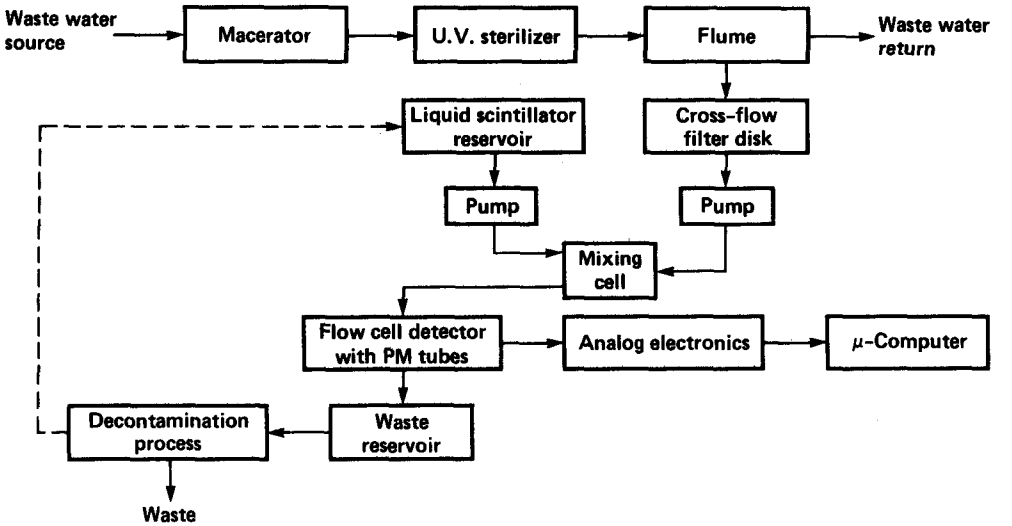

Figure 1. System block diagram.

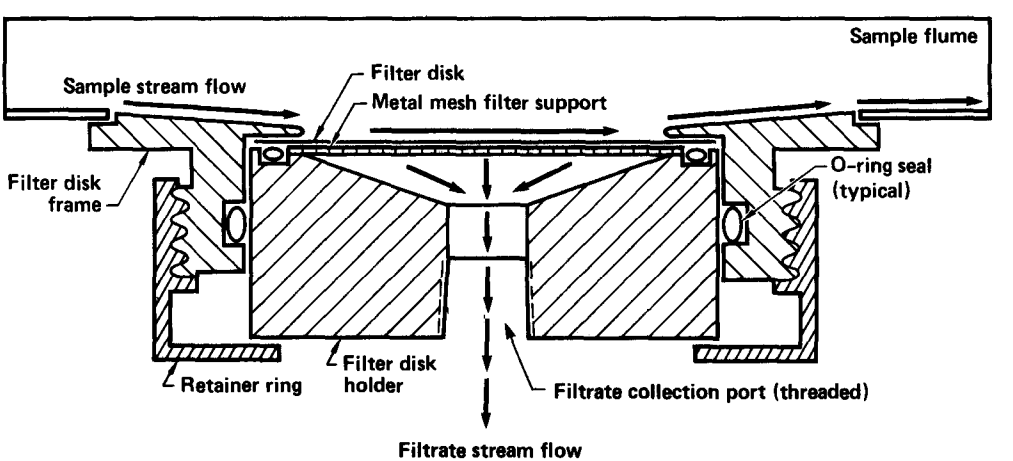

Figure 2. Cut-away view of the cross-flow filter.

TRITIUM-IN-AIR MEASUREMENTS BY PULSE-SHAPE DISCRIMINATION METHODS

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Introduction

The simplest method of monitoring ^{3}H is passing it through an ionisation chamber. However, the sensitivities obtained are far off the values required by regulations in many countries. (Table 1).

Section	Exposure Group	Concentration limit in	
		Bq m^{-3}	Ci/m^3
§ 52 (1) 1	Category A in controlled areas	1.8×10^5	5×10^{-6}
§ 52 (1) 2	Category B in controlled areas	5.9×10^4	1.6×10^{-6}
§ 52 (1) 3	Non-occupational exposure in monitored operating areas	5.9×10^3	1.6×10^{-7}
§ 52 (1) 4	Non-occupational exposure in monitored areas outside plants	3.7×10^2	1×10^{-8}
§ 46 (3)	General national territory (emission concentration)	3.7×10^2	1×10^{-8}

Table 1: Concentration Limits for airborne ^{3}H (German Radiation Protection Regulations)

Furthermore, in ionisation chambers it is very difficult to discriminate the signal due to ^{3}H from the effect of external gamma-radiation and higher energy gaseous radionuclides.

Ehret, Kiefer and Maushart (1) successfully introduced proportional counter tubes with volumes of several l for ^{3}H-monitoring.

The price for obtaining excellent sensitivity, stability and good discrimination against ambient gamma emitters is the necessity to continuously add counting gas to the sample air. As a consequence of the detector design, with separate sample and shield volumes, about 75% of the counting gas/air mixture is always in the shield or dead volume.

One of the objects of the present work was to use the entire detector volume as sample counter, therefore reducing counting gas consumption and/or increase sensitivity (2).

Principle

The pulse rise-time in a proportional counter tube is dependent upon the different drift times of the secondary electrons generated in the ionisation process as they travel to the counting wire. The shorter the range of the beta particle, the smaller is the average fluctuation of the arrival time at the counting wire, thus leading to shorter rise-times. Rise-time discrimination therefore allows to differentiate between radionuclides with different particle range.

Detector Design

Two different detectors are used:
1. Cylindrical design with 2 concentric counting volumes, each with a cylindrical array of counting wires, separated by ground wires, with total volume 3.0 l.
2. Rectangular design with a single plane of counting wires, volume 0.5 l.

Electronics

All counting wires are operated in parallel. Pulses are amplified by a fast (rise-time approx. 10 ns) charge-sensitive preamplifier followed by a voltage amplifier, then passed through a double delay-line pulse shaping network with zero-crossing discriminator. This allows to obtain pulses the width of which is a measure of the pulse-rise time. A pulse-width discriminator sorts all pulses into two channels (see also Fig. 2), channel A representing tritium-particles, and channel B representing background, gamma-radiation or radioactive noble gases.

Operation

For better understanding of the nature of the process, the pulses of the detector have been fed into a rise-time/amplitude converter, followed by a standard multichannel pulse-height analyzer.

Fig. 1 shows the rise-time spectra for a relatively high concentration of ^{3}H (16000 cpm), so that background contribution (1050 cpm) is hardly noticeable. A relatively narrow peak is clearly recognized.

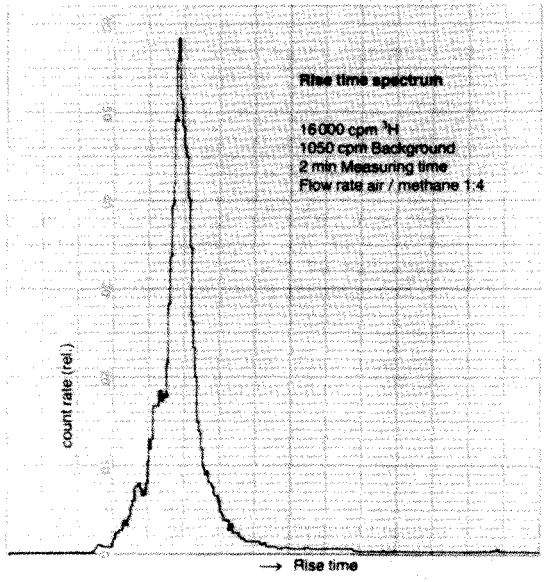

Fig. 1 Rise-time spectrum for ^{3}H in air/methane mixture.

Fig. 2 shows corresponding results for a smaller ^{3}H-concentration (300 cpm). The background contribution with its much longer rise-times is now clearly recognizeable.

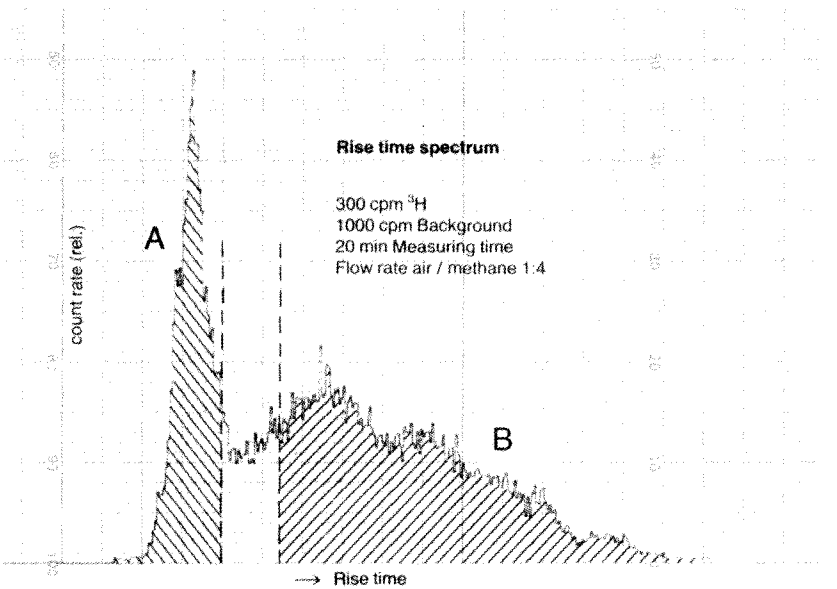

Fig. 2 Rise-time spectrum for ^{3}H and background (see text).

Rise-time scales have not been calibrated absolutely since the method is based upon the difference of relative rise-times only. For ^{3}H, rise-times for a methane/air mixture are below 20 ns, while the majority of background pulses shows rise-times around 100 ns. Using P10 gas and air, rise-times appear to be at least 5 times larger, for both tritium and background pulses.

Fig. 2 also explains the function of the rise-time discriminator. Channel A is set to enclose essentially the rise-time "peak" caused by ^{3}H-particles. Channel B covers the broad spectrum of long-range ^{3}H particles. It can be used to measure radioactive noble gases independent from ^{3}H, or to establish a monitoring channel for either noble gases or ambient radiation, in order to establish a spill-over factor into the ^{3}H-channel.

Results

Table 2 shows operating data and results, while table 3 shows detection limits as a function of measuring time.

Detector Volume	3.0 l	0.5 l	
Flow rate air/methane	1/3	0.25/0.75	l min^{-1}
Air sample volume inside detector	750	125	ml
Efficiency for ^{3}H	60	50	%
Background rate	0.6-1.5	0.2-0.4	s^{-1}
Spill-over (^{137}Cs gamma radiation → ^{3}H-channel)	4-5	4-5	%

Table 2 Results with 3.0 and 0.5 l detectors

	3 l - Detector		0.5 l - Detector	
Measuring Time	kBq·m^{-3}	Ci/m^{3}	kBq·m^{-3}	Ci/m^{3}
30 s	0.920	2.5×10^{-8}	4.8	1.3×10^{-7}
1 min	0.650	1.75×10^{-8}	3.4	9.2×10^{-8}
10 min	0.205	5.5×10^{-9}	1.1	1.0×10^{-8}
1 h	0.084	2.3×10^{-9}	0.44	1.2×10^{-8}
24 h	0.017	0.46×10^{-9}	0.09	2.4×10^{-9}

Table 3 Detection limits (3σ error) as function of measuring time. Values given apply for operation with methane, for P10 the detection limits are higher by a factor of 2.2.

Literature

1. Ehret, Kiefer und Maushart, "Fortschritte bei der kontinuierlichen Tritiumüberwachung in Luft", Direct-Information 2/63, G. Braun-Verlag, Karlsruhe
R. Maushart, "Tritiummessungen im Strahlenschutz", G-I-T p. 845 (1967).

2. German Patent No. 2500 510, "Verfahren zur Selektierung der Kernstrahlung bestimmter gasförmiger, in einem Trägergas in ein Zählrohr gebrachter Radionuklide, unter Diskriminierung nach der Impulsform sowie Anwendung dieses Verfahrens und Verwendung eines Durchflußzählrohres hierzu."

ALPHA PARTICLE ANALYSIS USING PEARLS SPECTROMETRY

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ABSTRACT

Alpha particle assay by conventional plate-counting methods is difficult because chemical separation, tracer techniques, and/or self-absorption losses in the final sample may cause either non-reproducible results or create unacceptable errors. PEARLS (Photon-Electron Rejecting Alpha Liquid Scintillation) Spectrometry is an attractive alternative since radionuclides may be extracted into a scintillator in which there would be no self-absorption or geometry problems and in which up to 100% chemical recovery and counting efficiency is possible. Sample preparation may include extraction of the alpha emitter of interest by a specific organic-phase-soluble compound directly into the liquid scintillator. Detection electronics use energy and pulse-shape discrimination to provide discrete alpha spectra and virtual absence of beta and gamma backgrounds. Backgrounds on the order of 0.01 cpm are readily achievable. Accuracy and reproducibility are typically in the 100 ± 1% range. Specific procedures have been developed for gross alpha, uranium, plutonium, thorium, and polonium assay. This paper will review liquid scintillation alpha counting methods and reference some of the specific applications.

LIQUID SCINTILLATION ALPHA SPECTROMETRY

Liquid scintillation has been recognized as an attractive method for counting alpha emitting radionuclides from the initial development of the method.[1,2] A common approach has been to use the readily available commercial liquid scintillation equipment employed for low-energy particle counting. The technique is useful and accurate for samples of known nuclide purity and known uniform matrix composition, and can provide backgrounds on the order of 10 to 20 cpm. Samples of unknown composition frequently exhibit variable quenching and backgrounds. The presence of beta and gamma radiation will increase backgrounds and may interfere with alpha detection. Because alpha particles produce high specific ionization in the solvent, they have only about 10% light production efficiency per MeV when compared with beta and gamma radiation. Consequently, alpha emitters in the 5MeV range create electronic signals similar to those generated by betas with energies on the order of 0.5Mev. Energy resolution is on the order of 20%.

A significant improvement in energy resolution may be obtained if the detection system is designed exclusively for alpha counting. When the nuclide of interest is extracted into the liquid scintillator extractant solution and counted using the detection arrangement outlined in the figure and standard electronics of the type used for sodium iodide-gamma spectrometry, reproducible spectra with an energy resolution on the order of 5% are possible. Beta and gamma decay events may be segregated from the alpha emissions by electronically processing the differences in pulse shape, a beneficial aspect of the large differences in specific ionization. To achieve effective pulse shape discrimination (PSD), samples must be deoxygenated. This is easily accomplished by sparging the scintillator/sample for 1 or 2 minutes with argon or nitrogen gas. Color or chemical quenching will also degrade PSD.

PEARLS (Photon-Electron Rejecting Alpha Liquid Scintillation) spectrometry maximizes the advantages of liquid scintillation detection by coupling solvent extraction methods with detectors and electronics having the ability for improved energy resolution and for pulse shape discrimination between alpha-produced and beta/gamma-produced pulses. Electronic components are connected as shown in the fiqure. The PSD information is used to gate the multichannel analyzer and remove the beta/gamma background interference from the alpha spectra. Representative spectrum for a radon experiment is included in the figure. Backgrounds are effectively reduced to as low as 0.01 cpm while retaining 100% counting efficiency.

Complete details of alpha detection, chemical extraction processes and PEARLS spectrometry are being prepared for publication in a monograph.[3]

There appear to be many potential applications for high resolution liquid scintillation techniques. To date the PEARLS detection concept has been coupled with solvent extraction separation techniques and applied to assay of uranium and thorium in phosphate fertilizers,[4,5] analysis of polonium in a uranium mill circuit[6,7] and determination of plutonium and transplutonium actinides in a variety of sample types.[6] A portable PEARLS spectrometer has been developed for field measurements.[8]

CONCLUSIONS

The application of liquid scintillation detection may be readily extended to assay alpha radionuclides in numerous sample types. The accuracy and reproducibility of the method is almost always better than with plate-counting methods. Commercially available beta liquid scintillation systems may be used, but the potential problems associated with interference from beta and gamma radiation must be recognized. Commercial systems may not be easily adapted to perform PSD. The use of equipment designed for alpha liquid scintillation, along with appropriate PSD electronics and extractive scintillators, will permit the collection of useful energy spectra, the rejection of nearly all beta and gamma background, and in most cases, improve the accuracy of the determination. Sample concentration by solvent extraction, 100% efficiency, and low virtual background may result in detection thresholds which are lower than in other methods. The opportunity to provide an alpha detection technique of improved accuracy coupled with increasing demands for rapid, accurate low-level radiation monitoring in a variety of applications provide an incentive to continue development of this method.

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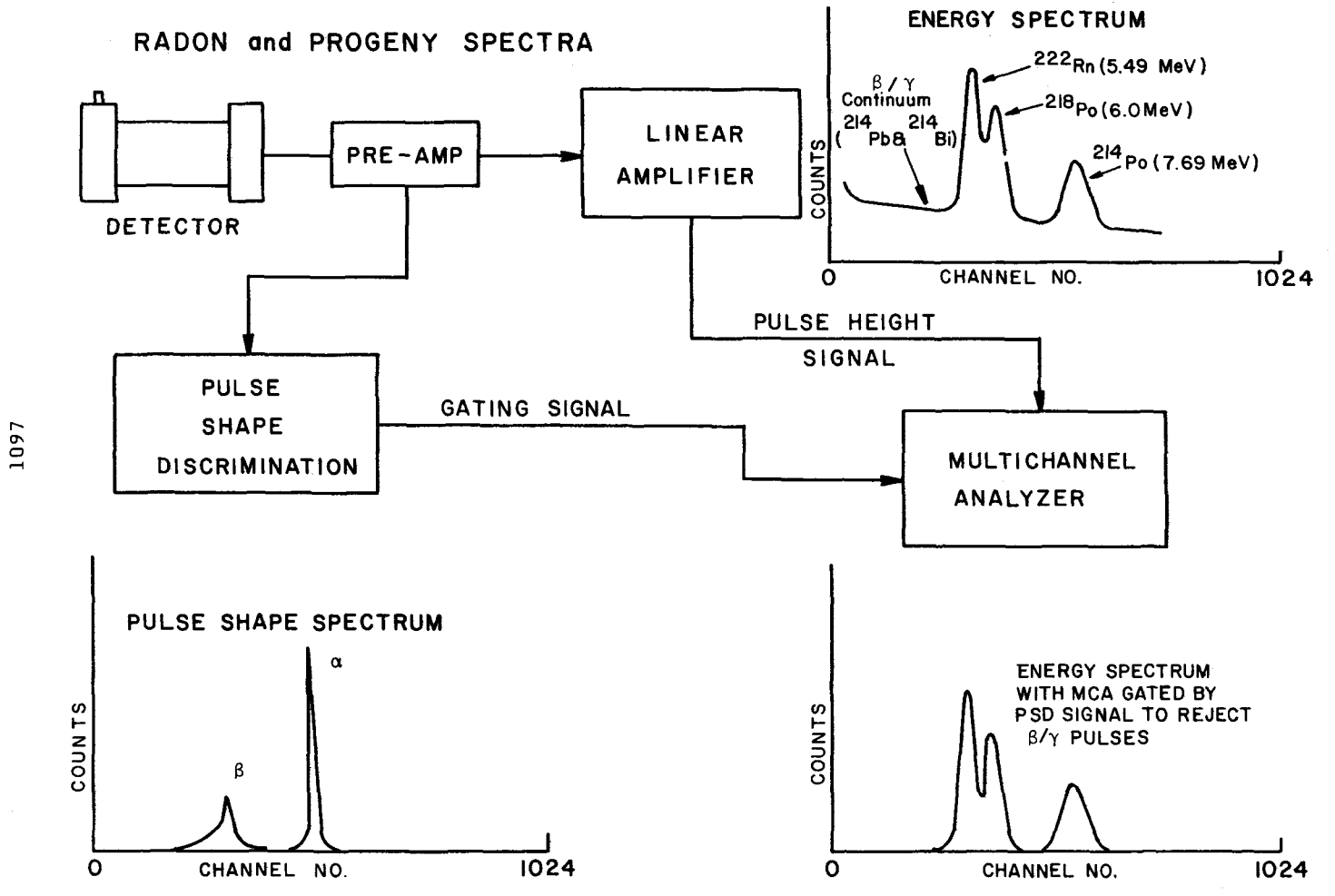
RADON and PROGENY SPECTRA
DETECTOR
PRE-AMP
LINEAR AMPLIFIER
ENERGY SPECTRUM
β / γ Continuum
214 Pb
214 Bi
222Rn (5.49 MeV)
218Po (6.0 MeV)
214Po (7.69 MeV)
COUNTS
0
CHANNEL NO.
1024
PULSE HEIGHT SIGNAL
PULSE SHAPE DISCRIMINATION
GATING SIGNAL
MULTICHANNEL ANALYZER
PULSE SHAPE SPECTRUM
α
β
ENERGY SPECTRUM WITH MCA GATED BY PSD SIGNAL TO REJECT β/γ PULSES

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REFERENCES

1. M. Ageno, M. Chiozzofto, and R. Querzoli, Phys.Rev., 79,720 (1950)

2. J. Basson and J. Steyn, Proc. Phys. Soc., London Sect. A, 67, 297 (1954)

3. W. J. McDowell, "Alpha Counting and Spectrometry Using Liquid Scintillation Methods," National Academy of Science National Research Council, Nuclear Science Series, to be published in 1983 by the Technical Information Center, U. S. Department of Energy.

4. E. J. Bouwer, J. W. McKlveen, and W. J. McDowell, "A Solvent Extraction-Liquid Scintillation Method for Assay of Uranium and Thorium in Phosphate-Containing Material," Nucl. Technol., 42, 102-111 (1979).

5. R. Metzger, J. W. McKlveen, R. Jenkins, and W. J. McDowell, "Specific Activity of Uranium and Thorium in Marketable Rock Phosphate as a Function of Particle Size," Health Physics, 39, 69-75 (1980).

6. G. N. Case and W. J. McDowell, Talanta, 29, 845-848 (1982).

7. J. W. McKlveen, N. Hubele, W. J. McDowell, and G. N. Case, "Po-210 Distribution in Uranium Mill Circuits," Trans. Amer. Nucl. Soc., ANS Winter Meeting, San Francisco 1983 to be published.

8 G. W. Klingler, J. W. McKlveen, and W. J. McDowell, "A Portable Photon-Electron Rejection Alpha Liquid Scintillation Spectrometer," Trans. Amer. Nucl. Soc., 39, 90-91 (1981).

DETERMINATION OF THE BETA ENERGY (E_{max}) USING THIN WINDOW INSTRUMENTS*

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The use of simple survey instruments for beta-energy analysis is complicated by large differences that exist in the beta spectra shapes. These spectral shapes are often complex and change continuously as the betas are absorbed in air. Changes are also caused by absorbing material between the source and the detector. One may frequently encounter a combination of beta energies, either from multiple emissions from a single isotope or from several isotopes in the sample being evaluated. There may also be monoenergetic conversion electrons present in the sample or low-energy X rays which are absorbed in a similar fashion to betas.

Obviously, a complete analysis of complex beta spectra cannot be performed using only survey instruments. We present two methods which will give the approximate E_{max} of the beta energy responsible for the most significant portion of the beta dose. Either technique should give adequate information about the beta spectra to provide necessary guidance for the health physics evaluation of the exposure.

Theory

If simple survey instruments are going to be used to determine the beta energy, the most reasonable approach would be to make absorption studies, but limit them to as few absorbers as possible. Three absorber thicknesses were used in this study.

Mantel published the calculated beta spectra shape for 59 beta emitters (Ma72). The absorption curve obtained from each spectrum depends upon its shape. We calculated the absorption curve for many of the spectra given by Mantel. When the shapes of the absorption curves were compared, significant differences were apparent. If analysis of E_{max} is to be based on absorption measurements (using the three filter thicknesses we chose earlier), the absorption curves have to be similar. Based on the shapes of the absorption curves obtained above, we divided the beta spectra given by Mantel into four spectra types. These spectra types are shown in Fig. 1.

Few beta spectra look exactly like the ones labeled as types 1 through 4 and we show the variation that occurs within each spectra. There is obviously an overlap region between types; for example, curves F and G are very nearly the same. By using the analysis technique we describe, the spectra type obtained for spectra that have shapes in this region could be identified as either Type 1 or Type 2 depending on the variations in count rate or instrument reading recorded for that source.

The Type 4 sources shown as Q, R and S are sources having two major beta energies being emitted by a single isotope or from two isotopes in a decay chain such as ^{90}Sr and ^{90}Y. Another kind of the Type 4 source category are those shown in curves T, U and V, where there is only one predominant beta. Using the technique described later, these spectra would be analyzed as having only the one predominant energy and therefore, the energy obtained experimentally may be different than the published E_{max}.

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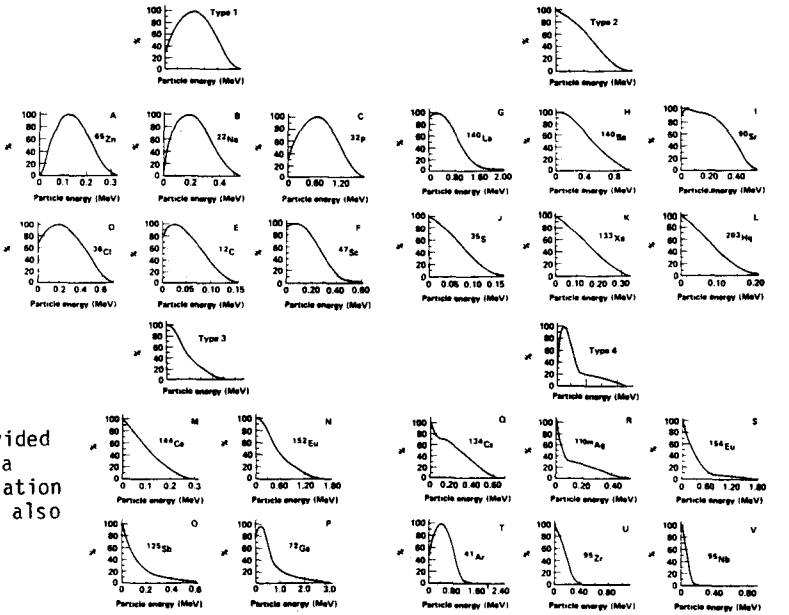

Fig. 1 The beta spectra were divided into four spectra types. The variation within a type is also shown.

A comparison of shapes of the absorption curves in the first decade shows that most of the Type 2 spectra are essentially straight lines, but the Type 1 and 3 spectra are curved with the Type 1 curves above and the Type 3 below the straight line curves of Type 2. The Type 4 sources have absorption curves (not shown) which have inflections in the curves and are obviously from two (or more) betas. The curves have a definite bend at the point where the first beta is absorbed. The absorption curves calculated for all the isotopes are different and we decided to use a typical or average curve for each source type. In Fig. 2 we show the average absorption curves used for the first three spectral types. These curves were used to develop additional curves, shown later, which were used to obtain the beta energy. The use of average curves will result in an error in the E_{max} evaluated for a specific beta depending upon how well the actual and average curves compare. This error is apparently small based on the good experimental results we obtained.

Experimental Method

To cover a wide range of dose rates, we used the Eberline Instrument Corporation Model RO-7 survey instrument and the Eberline Model E-120, GM equipped with the HP-210 pancake probe. The probe of the RO-7 has a window thickness of 7 mg/cm^2 and the window of the pancake probe varies from 1.5 to 2.0 mg/cm^2. The absorbers we used to distinguish between betas and gamma rays consisted of acrylic (Lucite). We used the ~1 cm thick Lucite "beta filter" shown in Fig. 3 provided by Eberline with the RO-7 instrument. The pancake probe was used with two 0.48 cm-thick disks with an 8.8 cm radius (the o.d. of the probe housing), giving us a nominal 1-cm thick beta filter equivalent to the RO-7 beta filter. The readings of the instruments without the Lucite filters was assumed to be beta plus gamma and the filtered readings were the gamma readings. To determine the beta energies, aluminum absorbers of three thicknesses, 6.42, 33.4 and 277 mg/cm^2, were prepared for both instruments (Fig. 3).

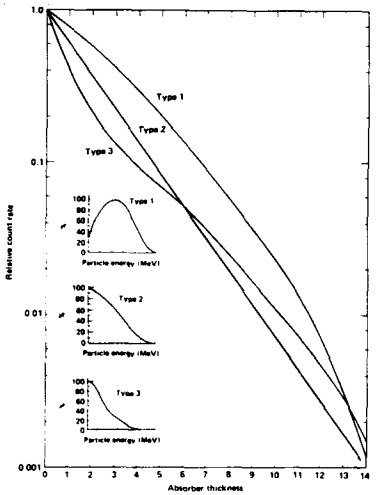

Fig. 2 Typical or averaged curves for each spectra type.

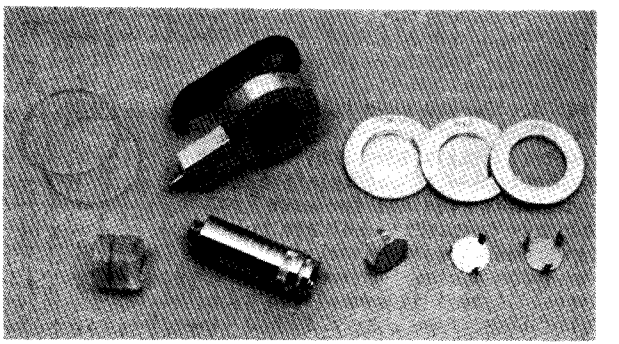

Fig. 3 Eberline HP-210 pancake probe and the Eberline R07BM. The Lucite beta shields are on the left and the aluminum absorbers are on the right.

To determine the beta energy and spectral type of a source, five measurements of the dose rate or counting rates are required. These readings are made with the bare probe, Lucite-covered probe, and with each of the three aluminum absorbers.

The X- and gamma-ray component of the instrument readings must be removed first by subtracting the reading of the Lucite-covered probe from each of the other readings. Then, the ratio (or relative readings) of the beta readings with each aluminum absorber and the readings without absorber is determined. This is done by dividing the beta reading of the instrument with the aluminum absorbers by the beta reading of the instrument without absorber. The relative readings for each of the three absorbers is then plotted on the solid lines in Fig. 4 which was derived from the curves shown in Fig. 2.

If the source is a Type 1 source, the points plotted on the solid lines in Fig. 4 will be at the same beta energy. (If the relative reading is >0.95 or <0.05, that piece of data is probably not useful and does not need to be plotted.) If the data points do not give the same energy on the solid line, the source is not a Type 1 source. Then, in order to determine the type source, the points are plotted on the dashed curve for Type 2 sources and the dot/dash curve for Type 3 sources. The type of line on which the data points give the same energy (align vertically) is the type of source being evaluated and the energy where alignment occurs is the E_{max} of the predominant beta in the spectrum.

If the spectra is a Type 4 source, alignment will not occur. The results with the thin absorber indicate a low energy and the medium and thick absorbers indicate progressively higher energies. The fact that this is a Type 4 source can be useful in health physics work. One can use this technique to evaluate the higher energy component of Type 4 sources by removing the lower energy component with an appropriate absorber, (see complete text for details, HA82.

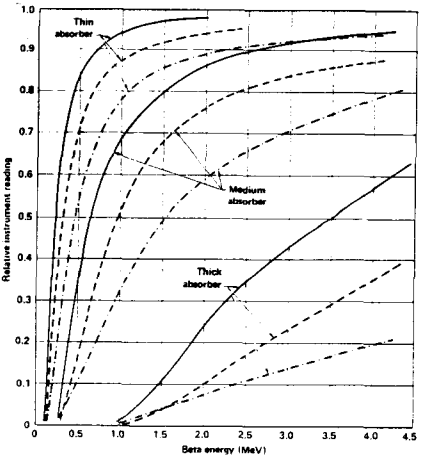

Fig. 4. Curves used to determine the energy and spectral type of beta sources. The beta energy and type is obtained when the relative instrument readings for each absorber thickness indicate the same energy (align vertically) on lines for one of the spectral types. Type 1 is symbolized by a solid line; Type 2 by a dashed line; and Type 3 by a dot/dash line.

The results obtained with numerous sources of all four types (see HA82) indicate that the known and evaluated source types and energies agree reasonably well and are adequate for health physics work. In most cases where differences were found, they could be explained by source shielding or other factors.

Field Surveys Using a Single Absorber

One goal of our study was to develop a technique where a single absorber thickness could be used to evaluate the exposure problems caused by beta particles in health physics field work. The field procedure consists of three dose-rate readings: (1) the bare probe, (2) the beta (Lucite) shielded probe and (3) the medium absorber reading. The reading with the beta shield (x and gamma rays) is subtracted from both the bare probe reading to obtain the beta dose rate and from the reading with the absorber. The beta dose rate and dose rate with absorber are compared and the amount of decrease caused by the absorber is determined. If it is small <25% the beta energy is high and is assumed to be >1.5 MeV. If the decrease is large, >75%, the beta energies are low <0.5 MeV. Readings between 25 and 75% would indicate beta energies between 0.5 and 1.5 MeV. Analysis of the beta energy using the single-filter technique gives a good estimate of the actual energy for the sources. However, its accuracy in a field location would depend upon the particular isotope or combination of several isotopes being evaluated, as well as the errors usually associated with field work. The technique is easy to use and a health physics technician making field surveys should be able to obtain the information necessary to evaluate protective clothing and personnel dosimetry requirements.

Ha82 Hankins, D.E., 1982, Evaluation of Beta Energy E_{max} and Spectral Type Using Survey Instruments, Lawrence Livermore National Laboratory, Livermore, CA UCRL-88275 [Submitted to Health Physics].

Ma72 Mantel J., 1972, "The Beta-Ray Spectrum and the Average Beta Energy of Several Isotopes of Interest in Medicine and Biology," International Journal of Applied Radiation and Isotopes 23, 407.

A PORTABLE CALORIMETER FOR ABSORBED DOSE DETERMINATION FROM HIGH ENERGY RADIATION

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INTRODUCTION

In the past the physical quantity exposure and its unit "Röntgen" were most widely used in radiation dosimetry. However, this quantity is only defined for photons up to several MeV and therefore not applicable for high energy radiation increasingly used today in radiation therapy. A further inherent problem is that the unit "Röntgen" is not compatible with the SI units. In order to overcome this difficulties the physical quantity "absorbed dose" (D) has widely replaced exposure and a new special name for the unit of absorbed dose the "gray" (Gy) has been introduced internationally(1).

This unit is used almost exclusively already in radiation therapy. In radiation protection the physical quantity dose equivalent with the new special name Sievert (Sv) is applied which is also based on the measurement of absorbed dose.

The portable graphite calorimeter described in this paper is a primary standard for the determination of absorbed dose. The energy from radiation absorbed is heating the graphite and rises its temperature. This temperature rise is compared to that produced by a well known amount of electrical energy released by an internal heater. The basic problem with the construction and use of an absorbed dose calorimeter lies in the fact that a temperature rise of 14 µK has to be measured to determine a dose of 1 cGy (1 rad).

CALORIMETER CONSTRUCTION

The construction of the calorimeter is based on the design by DOMEN at the NBS Washington(2). It consists of the following cylindrical graphite bodies:

1. The "phantom", the outermost part of the calorimeter (300 mm diameter, 100 mm thick), is supported by a wooden frame and not thermoregulated. Additional graphite plates can be mounted in front to determine depth dose distribution. All the other bodies are situated in a lucite vacuum chamber.

2. The "medium" is a high precision thermoregulated graphite cylinder enclosed in the vacuum chamber. The front of the medium is covered by an aluminized mylar window of 0,13 mm thickness. The working temperature of the medium is approximately 27°C. The short-term stability of the temperature is about 5.10^{-4}K, the long-term stability about 5.10^{-3}K.

3. The "shield" is mounted in the central hole of the medium. It is thermally isolated against the medium by vacuum gaps and coatings of aluminized mylar, 10 µm thick. It acts as a thermal buffer between the medium and the graphite bodies described below.

4. The "jacket" is thermally isolated against the shield by gaps and coatings as well. It has the shape of a cylindrical box and encloses the "core". It further reduces the temperature gradients.

5. The "core" is the actual point of measurement for the determination of absorbed dose. Although it is isolated against the jacket by vacuum gaps of 0,5 mm, it is not coated in order to minimize the presence of materials other than graphite. Fig. 1 is a photograph of the complete calorimter, Fig.2 shows a cross section of the basic calorimeter components. The short term temperature stability of the core is within $\pm$ 1 µK.

A spiral heater is embedded in the back of the medium. It is controlled automatically by a precision thermoregulator. The shield, jacket and core are equipped with electrical heaters which are operated manually.

The temperature in all graphite bodies except the phantom is measured by microthermistors with 0,3 mm diameter. Their nominal resistance at 25°C is 10 kΩ. The thermistors are switched into the arms of a Wheatstone bride (Fig. 3). Each thermistor is balanced by a variable resistor (R_M, R_S, R_J, R_{C1}, R_{C2}). The first arm in the bride (1) consists of two fixed 10 kΩ resistors and is used as a reference. The bridge is powered by an AC source (0,2 - 1 Vrms, 2 - 10 Hz) which also controls the sensitivity of a lock-in amplifier with respect to frequency and phase. The lock-in amplifier can be switched between each two points (1-6) in the bride circuitry as desired. By this way the temperature can be observed in each body.

The core contains two measuring thermistors (C_1, C_2) in order to double the sensitivity for low dose rate measurements. In this case the amplifier is switched between point 5 and 6. In order to obtain a heat-loss compensated calibration[2] it's also possible to measure the temperature rise in the core plus jacket (amplifier between 4 and 6). The amplifier output voltage is recorded on a strip chart recorder.

MEASUREMENTS AND RESULTS

The basic state of the calorimeter for any measurement is a temporary temperature equilibrium. In this state the power input by the measuring thermistors equals the heat-loss rate of each body. In practice precisely defined energy inputs into the bodies by the different heaters can shorten the time to reach temporary equilibrium considerably. The temperature drift rate of each body is a measure for the deviation of its temperature from the equilibrium.

The calibration of the instrument can be performed either in the heat-loss compensated mode or in a quasi-adiabatic mode achieved by input of an exactly measured amount of energy in the core heater. Radiation measurements are usually made quasi-adiabatically. The calorimeter response for a calibration- or irradiation-run is evaluated by determining the relative resistance change of the core thermistors. In order to be independent on the amplifier gain the bridge circuit is balanced

before and after electrical heating or irradiation to zero bridge output. This can be obtained by changing balancing resistors by a well known amount.

The calibration of the instrument was performed at an electrical power level equivalent to about 60 $mGy.s^{-1}$. The calorimeter calibration factor has be determined as - 140 mJ per percent resistance change (nominal) with a standard deviation of a single run of less than 0,1 %. The long-term stability of this calibration factor is better than 10^{-3}.

The quasi-adiabatic radiation measurements were made at a doserate of about 5 $mGy.s^{-1}$ in a ^{60}Co beam. The irradiation time was 4 min and the absorbed dose of each run ($\approx$1,2 Gy) could be determined with the standard deviation of a single run better than 0,3 %. On the chart recorder plot of a typical radiation run one scale division is corresponding to a temperature rise of 14 µK equivalent to a bridge output voltage of 100 nV and a dose of 10mGy.

CONCLUSION

The portable graphite calorimeter described proved to be a suitable primary standard for the determination of absorbed dose. The range of measurement extends from about 1 $mGy.s^{-1}$ up to the very high doserates produced by betatrons and linear accelerator applied in radiation therapy. Due to the application of a new high precision thermoregulator, an improved electronic measuring circuitry with AC-Wheatstone bridge and lock-in-amplifier a significant improvement in the limit of detection down to the range of protection level doses could be achieved.

REFERENCES

(1) ICRU Report 33 Radiation quantities and units: ICRU Publications, Washington DC, 1980.

(2) DOMEN, St.R., LAMPERTI, P.J.: A heat loss compensated calorimeter: theory, design and performance.
J.Res.Nat.Bur.Stand. (US) - A Vol. 78 A No. 5, 595-610, 1974 (Sept. - Oct.)

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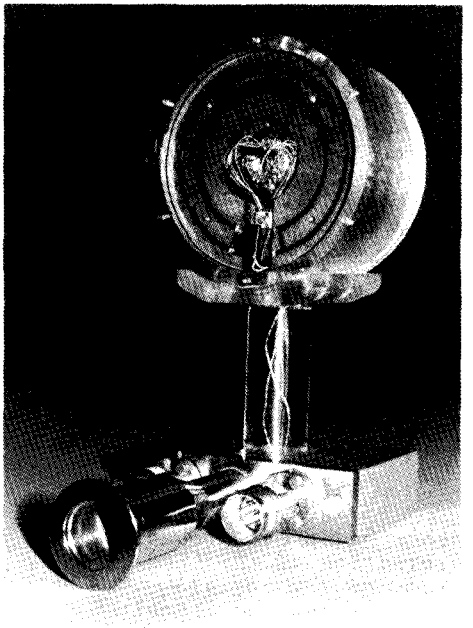


Fig. 1: View of Calorimeter Assembly

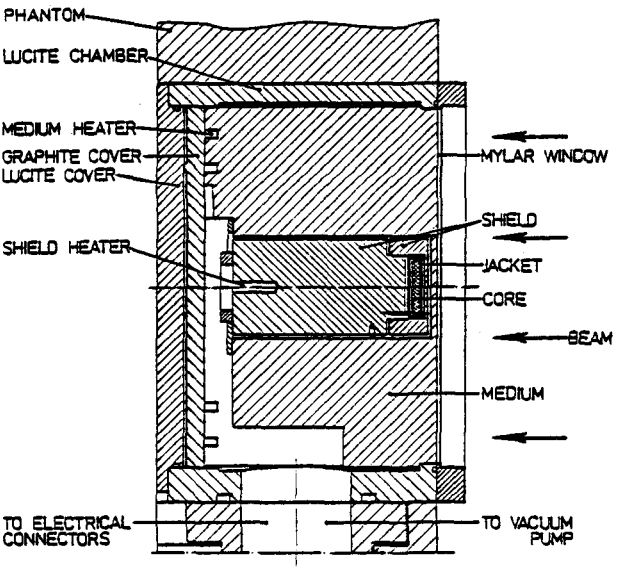
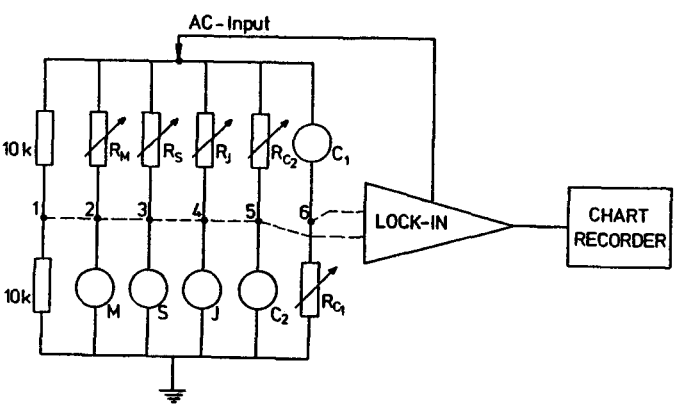


Fig. 2: Cross Section



$R_M, R_S, R_J, R_{C_2}, R_{C_1}$ ..... Balance Resistors
M, S, J, C_2, C_1 ..... Microthermistors

Fig. 3: Schematic Circuitry of Wheatstone-Bridge

THE CERN RADIATION MONITOR AND ALARM SYSTEM FOR EXPERIMENTAL AREAS

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CERN, Geneva

1. INTRODUCTION

At CERN beams of protons, pions and other short-lived particles having energies up to several hundred GeV are made to interact in targets of liquid hydrogen, copper, etc., in order to study the elementary particles of which all matter is composed. Experimental areas can contain many such beams; if particle intensities are below some 10^6 s^{-1}, then the beams are allowed to run in open-topped zones bounded often by only a meshed fence. Close to collimators and around beams whose intensity is greater than 10^7 s^{-1} concrete and steel are normally used to shield the targets and beam lines. These beams operate independent of each other, so that in a zone adjacent to or downstream of an operating beam there could be persons installing the apparatus for a separate experiment. The geography of a typical experimental area is illustrated schematically in Fig. 1. Inside the area there are zones of "permanent" occupation where physicists control the functioning and data-taking of their experiments. In addition there are many corridors and equipment-handling areas. This variety in the utilization of an experimental area imposes great versatility on an installed radiation monitor and alarm system. Full details of the monitor and alarm system are given elsewhere (Ref. 1); only a brief description is given here.

The layout and shielding of the experimental zones in an area are designed on the assumption that the radiation levels in critical or occupied areas will be continuously monitored and that alarm information is readily available in the area control rooms. The composition of the radiation field in an area is a function of position inside the area and the different possible beam operations. The three most important fields found are :

a) that due to a "direct view" of a target or beam interaction point in a sideways direction, normally through a relatively thick hadron shield (points marked A in Fig. 1); this field is dominated by the hadron cascade generated in the shield and contains protons, neutrons and pions having energies of up to several GeV;

b) that downstream of a primary beam interaction point or a secondary beam line where muons of energies greater than 10 GeV form the dominant component (points marked B in Fig. 1);

c) that dominated by $\leq$10 MeV neutrons which can be found outside the openings in a thick shield for access ways, cable ducts, etc. (points marked C in Fig. 1). This field also occurs at larger distances from shields where the dominant radiation component arrives at the point of interest by scattering and diffusion in the air or at the ground-air interface.

2. THE MONITORING FUNCTION

The radiation detectors and associated electronics have been selected according to the physical nature of the field. The time structure of the field (often in bursts of several microseconds duration with inter-burst intervals of up to 15 seconds) excludes the use of pulse-counting systems (GM tubes, BF_3 counters, etc.) with their finite resolution time. All detectors are ionization chambers. Argon-filled, high-pressure (5 litre, 20 atm) ionization chambers are provided for monitoring the dose-equivalent rates from photons, muons and other charged particles. Similar hydrogen-filled chambers are preferable where hadron fields dominate and the ionization chamber version of the Andersson-Braun counter is used for low-energy neutron monitoring.

The electronics associated with these ionization chambers is of the charge-digitizing type now in common use for low-level current measurement. The input current from the chamber is applied to a low-leakage solid-state integrator which, upon reaching a given threshold voltage, triggers a charge-pump circuit causing it to subtract from the integrator input a known amount of charge, i.e. to inject a fixed charge of opposite sign to that of the chamber current. This charge can be adjusted by regulating the length of the trigger pulse so that it corresponds to a defined amount of dose-equivalent, referred to a specific calibration condition, e.g. exposure to an Am-Be neutron spectrum. Consequently a defined quantity of radiation must be incident on the chamber before the threshold is again passed and the charge injection re-occurs. By counting the number of re-injections that take place in a simple scaler system and by using a long time-constant ratemeter both integrated dose-equivalent and dose-equivalent rate can be monitored. The digitizer control circuitry ensures that in the event of very high radiation levels a continuous pulse train is emitted and not a DC level (Ref. 2).

Pulses from the digitizers are transmitted along multicore cables to a central control point located in or near the experimental area. A microprocessor-based Digital Data Logger, DDL (Ref. 3,4), accepts pulses from up to sixteen monitors, multiplies the value of the pulse by a field-quality factor (determined from more detailed field measurements using different detectors), accumulates the dose-equivalent readings for periods of one hour, determines the dose-equivalent rate averaged over a period of about 100 seconds, acts as the interface between the monitor and the Radiation Protection (RP) computer where the main data base is kept, and provides signals to alarm matrices when preset dose-equivalent rates are exceeded. This alarm function will be described in more detail below. Direct interrogation of the monitors via the DDL by an RP surveyor is also possible.

A dedicated RP computer provides the link between the user and some 30 DDLs distributed around the CERN sites. Its data base contains, for each monitor, the accumulated dose-equivalent for the last 72 hours on an hourly basis, for the last 30 days as dose-equivalent per day, and for the last 55 weeks as dose-equivalent per week. Graphical displays of these data are available in the RP building and in Accelerator Main Control Rooms; simpler displays of the same data can be obtained on terminals in any experimental area through the CERN control-computer network. Special programs can be run to monitor the dose-equivalent rate at intervals of 120 seconds for specified monitors. It is also possible to ask for the dose rate of any monitor at any time. In this way effective control of radiation levels can be maintained over the many experimental areas at CERN.

3. THE RADIATION ALARM SYSTEM

Each experimental area can be divided into up to some 30 alarm-display areas. A radiation alarm is given locally by flashing lights indicating that either persons should leave the area, or immediate efforts should be made to reduce the radiation level by improved beam steering. There is also an audible signal. This warning (A-alarm) simply follows the radiation level read by one or several monitors guarding a display area. When the level falls, the alarm disappears automatically. The link between the monitors and the areas is made via a matrix connected to the alarm outputs from the DDL. An additional complication is given by the fact that the dose-equivalent rate in a beam line zone is normally above the alarm level during operation of the respective beam. During operation these zones are cleared of personnel before beam operation is allowed. An audible alarm in these circumstances would be annoying and unnecessary. Thus a signal is obtained from the access control system for the zone in question which is used to suppress the sonalert in that display area. The same displays are activated whenever beam is authorized in a beam-line zone. Any A-alarms occurring in

occupied areas are also displayed in the control rooms governing the particular experimental area.

The interconnection between the monitors and alarm displays is illustrated in Fig. 2. Also shown is a second matrix connected to another, higher alarm level (the B-alarm). This matrix attempts to associate an unwanted, high radiation level with a particular beam in an area. The alarms from these monitors are latched and coupled with a "good working-conditions" signal from the monitor before being sent to the matrix. The output from the matrix is an executive rather than warning signal and is sent only to the area control room. There it is used to cut off the offending beam, using the same interlocked beam-line magnets that are used to make a zone "safe-for-access". These alarms can only be cleared from the area control room by resetting the latches of the individual monitors causing the alarms; this will not succeed if the radiation level is still high! It is thus possible to diagnose mistakes caused by including a monitor in the wrong line of the matrix.

4. CONCLUSIONS

The radiation monitor and alarm system described here was first installed in an experimental area of the 450 GeV proton accelerator at CERN, the SPS, in 1978. Since then identical systems have been installed in the remaining six experimental areas of the CERN accelerators. These systems have proved extremely reliable in operation, the computer surveillance of the "good working-conditions" signal from the DDL allowing quick identification of a monitor fault. The ease of interrogation of the monitors and alarms via the computer system is much appreciated by the users. However, it must be remembered that the computer is mainly a bookkeeper and the alarm function is ensured by hard-wired signals from the DDL. The "firm" rather than "hard" logic provided by the DDL is acceptable in the CERN experimental areas because the design philosophy of the areas must make it impossible for lethal or dangerously high (>0.1 Sv/h) radiation levels to occur in an occupied area under any condition.

REFERENCES

1. B. Moy, G. Rau, A. Shave and G.R. Stevenson, The CERN radiation monitor and alarm system for experimental areas, HS Division Report CERN HS-RP/053 (1980).

2. R. Shafer, Comments on current-digitizer design for ion chambers, LAB-II Internal Report, CERN LabII-RA/Note/73-12 (1973).

3. T.M. Napier, A digital data logger, CERN 78-15 (1978).

4. B. Moy, G. Rau and A. Shave, The data acquisition system for radiation measurements, HS Division Report CERN HS-RP/058 (1981).

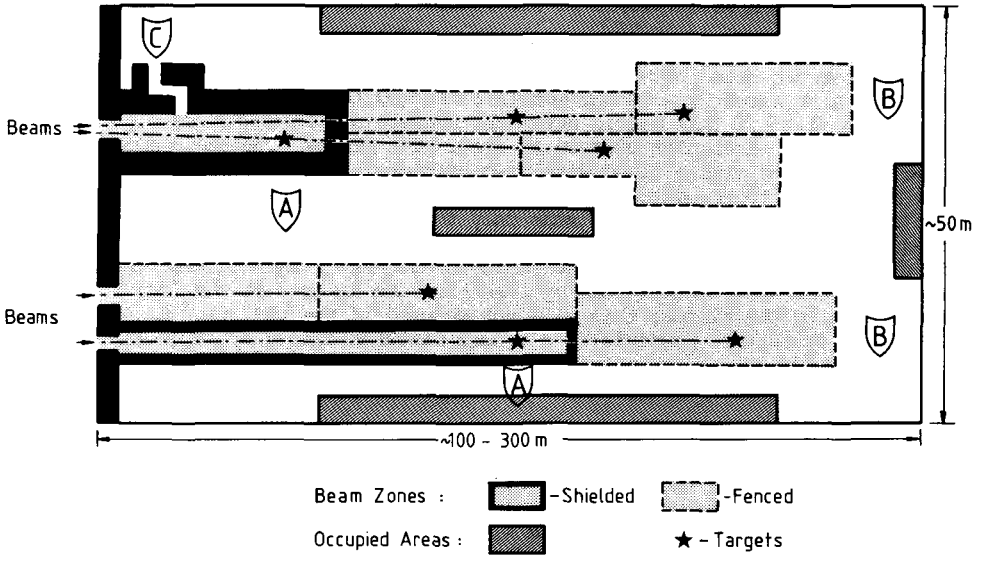

Fig. 1. Schematic layout of an experimental area.
See text for meaning of letters A, B and C.

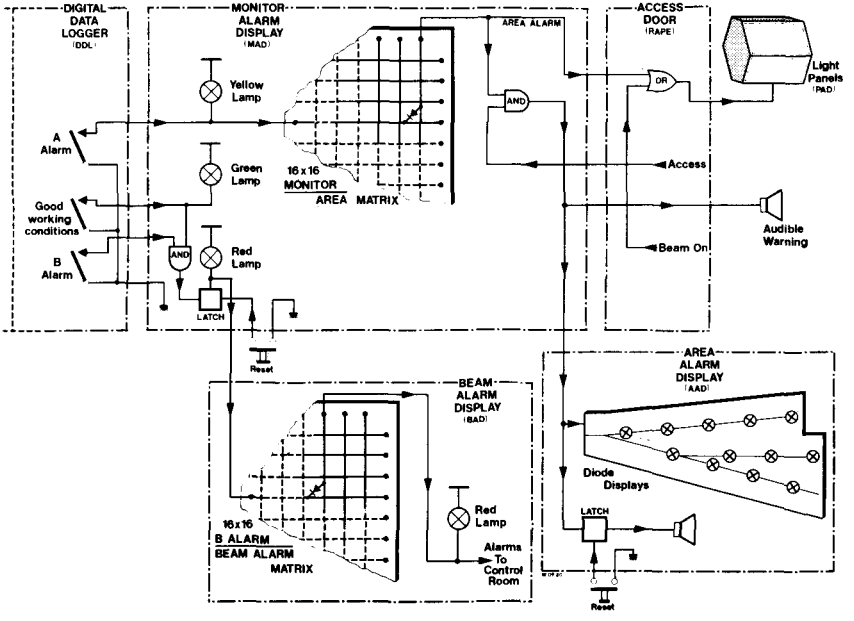

Fig. 2. Simplified functional diagram of the radiation alarm system.

THE NRPB CR-39 FAST NEUTRON PERSONAL DOSEMETER

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1. INTRODUCTION

The United Kingdom National Radiological Protection Board provides a postal neutron personal monitoring service. Until recently the service relied solely on the nuclear emulsion dosemeter[1], which is sensitive to thermal neutrons and fast neutrons above a threshold of about 500 keV. Although this energy threshold does not present a problem for the users of small neutron sources (used with relatively little or no shielding), when combined with the photon sensitivity of the emulsion, it can become restrictive on the use of the nuclear emulsion dosemeter in some applications. The operation of a nuclear emulsion dosemeter service also involves great care to prevent oxidation (fading) of the latent image and also tedious but highly skilled microscope scanning. For these reasons a dosemeter utilising the proton response of CR-39 plastic is being introduced.

CR-39 (Columbia Resin-39) is the commercial name for the thermoset plastic which is the polymeric form of diethylene glycol bis(allyl carbonate). It has widespread commercial use as it is extremely transparent and mechanically rigid. In 1978 its excellent nuclear track recording properties were discovered by Cartwright, Shirk and Price[2]. CR-39 will record particles with $z^2/\beta^2 > 36$, for example all natural alpha particles at full energy and protons up to 18 MeV[3]. Tracks are revealed by etching in alkaline solution (NaOH or KOH) followed by examination under an optical microscope using magnification of up to x1000. Tracks revealed by chemical etching may be enlarged by the technique of electrochemical etching developed by Tommasino in 1979[4] to the point where tracks are visible to the naked eye and may be conveniently counted at low magnification (x20 to x40).

2. THE NRPB DOSEMETER

2.1 Dosemeter Design and Detector Processing

The detector consists of a flat piece of CR-39 plastic approximately 25 mm x 40 mm and of thickness in the range 500 μm to 700 μm. The detector is supplied by the manufacturer, and issued, covered on both sides with a thin layer of opaque plastic. This reduces handling problems. Before issue the detectors are uniquely binary encoded by laser cut holes in an 8 x 4 location matrix. Detectors are stored in a dry nitrogen atmosphere and before issue are hermetically sealed inside polythene/aluminium laminate pouches. This reduces any oxidation effects and radon ingress. The pouch is numbered and is fitted inside a simple polypropylene holder with an open window through which this number is presented.

Only the back surface of the CR-39 detector is electrochemically etched. For this surface the detector itself is the proton radiator for all energies less than about 10 MeV. At greater energies some contribution to the response will be from protons produced in the polypropylene dosemeter holder and from (n,p) and (n,np) reactions in the aluminium of the pouch. Etching the back surface of the detector has the advantage that the response is, in general, uninfluenced by the detector covering or by any cleaning etches which may need to be introduced. Also, it has been reported that the electrochemical response is more consistent for the rear surface[5].

The processing of detectors consists of a chemical pre-etch of one hour, an electrochemical etch of sixteen hours and a post-etch of three hours (no applied

field). This regime has proved operationally convenient.

The current method of readout is to mount the detector in a photographic slide mount and back project the image onto a semi-opaque screen. The readout is being automated using a relatively inexpensive image analysis system. Areas of about 1 cm^2 are scored. The upper limit to dose equivalent which can be readily assessed is a few rem. For higher doses, it is necessary to examine, at high magnification, areas of the detector not electrochemically etched and score the chemically etched pits.

2.2 Dosemeter Performance

Batches of CR-39 plastic have been obtained from Pershore Mouldings Ltd (PM) and Bristol University (BU) with either di-isopropyl peroxydicarbonate (IPP) or di-cyclohexyl peroxydicarbonate (CHPC) initiator, and with di-octyl phthalate (DOP) plasticiser. The use of CHPC has advantages in that it is more convenient to store. Background levels and low energy response are dependent (inter alia) on the applied field strength. Background pit density increases with increasing field strength, low energy neutron response decreases with decreasing field strength. Adequate low neutron energy response with acceptably low backgrounds are obtained for field strengths between 20 and 25 kV cm^{-1}. The background levels and standard deviations shown in Table 1 are representative of those found for the fifteen batches of CR-39.

Table 1 Performance of Dosemeter

En (MeV)	Angle to normal (^{o})	Response (pits cm^{-2}) of dosemeter on phantom per unit depth dose equivalent at 10 mm, normalised to 1.5 at 0.57 MeV			
		BU Batch F 4½% IPP/0.2% DOP		BU Batch L 7% CHPC/0.2% DOP	
0.15	0	0.43	(0.21)	0.47	(0.06)
	45	0.2	(free air)		
0.57	0	1.5	(0.23)	1.5	(0.09)
	45	0.59	(0.14)	0.64	
	75	0.17	(0.05)	0.32	(0.04)
1.06	0	1.31	(0.08)	0.94	(0.11)
	45	0.55	(0.08)	0.56	(0.04)
	75	0.28	(0.04)		
5	0	0.50	(0.14)	0.47	(0.11)
	45	0.39	(0.10)	0.30	(0.08)
	75	0.23	(0.04)	0.22	(0.02)
14.7	0	0.53	(free air irradiation)		
Background (^{252}Cf equiv.) (pits cm^{-2} Sv^{-1} 10^5)		15	(8)	10	(5)

Table 1 shows results of investigations of the dependence of response of the dosemeter on neutron energy and angle. Irradiations were carried out at the National Physical Laboratory. For the phantom irradiations, dosemeters were

mounted on the front surface of a 20 cm x 30 cm elliptical cylindrical polyethylene water filled thorax phantom. The figures in parentheses give the standard deviations for the dosemeters irradiated. The uncertainties in the fluences were less than 5%. These fluences were converted to depth dose equivalent at 10 mm in the ICRU tissue equivalent sphere[6,7]. The dosemeter results are expressed as pits cm^{-2} per unit depth dose equivalent at 10 mm, normalised to a value of 1.5 for 0.57 MeV neutrons.

A series of free-air irradiations have been carried out to investigate batch variations in energy dependence of response. When normalised to the individual batch sensitivity to a ^{252}Cf spectrum, the mean values and standard deviations of the response per unit depth dose equivalent at 10 mm for seven batches of BU and PM plastic were as follows: 0.15 MeV : 0.37 (0.12); 0.57 MeV 1.6 (0.3); 1.06 MeV : 1.2 (0.3); 2.5 MeV : 0.95 (0.18); 5 MeV : 0.52 (0.11). Extensive investigation of the angular dependence of response per unit incident planar neutron fluence of different batches for monoenergetic neutrons has been carried out. These show general agreement with the results shown in Table 1 for phantom irradiations and are consistent with a series of ^{252}Cf irradiations at different angles. For a ^{252}Cf energy spectrum, the response is approximately independent (within $\pm$20%) of the planar fluence for angles to the normal up to 75°.

Table 2 Calculated Response of Dosemeter to Some Representative Neutron Spectra

Neutron Spectrum	Calculated response (pits cm^{-2}) per unit dose equivalent at 10 mm in ICRU sphere relative to uncollided fission spectrum	
	BU Batch L 7% CHPC/0.2% DOP On phantom	PM Batch R 1.6% IPP/0.2% DOP Free air
Uncollided fission	1	1
Fission neutrons thro' 10 cm D_2O	0.93	0.9
Fission neutrons thro' 40 cm concrete	0.96	0.96
Fission neutrons thro' 10 cm iron	1.2	0.98
Fission neutrons reflected 20 cm concrete	1.1	1
Spectrum A	0.76	0.68
Spectrum B	0.81	0.66

Responses have been calculated for several appropriate neutron energy spectra[8], and two measured spectra A and B (Table 2). These latter spectra are 'soft' spectra (~40% of dose equivalent below 100 keV) measured in a nuclear facility. The responses of the NRPB CR-39 fast neutron dosemeter measured in these spectra are in good agreement with the calculated responses and indicated field angular distributions.

3. FUTURE DEVELOPMENTS

Development is continuing to improve both the plastic quality and processing techniques. It is expected that CR-39 of lower intrinsic background and better uniformity will become available which, when combined with modifications in processing will result in dosemeters of higher sensitivity, lower detection limit and lower neutron energy threshold. To extend the range of the dosemeter, the inclusion of an additional CR-39 detection element is being considered. When combined with a suitable n/α converter, this will enable the estimation of the dose equivalent contribution in the range of neutron energies from thermal to about 20 or 30 keV[9].

Acknowledgements

The authors wish to acknowledge helpful discussions and exchange of data with the UK Track Etch Group, viz: J R Harvey and A R Weeks (CEGB), K G Harrison and R M Haigh (Harwell), R Proud (BNFL), D L Henshaw and T W Turner (Bristol University), A G Ramli (Birmingham University) and J C H Miles (NRPB).

References

1. Bartlett, D T, Bird, T V and Miles, J C H (1980). The NRPB Nuclear Emulsion Dosemeter, NRPB Report R-99 (HMSO, London).

2. Cartwright, B G, Shirk, E G and Price, P B (1978). A Nuclear Track-Recording Polymer of Unique Sensitivity and Resolution. Nucl. Instrum. Meth. 153, pp. 457-460.

3. Henshaw, D L (1982). Applications of Cr-39 Nuclear Track Detector in Medicine and Technology. Phys. Technol. 13, pp. 266-280.

4. Tommasino, L. (1970). Electrochemical Etching of Damaged Track Detectors by H V Pulse and Sinusoidal Waveforms. CNEN Report RT/PROT.

5. Harrison, K G and Haigh, R M (1982). Personal communication.

6. Chen, S Y and Chilton, A B (1979). Calculation of Fast Neutron Depth-Dose in the ICRU Standard Tissue Phantom and the Derivation of Neutron Fluence to Dose Index Conversion Factors. Rad. Res. 78, pp. 335-370.

7. Chilton, A B and Shiue, Y L (1981). Progress Report on Studies Concerning Doses in the ICRU Sphere Resulting from Neutron Beams. In: Proc. 4th Symp. on Neutron Dosimetry, Munich-Neuherberg, June, pp. 79-79, EUR 7448 EN.

8. Ing, H and Makra, S (1978). Compendium of Neutron Spectra in Criticality Accident Dosimetry. IAEA Tech. Report, 180. (IAEA, Vienna).

9. Bartlett, D T (1983). Measurement of Neutron Dose Equivalent from Thermal Energies up to 10 keV. Health Phys. 45, (1), p. 181.

A NEW TYPE OF MODERATE-AND-CAPTURE NEUTRON SPECTROMETER

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INTRODUCTION

The Environmental Measurements Laboratory has been using the multisphere spectrometer system (1) to measure neutron spectra for many years with quite acceptable results (2,3).

Recent experience when studying the neutron environment inside pressurized water reactor containments (3) confirmed difficulties in using this type of spectrometer for field measurements. When the spheres are exposed simultaneously, as with ^{6}LiF and ^{7}LiF thermoluminescence detector pairs (TLD-600, TLD-700), the measurement has a spatial dependence. When exposed sequentially, as with ^{6}LiI scintillation detectors, the measurement has a time dependence. Therefore, variations of the neutron flux and/or spectrum with time or space, can produce errors in the measurement distribution. Though steps can be taken to minimize the effects of such variations, they remain a concern. In addition the number and weight of the detector moderator combinations comprising the multisphere spectrometer system, make its use under field conditions cumbersome.

One approach to solving the problem of time dependence is to expose passive detectors simultaneously, and spatial dependence can be overcome if a single moderator is used. An attempt at this approach placed TLDs at various depths along 14 radii of an 18 inch diameter polyethylene sphere (4). To reduce the weight (25 kg as opposed to 46 kg for the 18 inch sphere) of the moderator and provide a geometrical shape easier to manipulate, we chose to use a single polyethylene cube, 30 cm on an edge with a TLD-600, and TLD-700 detector pair at various depths along each of the major axes. The difference in signal between the TLD-600 and TLD-700 detectors, due to the ^{6}Li(n,α)^{3}H reaction, is used as the neutron signal for that pair of detectors.

RESPONSE CALCULATIONS

Monte Carlo calculations of detector response were performed for detector depths of 1, 3, 5, 7, 9, 11, 13, and 15 cm along the major axes of the cube. These calculations were performed in the adjoint mode with the three dimensional Monte Carlo code Morse-CG (5), using both Russian roulette and splitting.

In these calculations the TLD chip used is 0.3175 x 0.3175 x 0.0889 cm on an edge, has a density of 2.678 g/cm^{3}, and is enriched

to 95.62% ^{6}Li. The polyethylene cube used is 30 cm on an edge and has a density of 0.9217 g/cm^{3}. The cross sections used are the same as in our previous calculations for the multisphere spectrometer (6).

Large numbers of batches, each batch comprising several thousand histories, were run. Averages and standard deviations were then determined, for each energy group, among the batches. Enough batches were run to reduce the statistical uncertainty to less than 10% at the 1, 3, 5, and 7 cm depths, and to between 18% and 22% at greater depths. This statistical uncertainity was as low as could practically be achieved because of computer time restrictions. The results of these calculations are shown in Figure 1. Although the geometries are different, a comparison of these results with those found in references 4 and 6 indicates the reasonableness of these responses.

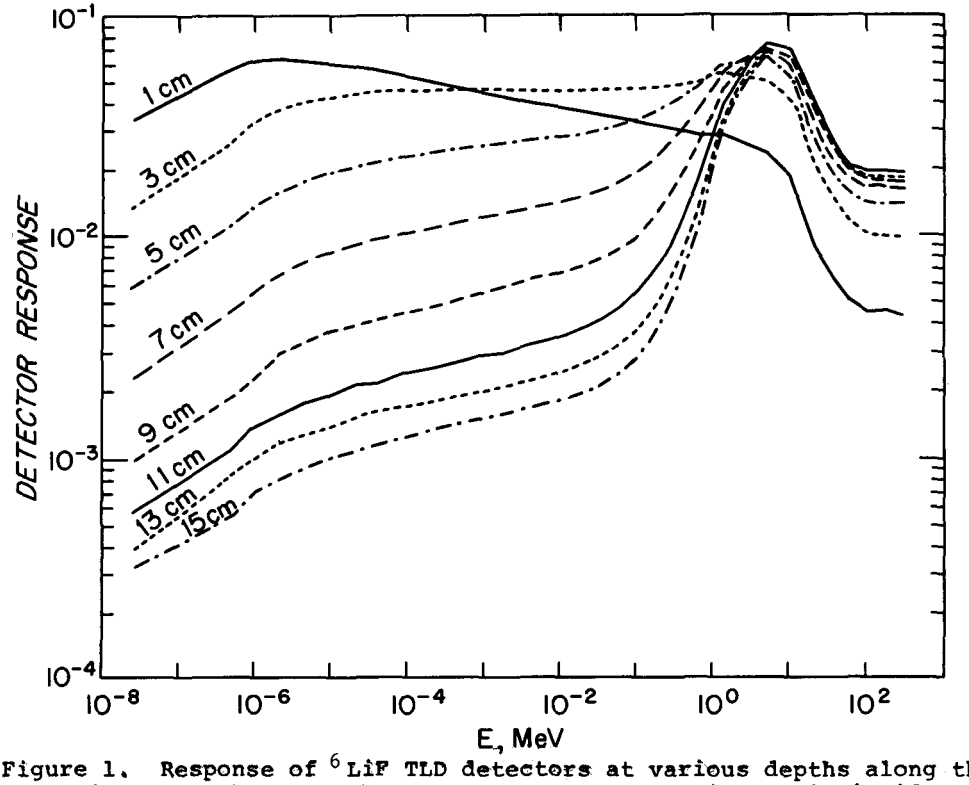

Figure 1. Response of ^{6}LiF TLD detectors at various depths along the axes of a polyethylene cube 30 cm on an edge for isotropic incidence.

TESTING THE SPECTROMETER

Mathematical testing of the ability of a spectrometer with the above responses to unfold neutron spectra from measurement distributions was performed with satisfactory results.

The response matrix was used to synthesize measurements from realistic, and idealized single and double line spectra. These pseudo-measurement distributions were analysed using the Monte Carlo spectral unfolding code SWIFT (7) to demonstrate the ability of the spectrometer to unfold neutron spectra from a variety of sources.

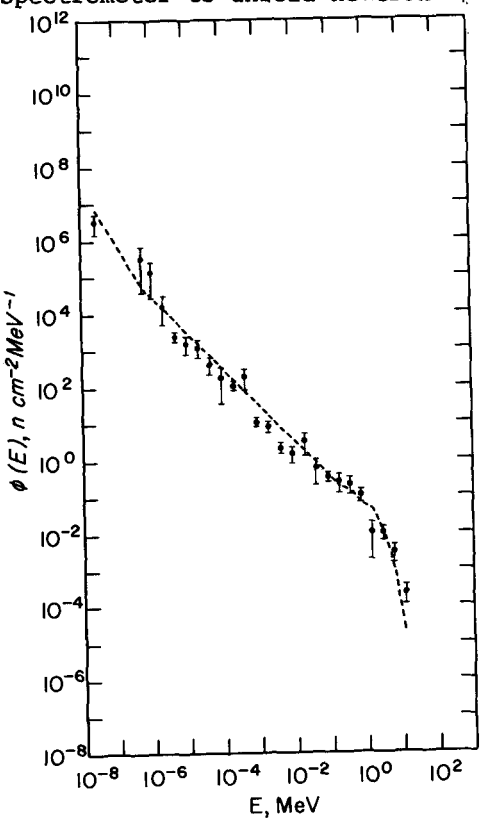

Figure 2. Comparison of original (dotted line) and unfolded spectra (points with error bars) in a beryllium reflector of a homogenous light-water reactor (8).

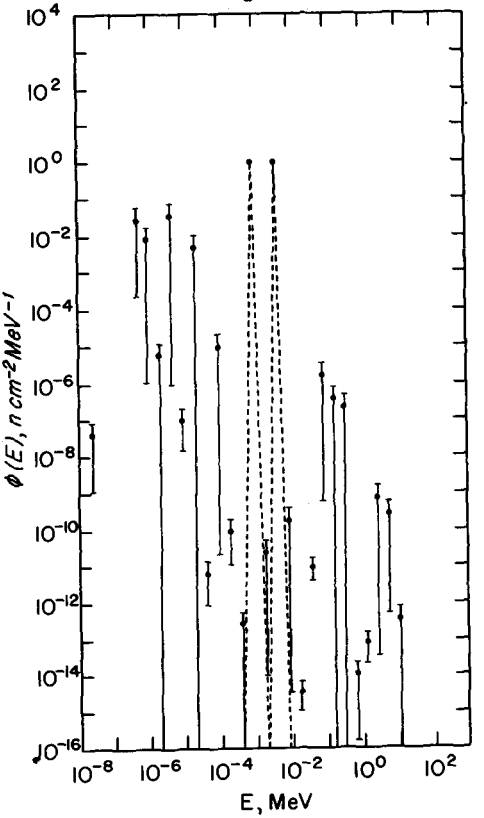

Figure 3. Comparison of original (dotted line) and unfolded spectra (points with error bars) for a double-line spectrum in energy groups 12 and 14.

Figures 2 and 3 compare the original spectrum in a beryllium reflector of a homogenous light-water reactor (8), and a double line spectrum with the unfolded spectrum from their measurement distributions. Agreement is good and is similar to that obtained for the other spectra tested. For the double line spectrum, Figure 3, the two lines contain 99.99% of the total flux and the precision is such that the standard deviations of the peaks are less than 0.5%, also the "noise" for any other energy group is less than 4% of the peaks.

In further tests, random errors were synthesized into the pseudo-measurements by sampling from cumulative Gaussian distributions. The means were the exact pseudo-measurement distributions and the standard deviations were 5%, 10%, 15%, and 20%. Four new distributions were generated for each standard deviation, unfolded, and the results compared to the original spectrum. These tests indicate that deterioration of the unfolded spectrum becomes marked when the level of error reaches 10%.

CONCLUSIONS AND ONGOING STUDIES

The cube should prove easier to use in the field and provide results of similar quality to those obtained with the multisphere spectrometer. Ongoing studies include empirical verification of the calculated responses and of the isotropy of response of the cube. Additional work will include a study of a cube, 25 cm on an edge, in an attempt to reduce even further the weight of the moderator. We also intend to study the feasibility of placing the TLDs along the diagonals of the cube to determine if better responses are obtained.

REFERENCES

1. Bramblett, R. L., Ewing, R. I., and Bonner, T. W., "A New Type of Neutron Spectrometer", Nucl. Inst. Meth., 9, 1, 1960.

2. O'Brien, K., Sanna, R. S., and McLaughlin, J. E., "Inference of Accelerator Stray Neutron Spectra from Various Measurements", First Symposium on Accelerator Radiation Dosimetry and Experience, Brookhaven National Laboratory, November 3-5, 1965, CONF-651109.

3. Sanna, R. S., Hajnal, F., and McLaughlin, J. E., "Energy-Dependent Effects on Neutron Monitor Performance in PWR Containments", Health Physics, 43, 263, 1982.

4. Piltingsrud, H. V., and Engelke, M. J., "A Passive Broad-Energy-Response Neutron Spectrometer-Dosimeter", Proceedings of a Symposium on Neutron Monitoring for Radiation Protection Purposes, Vienna, December 11-15, 1972, IAEA-SM-167/51.

5. Radiation Shielding and Information Center, "MORSE-CG General Purpose Monte Carlo Multigroup Neutron and Gamma-Ray Transport Code, Combinatorial Geometry", Oak Ridge National Laboratory Report, RSIC Code Package CCC-203.

6. Sanna, R. S., "Thirty One Group Response Matrices for the Multisphere Spectrometer, Over the Energy Range Thermal to 400 MeV" U. S. Atomic Energy Commission eport HASL-267, March 1973.

7. O'Brien, K., and Sanna, R. S., "Neutron Spectral Unfolding Using the Monte Carlo Technique", Nucl. Instr. Meth. 185, 277, 1981.

8. Moteff, J., Radiation Dosimetry, Vol. 3, eds., Attix, F. H., and Tochilin, Academic Press, New York, 1969.

UN APPAREIL DE MESURE DES DOSES NEUTRONIQUE , PHOTONIQUE ET DU FACTEUR DE QUALITE DANS UN CHAMP MIXTE NEUTRON-GAMMA ,FONDE SUR L'UTILISATION D'UN COMPTEUR PROPORTIONNEL DE ROSSI.

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I-INTRODUCTION :

A l'aide des distributions microdosimétriques mesurées grâce à un compteur proportionnel de Rossi, il est possible de calculer le facteur de qualité moyen $\overline{Q}$ d'un champ mixte neutron gamma et de déterminer l'équivalent de dose correspondant.

Si $\frac{D_g}{D_t}$, $\frac{D_p}{D_t}$, $\frac{D_{il}}{D_t}$ représentent les contributions relatives en dose dues respectivement aux électrons, protons et particules lourdes, nous avons (/1/) :

$$\overline{Q} = \frac{D_g}{D_t} + (0{,}8 + 0{,}16\ \overline{L}_{Dp})\ \frac{D_p}{D_t} + 20\ \frac{D_{il}}{D_t}$$

avec $\overline{L}_{Dp} = \overline{L^2}/\overline{L}$ dans la zone protonique seule.

$\overline{L}_{Dp} = \frac{8}{9}\ \overline{y}_{D,p}$ lorsque les protons de recul sont de grande énergie (la distribution des cordes est triangulaire)

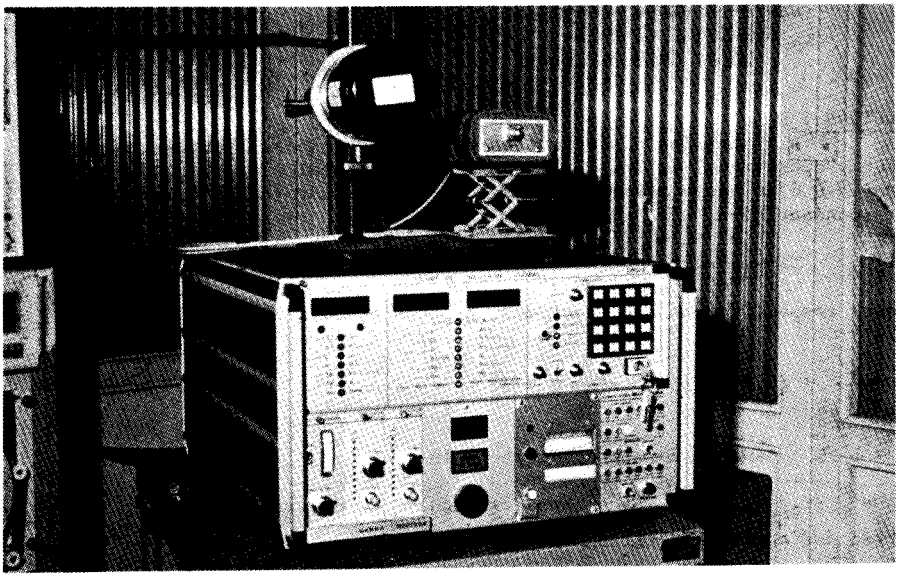
$\overline{L}_{Dp} = \overline{y}_{D,p}$ lorsque la distribution est rectangulaire. Ceci correspond aux protons de recul faiblement énergétiques, cas le plus fréquemment rencontré en radioprotection.

Un tel capteur (type Let 1/2 ou Let 2 de la firme FWT (x) associé à un préamplificateur de charge et à une électronique portable permet d'obtenir en un seule mesure le facteur de qualité et l'équivalent de dose d'un champ mixte neutron-gamma.

(x) FWT : FAR WEST TECHNOLOGY - Inc. GOLETA (CALIFORNIE)

2. PRINCIPALES CARACTERISTIQUES DE L'APPAREIL CENG (Contrôleur d'Environnement en Neutron Gamma)

Le compteur proportionnel est un capteur Let 2 de la firme FWT (x)



L'appareillage mis au point et réalisé par la Société NOVELEC (xx)remplace l'analyseur multicanal d'amplitude, l'amplificateur et effectue les calculs nécessaires à l'obtention des principales données dosimétriques du champ mixte neutron/gamma étudié.

Pendant un temps de mesure prédéterminé Tm, les impulsions issues de l'ensemble compteur-proportionnel-préamplificateur de charge sont dirigées, selon leur amplitude, vers deux voies identiques ayant un rapport d'amplification voisin de 70. Chaque voie est connectée à un détecteur de crête et à un codeur d'amplitudes. Les impulsions sont ensuite acheminées vers un microprocesseur 6809 qui à l'aide des paramètres correspondant à l'expérience effectue les calculs nécessaires.

Dans le mode mesure (qui correspond à la fonction essentielle de l'appareil) les résultats suivants (importants en radioprotection) sont affichés et peuvent être imprimés :

- dose totale et débit de dose
- équivalent de dose total et débit d'équivalent de dose
- rapport dose gamma/dose neutron
- facteur de qualité moyen
- dose gamma
- dose neutron
- équivalent de dose neutron

(xx)NOVELEC: ZIRST (Chemin des Clos), 38240 MEYLAN (FRANCE)

Trois autres modes de fonctionnement sont aussi possibles :

- un mode "paramètres" permettant de rentrer sur clavier les valeurs des différents paramètres nécessaires aux calculs ultérieurs et notamment le temps de mesure, le diamètre géométrique du dosimètre, le facteur de qualité pour les photons, les ions lourds, etc ...

- un mode "ictomètre" pour l'évaluation rapide du taux de comptage dans les zones gamma et neutron et donc de l'adaptation de la sensibilité du capteur employé.

- un mode "étalonnage" permettant la détermination du facteur d'amplification du dosimètre utilisé lorsque la source interne alpha d'^{241}Am ou de ^{244}Cm est démasquée, pour une position particulière de ce dernier et donc de la correspondance volt→keV.μm^{-1}.

La fréquence maximale admissible est voisine de 15 kHz ce qui correspond à des débits de dose photonique et neutronique respectivement de 0,1 rad.h^{-1} et 10 rad.h^{-1} pour un compteur de 5.7 cm de diamètre. La puissance consommée est de l'ordre de 60 W.

3 - RESULTATS PREMIMINAIRES OBTENUS

3.1 - IRRADIATION PHOTONIQUE PURE (^{60}Co, ^{241}Am).

Pour un diamètre simulé de 2 µm, nous avons obtenu les résultats suivants :

Source	$\dot{D}_{rg}$ (mrad.h^{-1})	$\dot{D}_m$ (mrad.h^{-1})	$\bar{Q}$	R	T_m (minutes)
^{60}Co	51.4	38.7	1,4	0,75	2
	23.6	18.3	1.5	0.78	
	8.9	7.5	2.1	0.84	2
	3.3	3.0	2.2	0.91	
	2.3	2.2	2.6	0.96	10
^{241}Am	48.9	53.2	1.3	1.09	
	3.1	3.7	1.9	1.20	
	1.4	1.8	2.6	1.30	

.$\dot{D}_{rg}$ (mrad.h^{-1}) est le débit de dose photonique de référence dans les tissus sous 0.3 g.cm^{-2}.

.$\dot{D}_m$ (mrad.h^{-1}) est le débit de dose mesuré

.$\bar{Q}$ est le facteur de qualité mesuré

.$R = \dot{D}_m/\dot{D}_{rg}$

. T_m est le temps de mesure en minutes.

Nous constatons que :
$\bar{Q}$ est d'autant plus proche de l'unité que le débit d'exposition est plus grand devant le bruit du dosimètre. La sensibilité du détecteur dépend de l'énergie photonique incidente.

3.2 -MESURES NEUTRONIQUES (E_n = 14,7 MeV)

Les deux composantes du champ mixte n/γ sont déterminées à l'aide d'une chambre d'ionisation équivalent tissu et d'un compteur Geiger-Müller. Nous reportons dans le tableau suivant les mesures effectuées pour divers débits de dose, des temps de mesure Tm différents et un diamètre simulé de 2 µm :

$\dot{D}_{rn}$ mrad.h^{-1}	R_g	R_n	R_T	$\bar{Q}$	Tm minutes
3.4	0.84	1.05	1.03	8.05	2
	0.94	1.03	1.02	7.88	20
16.6	0.71	1.04	1.0	7.74	2
	0.73	1.04	1.0	7.79	10
45.6	0.69	1.01	0.97	7.66	5

.$\dot{D}_{rn}$ est le débit de dose neutronique dans les tissus de référence.

.$R_g = \frac{D_{mg}}{D_{rg}}$ D_{mg} et D_{rg} sont respectivement pendant le temps T_m, les doses mesurées et de référence dans les tissus dues à la composante photonique

.$R_n = \frac{D_{mn}}{D_{rn}}$ D_{mn} et D_{rn} sont respectivement pendant le temps T_m, les doses mesurées et de référence dans les tissus dues à la composante neutronique.

.$R_T = \frac{D_m}{D_{rn} + D_{rg}}$ Dm est la dose mesurée totale pendant Tm.

Nous constatons que :

. L'influence du bruit du dosimètre sur R_g (de l'ordre de 0,5mrad.h^{-1}) est d'autant plus importante que $\dot{D}_{rn}$ est plus grand.

. Le facteur de qualité moyen $\bar{Q}$ est en bon accord avec la valeur de référence /1/

Les mesures, pour d'autres énergies neutroniques, sont en cours d'exploitation.

4 - CONCLUSION (/2/)

Dans le cas des neutrons de 14.7 MeV, une mesure de 2 minutes avec un capteur de 5.7 cm de diamètre conduit à une précision inférieure à 40 % sur le facteur de qualité $\bar{Q}$ et la composante neutronique, pour des débits de dose de l'ordre de 3.0 mrad.h^{-1}.

En revanche, la connaissance de la composante photonique nécessite la réalisation de capteurs de grande fiabilité. Des développements ultérieurs, tels que la réalisation de balises, d'appareils portatifs ou à plus long terme, l'étude d'un dosimètre individuel neutron-gamma peuvent être envisagés sur ce principe.

5-REFERENCES BIBLIOGRAPHIQUES :

/1/ : LEROUX J.B. , HERBAUT Y: RAPPORT CEA R 5208 (1983)

/2/ : LEROUX J.B. , HERBAUT Y :8 th SYMPOSIUM ON MICRODOSIMETRY EUR 8395 (1982) p 1141-1152.

A PASSIVE FAST-NEUTRON SPECTROMETER-DOSIMETER BASED ON CR-39

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We describe here a new passive miniature fast neutron spectrometer-dosimeter. The device is based on the detection of proton tracks by electrochemical etching of CR-39 foils covered with thin polyethylene layers of different thicknesses. By means of this device it is possible to assess the fast neutron energy spectrum in 10 energy intervals in the energy range 0.5-15 MeV. Dose equivalents can be determined in the dose equivalent range 20 mRem to 8 Rem, approximately.

The device has been used by us for the evaluation of the photoneutron spectrum in the beam of a high-energy (16 MeV) medical linear accelerator (1). We anticipate the use of this system for the assessment of fast neutron spectra in fusion reactor blankets, for the experimental verification of isodose distributions in neutron radiotherapy and possibly as a personnel fast neutron spectrometer-dosimeter in high-energy accelerator facilities.

The miniature neutron spectrometer (MNS) is composed of a CR-39 foil (American Acrylics and Plastics, Stratford, Connecticut), ∿500μm thick, and polyethylene of various thicknesses in contact with it. The CR-39 foil detects recoil protons emitted from the radiators, due to elastic scattering of the incident neutrons with hydrogen nuclei in the polyethylene. A schematic diagram of the 27x27x0.5mm spectrometer is shown in Fig. 1, and a photograph of it in Fig. 2. The foil is

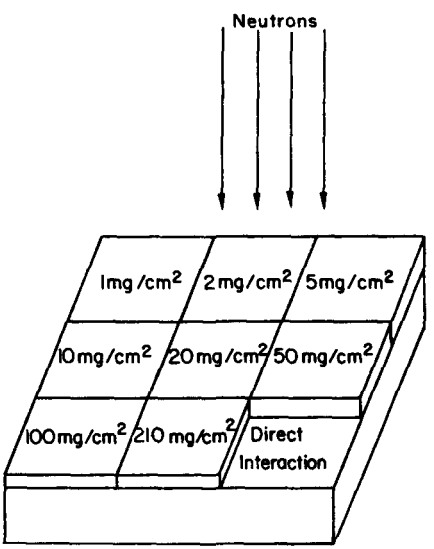

Fig. 1. Schematic diagram of the spectrometer-dosimeter,

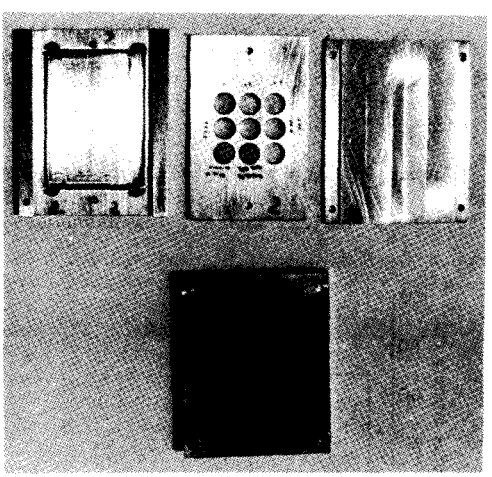

Fig. 2. Photograph of the spectrometer-dosimeter.

divided into nine areas, eight of which are masked by polyethylene radiators of thicknesses ranging from 1mg/cm^2 to 210mg/cm^2. The ninth area is not masked and is used to assess the number of damage sites resulting from the foil background and from direct neutron interactions with the CR-39. The CR-39 foil is embedded in a stainless steel box with a wall thickness of about 400mg/cm^2. This box absorbs high energy recoil protons due to elastic scattering of fast neutrons with moisture in the air or with the phantom material. Two MNSs were built, each having a different set of polyethylene thicknesses. Thus, the simultaneous use of 14 proton radiators makes it possible to extract the neutron spectrum with much more detail.

The lower limit for the extraction of neutron spectra is imposed by the low sensitivity of the CR-39 for detection of protons having energies lower than 300 keV. Added to this is the technical difficulty of mounting polyethylene layers of about 0.7mg/cm^2 thickness, which corresponds to the range of 0.5 MeV protons. The maximum thickness of polyethylene radiator through which protons can pass and reach the CR-39 detector with sufficient energy to cause damage sites, increases with neutron energy. The device is based on the fact that for lower neutron energies, $\sim$ 0.5 MeV, there is no difference in the number of protons emerging from the radiators of different thicknesses, but for higher neutron energies, $\sim$15 MeV, (where the range of the most energetic proton is about 200mg/cm^2), there is a difference of about two orders of magnitude between the number of protons emerging out of the thickest and the thinnest radiators.

The energy spectrum was found in 10 energy channels with 14 polyethylene radiators, by a least squares minimization procedure which involves the proton yields produced beyond all the radiators and the revealing efficiencies of proton tracks in the CR-39 material. The damage sites on the CR-39 foil are revealed by chemical pre-etching for 24 hours at 35^{o}C, followed by electrochemical etching for 3 hours at 25^{o}C. The etchant solution is 9N KOH and the voltage and frequency are 1000V (RMS) and 1900 Hz, respectively. Either one or both sides of the foil were etched. It was found that electrochemical etching of only one side greatly enhances the contrast, giving a higher proton detection efficiency.

The spectrometer was irradiated with 14.8 MeV monoenergetic neutrons, as well as ^{252}Cf and ^{239}Pu-Be neutron sources. The calculated yield of protons emitted from each radiator was adjusted to give the same experimental ratios obtained with 14.8 MeV neutrons. Experimental yields are better reproduced when assuming a zero revealing efficiency for protons above 10 MeV and a maximum dip angle of 55^{o}.

Figure 3 shows the obtained ^{252}Cf spectrum as compared to the photoneutron spectrum obtained from a 16 MeV M.E.L. medical linear electron accelerator (Oncology Dept., Sheba Medical Center, Tel-Hashomer). The expected shift towards lower

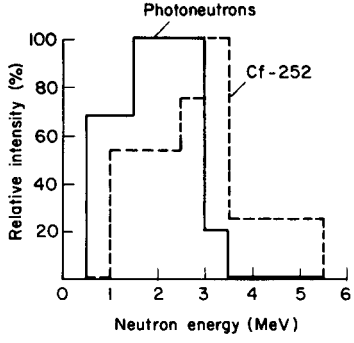

Fig. 3. ^{252}Cf and photoneutron spectra obtained with the miniature neutron spectrometer.

energies of the photoneutron spectrum is clearly seen (2).

When both sides of the CR-39 foil were etched it was observed that in the region not masked by polyethylene radiators the back side of the foil shows a larger yield than the front side. This is due to the fact that the CR-39 molecule contains hydrogen . (The thickness of the CR-39 foil used is equivalent to about 20mg/cm^2 polyethylene.) The back-to-front ratio of yields for ^{252}Cf and ^{239}Pu-Be sources is about 3.5, while for 14.8 MeV incident neutrons it decreases to about 1.5. This experimental fact might be used to obtain additional spectroscopic information.

Acknowledgement

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References

1. S. Faermann, Y. Eisen, E. Ovadia, T. Schlesinger, A. Werner, L. Zhigun and A. Kushilevski, 10th Joint Annual Meeting of the Israel Nuclear Societies, Tel-Aviv, December, 1982.
2. M.E. Sanders, K.Z. Morgan and P.A. McGinley, 25th Annual Meeting of the Health Physics Society, Seattle, Washington, July 23, 1980.

STANDARD IRRADIATION FACILITY FOR THE CALIBRATION OF RADIATION PROTECTION INSTRUMENTS EMPLOYING RADIONUCLIDE SOURCES

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INTRODUCTION

Measuring instruments used for radiation protection purposes should in principle be calibrated in terms of those quantities for which the limits have been specified by regulations or recommendations. In practice, however, operational quantities must be used in order to link the calibration quantities to the limiting quantities /1/. The dose equivalent ceiling /2/ has up to now been used as operational quantity. It is based on the recommendations given in ICRP Publication 21 /3/.

In routine testing radionuclide sources are particularly suitable for producing standard neutron fields. The ISO-working group TC 85/SC 2/WG 2/SG 3 has therefore proposed neutron reference radiations for the calibration of neutron measuring devices used for radiation protection purposes. These include the following radionuclide sources: ^{252}Cf (bare and moderated by D_2O), ^{241}Am-B(α,n), and ^{241}Am-Be(α,n). From these sources bare ^{252}Cf and ^{241}Am-Be(α,n) should preferably be used for calibrations /4/. The American National Standards Institute (ANSI) has recently adopted the moderated ^{252}Cf-source as the sole source for calibrating personnel dosimeters /5/. At the Physikalisch-Technische Bundesanstalt (PTB) a standard irradiation facility is being installed which makes use of sources of this type.

The following sections deal with the calibration procedure in general and with the special arrangements to be made. The specifications of the sources and results of the calibration of a commercial remmeter are given for demonstration purposes.

CALIBRATION PROCEDURE

To calibrate it, an instrument is brought into a well-defined neutron field. The dose equivalent H produced at the point of reference is related to the indication M of the instrument (i.e. the number of counts) which would instead be produced at the same place under defined conditions. The calibration factor is then defined by $N = H/M$, or $N = \dot{H}/\dot{M}$ if time-related quantities are used. In the special case of irradiations with a broad parallel beam of unidirectional monoenergetic neutrons of energy E the dose equivalent can be calculated from the fluence ϕ using fluence-to-dose equivalent conversion factors $h_\phi(E) = H/\phi$ which are based on the recommendations given in ICRP Pub. 21 /3/ and on German regulations /6/. The behaviour of the instrument in this special field is described by the fluence response $R_\phi(E) = M/\phi$. In the case of neutrons with a broad energy spectrum as is generally emitted by radionuclide sources, mean conversion factors $\bar{h}_\phi$ averaged over the neutron energy spectrum are used together with the fluence ϕ /7/. In this case the mean fluence response is given by $\bar{R}_\phi = M/\phi$ where M is the indi-

cation of the instrument in the broad neutron energy spectrum with fluence ϕ. The calibration factor is then given by $N = \bar{h}_\phi / \bar{R}_\phi$. It should be kept in mind that the value of N is valid only for the special neutron energy spectrum and the irradiation geometry used for calibration ("calibration conditions"). In practical measurements the energy of the neutrons and their directional dependence are not known or only to a certain extent.

Several problems arise in the evaluation of $\bar{h}_\phi$ and $\bar{R}_\phi$:

For the calculation of $\bar{h}_\phi$ the neutron energy spectrum of the source must be known. In the case of the D_2O-moderated ^{252}Cf source the spectrum was calculated by Ing /8/ applying a Maxwell distribution with a spectrum parameter of 1.42 MeV for the bare ^{252}Cf source which corresponds to a mean energy of 2.13 MeV. $\bar{h}_\phi$ amounts for the moderated source to $9.0 \cdot 10^{-11}$ Sv.cm^2 /8/ and to $3.4 \cdot 10^{-10}$ Sv.cm^2 in the case of the bare ^{252}Cf source /4/. For the ^{241}Am-Be(α,n) source the neutron energy spectrum was recently measured yielding an $\bar{h}_0$-value of $3.8 \cdot 10^{-10}$ Sv.cm^2 /7/.

For the determination of the mean fluence response $\bar{R}_\phi$ the fluence ϕ and the indication M of the instrument induced by neutrons coming directly from the source are needed. The fluence ϕ can be calculated from the source strength taking into account the anisotropic emission of neutrons from the source. The value of ϕ must be corrected for air scattering. If the background is measured by means of a shadow cone only outscattering has to be considered using mean values of the cross section density for scattering on air averaged over the neutron energy spectrum of the special neutron source used.

The contribution of room and air-scattered neutrons to the indication of the instrument can be taken into account by measuring the scattered portion with a shadow cone the transmission of which can be neglected, or by recording the indication at different distances. From the latter, the fluence response $\bar{R}_\phi$ is derived assuming a constant background of room-scattered neutrons /9/ or a distance dependence of the indication M which can be described by the relation $M(L) = A/L^2 + B/L + C$ (L distance) /10/. Shadow cone measurements have shown that there is a noticeable distance dependence of background due to scattered neutrons. If the fluence response is determined assuming a constant background /9/, in special cases the deviations to be anticipated for $\bar{R}_\phi$ can be corrected /11/.

The effective centre of the instrument must be determined. It is defined as the point of reference in the instrument for which the indication of neutrons coming directly from a point source follows the $1/L^2$ law. It can be determined from the indication of direct neutrons at different distances /12/.

In addition, the value of $\bar{R}_\phi$ must be corrected in the case of a divergent beam of incident neutrons. This correction is known only for spherical devices and point sources /13/.

The following requirements for a calibration facility must be seen in connection with the procedure and the problems mentioned above:

1. The room used for calibration should be as large as possible

in order to minimize the contribution of room-scattered neutrons to the indication. In general, thick walls will be necessary because of radiation protection when sources of large source strength are used.

2. The source should be positioned in the middle of the room in order to minimize the local dependence of fluence or fluence rate on room-scattered neutrons.

3. There should be as little scattering material as possible in the vicinity of the source and the detector.

4. For distance measurements, the neutron measuring device must be precisely positioned by remote control with respect to the neutron source. An uncertainty of about ±0.2 mm is sufficient also for very small distances down to 10 cm (centre to centre).

5. Sources of different type and source strength should be available by remote control to meet the requirements of the particular calibration with regard to dose equivalent average energy and dose equivalent rate.

STANDARD IRRADIATION FACILITY

The standard irradiation facility of the PTB is being installed in a "bunker" (dimensions 7 m x 7 m x 6,5 m) with concrete walls and ceiling 1 m thick. Five different sources can be positioned in the middle of the room at a height of 3.25 m, including a moderator sphere (30 cm in diameter) filled with heavy water (concentration: 99.67%), into the centre of which a ^{252}Cf source can be placed. This source is situated on the top of a small cylinder also filled with D_2O, so that the ^{252}Cf source is completely surrounded by heavy water. The moderator sphere is completely shielded by cadmium 1.1 mm thick. The source strength of the ^{252}Cf source used in the moderating sphere amounts to $6.50 \cdot 10^7$ s^{-1} (1 Oct. 1983) which corresponds to a source strength of the moderated source of $5.79 \cdot 10^7$ s^{-1} taking into account a factor of 0.89 for the absorption af thermalized neutrons in the Cd shielding /8,9/. A dose equivalent rate of 40.5 nSv/s (14.6 mrem/h) is then produced at a distance of 1 m.

Up to three bare ^{252}Cf sources with different source strengths are used. By changing the distance between source and detector, dose equivalent rates within the range from about 3.4 pSv/s (1.2 µrem/h) to about 13 µSv/s (4.7 rem/h) (1 Oct. 1983) can be produced for this type of source. In addition, a ^{241}Am-Be(α,n) source can be used for calibrations producing a dose equivalent rate of 9.3 nSv/s (3.3 mrem/h) at a distance of 1 m.

Source strength B and the function describing the anisotropic emission were determined for all sources. B is measured by gold foil activation in a water bath to an uncertainty of about ±1.5 % (standard deviation) /14/.

In a second procedure, the source strength can be determined by means of a long counter comparing the emission of the source under investigation with that of a reference source of the PTB /15/.

CALIBRATION OF A SPHERICAL REMMETER

A commercial remmeter (Centronic, diameter of moderator: 20.8 cm) was calibrated using a ^{252}Cf source (bare and moderated) and a ^{241}Am-Be(α,n) source. The background of scattered neutrons was measured using shadow cones. In the case of the moderated source a shadow cone of quadratic cross section 50 cm in length and consisting of borated polyethylene was used. The dose equivalent response of the instrument was adjusted in accordance with the instructions given by the manufacturer. Table 1 shows the results of the $\bar{R}_\phi$ values from which the calibration factor N was calculated using the mean conversion factors quoted. The quoted uncertainties (standard deviations) include contributions from the measurement of source strength, the correction for outscattering due to air, the background determination by means of the shadow cone, the choice of the geometric centre for distance measurement, the correction for the case of a divergent beam, and for counting statistics. No uncertainties were taken into consideration for the mean conversion factors used.

Table 1

Source	$\bar{R}_\phi$ in cm^2	$\bar{h}_\phi$ in $Sv.cm^2$	N in nSv
$^{252}Cf + D_2O$	0.112 ± 0.003	$9.0 \cdot 10^{-11}$	0.80 ± 0.02
^{252}Cf	0.268 ± 0.005	$3.4 \cdot 10^{-10}$	1.27 ± 0.02
^{241}Am-Be(α,n)	0.252 ± 0.005	$3.8 \cdot 10^{-10}$	1.51 ± 0.03

REFERENCES

/1/ Wagner, S.R.; Rad. Prot. Dos. Vol. 4 No. 2 (1983) p. 67
/2/ Harvey, J.R.; Phys. Med. Biol. 20(1975) p. 1003
/3/ ICRP.; Suppl. to ICRP Publ. 15 and ICRP Publ. 21, (1973)
/4/ Champlong, P., Chartier, J.L., Cosack, M., Wagner, S., Alberts, W.G., Kluge, H., Schraube, H., Delafield, H.J., Hunt, J.B., Thompson, I.M.G., Lembo, L., Schwartz, R.B., Widell, C.O.; Proc. Fourth Symp. on Neutron Dosimetry, EUR 7448, Vol. I (1981) p. 387
/5/ American National Standards Institute; ANSI Standard N13.11 (1983)
/6/ Strahlenschutzverordnung-StrSchV; BGBl. I (1976) p. 2905
/7/ Kluge, H., Weise, K.; Rad. Prot. Dos. Vol. 2 No. 2 (1982) p. 85
/8/ Ing,H., Cross, W.G.; Health Phys., in press
/9/ Schwartz, R.B., Eisenhauer, C.M.; National Bureau of Standards Special Publication 633 (1982)
/10/ Hunt, J.B.; Private communication
/11/ Kluge, H., Rasp, W.; Bericht 16. Jahrestagung des Fachverbandes für Strahlenschutz e.V., FS-83-30-T, (1983) p. 117
/12/ Hunt, J.B.; NPL Report RS 5, April 1976
/13/ Harrison, K.G.; Nucl. Instr. Meth. 184 (1981) p. 595
/14/ Matzke, M.; Proc. Symp. int. sur l'utilisation du californium 252, Paris 1976, Conf-760436, Vol. II p. II-175
/15/ Zill. H., Ebeling, G.; Nucl. Instr. Meth. 174(1980) p. 491

DEVELOPMENT OF EVALUATING PHOTON SPECTRA IN THE RADIATION FIELDS AROUND NUCLEAR REACTORS

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1. INTRODUCTION

Many problems are kept to be clarified in the radiation field around nuclear reactors. For an example, there have been little information on photon energy spectra around nuclear reactors under operation. The works made preliminarily by the authors around a research reactor (KUR) showed that photons would be found up to about 9 MeV on the observed pulse-height distributions using a scintillation spectrometer. Energy spectra give substantial informations on the environment of interest.

The fact that an photon energy spectrum might be different from that in natural environment was revealed through the preliminary works. Therefore, in order to acquire the detailed knowledge on the radiations to which occupational workers have been exposed, it was necessary to develop the measurement method for photon spectra and to clarify the differences of photon spectra from those in natural environment, the spatial changes of the spectra in the environment around nuclear reactors and the difference in the spectra according to the types of reactors.

The present paper describes the method evaluating the photon energy spectra, and some examples of the photon spectra observed around research reactors (YAYOI and KUR) and in PWR site. And a prudent attitude is suggested for radiation protection control in the near future.

2. EXPERIMENTAL

To obtain pulse-height distributions due to photons having those energy up to about 9 MeV, a scintillation spectrometer with a 3"ϕ X 3"L NaI(Tl) was used. The pulse-height distribution obtained was unfolded by an iterative technique (1) with 46 X 46 response matrix of the detection system. The present response matrix was constituted referring to the results of Monte-Carlo simulation made by Zerby & Moran and others (2, 3, 4). Pulse-height resolution, peak efficiency and peak-to-total ratio were modified so as to fit their results to the real detection system.

Spectral measurements were made around two types of research reactors and in the PWR site. One research reactor (YAYOI; 2 kW) could be operated both in the concrete shield and in the lead one. The other research reactor (KUR; 5 MW) was a swimming pool type. The PWR was operated only at sixty percent of maximum power, spectral measurements being made.

3. RESULTS and DISCUSSION

The dependence of the photon spectra on thermal power of the reactor was studied in the reactor room of YAYOI, when the reactor core was set in a heavy concrete shield and operated. The result was given in Fig. 1. Thermal power was stepped up from 1 W to 1 kW by each ten times. The figure shows that the spectral intensity in the region above 2.6 MeV increased almost proportionally with the increase of the thermal power of the reactor. Under low power operation, distinct peaks of ^{40}K (1.46 MeV) and ^{208}Tl (2.6 MeV; Th-series) can be clearly seen. These natural photons

would be emitted from building materials.

A similar result was obtained in the environment around KUR, and given in Fig. 2. The spectra given in Fig. 2 were observed at the place where was hardly affected by photons scattered by the surrounding materials. Therefore, the rapid increase in intensity could not be found in the spectra with the decrease of photon energy.

When the reactor was operated in the lead shield, the spectra were observed as smoother ones than those in the case of the heavy concrete shield. This could be due to the difference of the shielding power for photons between heavy concrete and lead. However, the photon peaks were not clearly found in the spectra because of the poor pulse-height resolution of the present detector. And spectral shapes of the photons were largely varied from point to point.

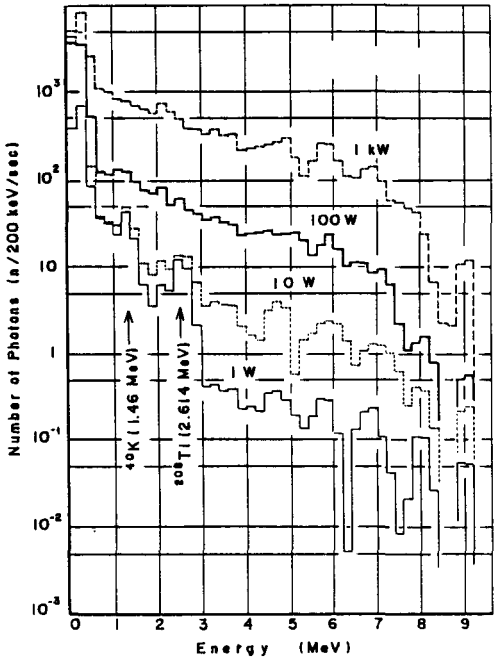

Fig. 1 The relation between thermal power and photon energy spectra, YAYOI being operated inside the heavy concrete shield.

It was difficult to find out substantial differences between the spectra observed at different types of the research reactors so far as these spectra were measured with a NaI(Tl) scintillation spectrometer from the comparison of characteristic patterns of photon spectra of YAYOI with those of KUR. Photons originated in the reactor might be mainly composed of photons associated with prompt fission of ^{235}U and those resulting from thermal neutron capture by structural materials of the core and the shield.

Figure 3 is one of the photon spectra obtained at the place of about 100 meter apart from the PWR core. Generally the maximum energy of photons observed in the natural environment is clearly 2.6 MeV. However, the spectral distribution of photons was extended from 2.6 to about 8 MeV. Other measurements of exposure rate with an ionization chamber (15 l, air filled at 1 atm.) at the same point showed that it kept almost constant value regardless of the increase of the reactor power. In spite of that the spectral distribution above 2.6 MeV was weak, this appearance of spectrum above 2.6 MeV was attributable to reactor operation.

Other spectrum component in the region above 2.6 MeV except cosmic ray contribution were not found in the spectrum observed at the place of about 900 meter apart from the reactor on the leeward.

The spectra in higher energy region around research reactor and in PWR site were clearly different from those in the natural environment. In the present study the spectral distribution of photons above 2.6 MeV was clearly found around reactors in operation. Therefore, the present method would be effectively applicable to the environmental surveillance

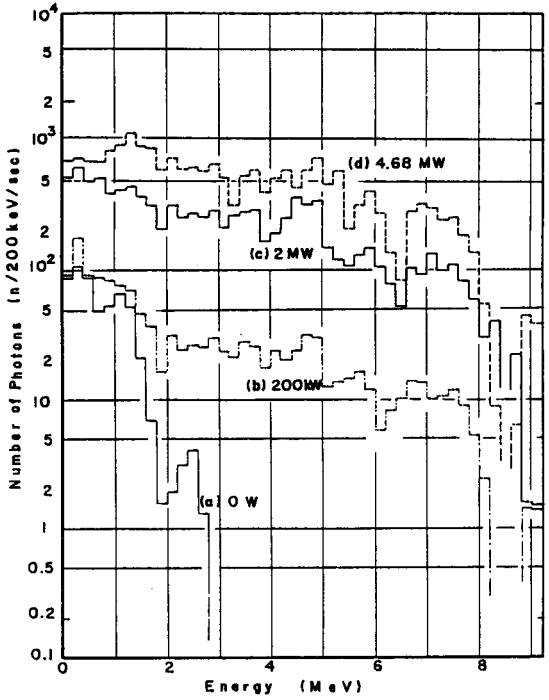

Fig. 2 Photon energy spectra for various level of reactor power of KUR.

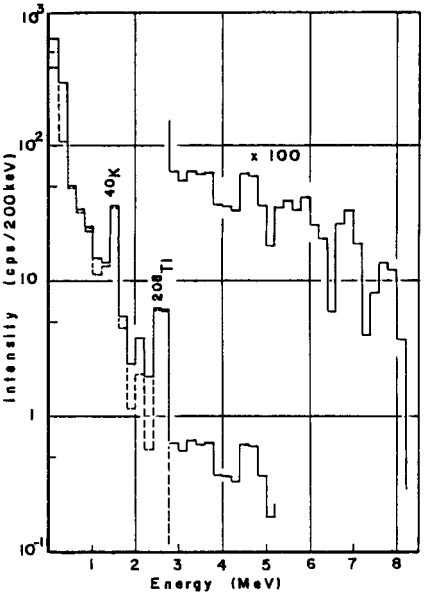

Fig. 3 Photon energy spectra at 0 W (dotted) and 60 % of full power (solid) at a PWR site.

around nuclear reactors.

The successive iteration technique has been widely used, but fails to take into account the statistical errors of experimental measurements. To examine the errors, comparison of the product of the unfolded photon spectrum and the response matrix with the corresponding observed pulse-height distribution were made. The results showed that relative differences of the two for each histogram were below 0.1 % at most.

The ICRP has pointed out in the former recommendation; "any departure from the environmental condition in which man has evolved may entail a risk of deleterious effects." This reflects a prudent attitude of ICRP's in the early stage. An integrated dose or dose equivalent of workers' exposure have been recorded for the purpose of radiological protection except doses both due to natural background radiations and due to medical procedures.

In the growing use of nuclear energy, it is basically important to record the differences from natural environment. The author thinks that the spectrum in the field of interest gives useful information, because we can know, based on it, not only exposure but also the extent of the differences from the natural environment being considered as a standard of comparison for exposures to man-made sources. Therefore, in the radiation protection control of reasonably high quality in future it is suggested that the practical record of exposure, occupational workers being recieved inside the facilities, should be accompanied with a set of

spectra, if the spectrum in a working field is significantly different from that in the natural environment. This might be prudent attitude for worker's exposure in future reassessment of dose, if necessary.

4. CONCLUSION

In order to clarify the radiation field around nuclear reactors, the measurement method was developed using a NaI(Tl) scintillation spectrometer. The method involved the unfolding technique by successive iteration usingthe response matrix constituted in the present study covering photon energy up to about 9 MeV.

Photon energy spectra were obtained in the radiation fields around research reactors, as well as in PWR site. The spectra were distinctly different from those in natural environment. The photon spectra up to about 9 MeV were attributable to operation of the reactors.

It was suggested that in the radiation protection control in future the practical record of exposure should be accompanied with a set of spectra, provided the spectrum of interest is significantly different from that in natural environment.

REFERENCES

1) J. F. Mollenauer: UCRL-9748 (1961).
2) C. D. Zerby, et al.,: ORNL-3169 (1961).
3) M. Nakata: Personal Communication.
4) M. J. Berger, etal.,: Nuclear Instr. Methods, 104, 317 (1972).

THE ASSESSMENT OF RADIATION RISKS IN MIXED HIGH-ENERGY RADIATION FIELDS

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The philosophy behind the early routine dosimetry measurements (pre-1965) made around the CERN accelerators, at that time a 660 MeV proton synchro-cyclotron and a 28 GeV proton synchrotron, was to use radiation detectors whose response in some way parallelled the response of the human body to the stray radiation field (Ref. 1,2). Since it was known that neutrons of energies greater than 1 MeV formed a significant component of the stray field it was natural to use proton recoil counters or to count proton recoils in nuclear emulsions to obtain an estimate of the dose equivalent from this component. Since for higher energy hadrons the main component of dose equivalent came from the inelastic nuclear interactions (stars) in the carbon, nitrogen, etc., of tissue, it was natural to directly relate dose equivalent to the number of stars seen in a nuclear emulsion (Ref. 3) or to the ^{11}C activity induced in a plastic scintillator by inelastic interactions (Ref. 4). A measurement of ionization in a CO_2 ionization chamber, or of blackening of an X-ray film, and an assumption of equilibrium in the radiation field completed the picture with respect to the contributions from direct ionization and enabled an estimate of dose equivalent from all energy deposition mechanisms to be obtained (Ref. 5).

Measurements of absorbed dose were also made with tissue-equivalent chambers. These had two objectives: firstly the quality factors obtained by dividing the dose equivalent by the absorbed dose should be seen to be correctly placed in the range of 1 to 10, given other known physical parameters of the field (e.g. the dominance of muons behind end-stops of low-energy neutrons outside maze entrances). Secondly a measurement of dose by means of a tissue-equivalent chamber when multiplied with a quality factor of about 3 will always give a value of dose equivalent which is accurate to within a factor of 2. Early dosimetry research at CERN tended to concentrate on other ways of determining values of the quality factors, for example from measurements of differential recombination in ionization chambers in order to improve this accuracy (Ref. 6,7).

For technical reasons of sensitivity, reliability and stability of discriminator levels as well as that of international "acceptability", the proton recoil counters were replaced by Andersson-Braun counters, or their ionization chamber version where the pulse structure of the accelerator field made this imperative, for the measurement of dose equivalent from neutrons of several MeV. This carried with it an inherent shift in philosophy since the instrument is based on a BF_3 counter which records the alpha particles produced by the absorption of thermal neutrons in boron. This in no way parallels the mechanisms of dose deposition in tissue but uses calculations of dose equivalent at or near the surface of a tissue-equivalent phantom as a function of neutron energy to define a required instrument response curve, which is then matched by varying the thickness of the polythene moderator or of boron-loaded plastic around the BF_3 counter. Thus the instrument measures the dose equivalent from incident neutrons no matter what the equilibrium condition of the radiation field. This corresponded to the philosophy then in use at accelerator laboratories other than CERN, where activation detectors, multisphere detectors and other forms of particle spectrometers, more or less crude, were used to define the radiation field in terms of fluence quantities. Calculations of dose equivalent as a function of depth in tissue-equivalent phantoms, similar to those referred to above, were used to determine the actual dose equivalent to be attributed to these spectra (Ref. 8). At higher particle energies, however, the maximum of the depth-dose-equivalent curves for monoenergetic particles is not at or near the surface of the body. This led to severe confusion in the determination of dose equivalent in spectra of particles which was not helped at all by the recommendations of ICRP Publications 4 and 21 (Ref. 9,10) or

by the ICRU explanations of their dose-equivalent index (Ref. 11). It was evident that for most spectra around a high-energy accelerator the maximum of the depth-dose curve in a phantom exposure is to be found at a depth of about 1 $g.cm^{-2}$ (Ref. 12).

It has recently been shown that the dose equivalent to be attributed to ^{11}C activity in a plastic scintillator is close to the value originally accepted at CERN for pure neutron spectra to be found outside accelerator shields (close to the original calibration philosophy, Ref. 13). This may have to be modified slightly if theoretical predictions of a significant pion component of the field at energies between 20 and several 100 MeV prove to be correct. A typical (theoretical) neutron spectrum is shown in Fig. 1 (Ref. 14). Below 1 MeV the spectrum has essentially the 1/E form. Above this energy the peaks from evaporation and the intranuclear cascade can be seen. Also shown are cascade spectra of neutrons, protons and charged pions having energies above 20 MeV, normalized by equating the peak values of the neutron spectrum to the first-mentioned calculations (Ref. 13). Figure 2 shows the dose equivalent at a depth of 1 cm in tissue for monoenergetic neutrons, protons and pions without the contribution from primary ionization for protons and pions (Ref. 13). When the spectra of Fig. 1 are folded with these dose-equivalent curves, then it is found that pions contribute about one half of the dose equivalent from incident particles with energies above 20 MeV.

In routine dosimetry one is always faced with the same problem. The techniques used to determine fluence quantities are robust, of high sensitivity and easy to use in an accelerator environment. The techniques based on measuring the interactions in the human body, e.g. LET or event-size spectrometry, may be satisfactory laboratory techniques but generally do not have the required sensitivity for use in occupational radiation protection levels, nor can they cope satisfactorily with the pulsed nature of most stray fields around high-energy proton accelerators. The principal way of determining dose equivalent at CERN is still based on the techniques mentioned above. The multi-detector set consists of a REM/ION Chamber, an air-filled ionization chamber and a tissue-equivalent chamber used with a measurement of ^{11}C activation in a plastic scintillator. This is called the CERBERUS system (Ref. 15). The validity of this system is supported by absolute measurements of the energy response of the various detectors (Ref. 16). In addition, intercomparisons between the CERBERUS system and other methods of assessing dose equivalent, although restricted to experimental non-routine situations, have shown good correspondence between the results obtained (Ref. 17). Routine measurements of the stray radiation field of the CERN accelerators are based on single detector systems, i.e. a single hydrogen-filled ionization chamber or REM/ION chamber. Since the energy spectrum and composition of the field remain fairly stable in time, it is possible to "calibrate" the single detector using the CERBERUS system in the field to determine an appropriate field-quality factor for each detector.

CONCLUSIONS

The risk estimate for persons exposed to mixed radiation fields at CERN is based on a detailed knowledge of the radiation field and of the detectors used, coupled with the results from Monte Carlo calculations of dose equivalent as a function of depth in tissue-equivalent phantoms. Recommendations of ICRP and ICRU have never been slavishly followed, e.g. the conversion factors at high energies of ICRP Publication 4 or the detailed application of the Index Quantities of ICRU, but the philosophy has always been to obtain values of dose equivalent which will overestimate the risk. The introduction of Effective Dose-equivalent in ICRP Publication 26 has clarified which dose equivalent is appropriate to risk estimation, but so far no account has been taken in routine surveys at CERN of the isotropy factors which implies a safety factor of two in the assessment of dose equivalent from low-energy neutrons (Ref. 18).

REFERENCES

1. J. Baarli, Radiation dosimetry and personal radiation exposure at CERN, CERN Internal Report DI/HP/8 (1962).
2. J. Baarli and A.H. Sullivan, Radiation dosimetry for protection purposes near high-energy particle accelerators, Health Physics 11, 353 (1965).
3. J. Baarli and J. Dutrannois, The nuclear stars in personal neutron track films carried at CERN, Proc. Symp. on Personnel Dosimetry Techniques for External Radiation, ENEA, Madrid (1963), p. 283.
4. K.B. Shaw, High energy flux measurement using plastic scintillators, CERN Internal Report DI/HP/6 (1962).
5. K. Goebel, A. Rindi, A.H. Sullivan and J. Baarli, The purpose, interpretation and utilization of area monitoring measurements near the CERN accelerators, in Radiation Dose Measurements, their Purpose, Interpretation and Required Accuracy in Radiological Protection, ENEA, Paris, CONF-670621 (1967), p.435.
6. A.H. Sullivan and J. Baarli, An ionization chamber for the estimation of the biological effectiveness of radiation, CERN Report 63-17 (1963).
7. A.H. Sullivan, An approach to a Rem-dosimeter using ionization chambers, CERN Internal Report DI/HP/29 (1964).
8. K.B. Shaw, G.R. Stevenson and R.H. Thomas, Evaluation of dose equivalent from neutron energy spectra, Health Physics 17, 459 (1969).
9. International Commission on Radiological Protection, Protection against electromagnetic radiation above 3 MeV and electrons, neutrons and protons, ICRP Publication 4, Pergamon Press, Oxford (1964).
10. International Commission on Radiological Protection,'Data for protection against ionizing radiation from external sources - Supplement to ICRP Publication 15', ICRP Publication 21, Pergamon Press, Oxford (1973
11. International Commission of Radiation Units and Measurements, 'Dose-equivalent - Supplement to ICRU Report 19', ICRU Report 19S, Washington D.C. (1973).
12. G.R. Stevenson, M. Höfert, J. Neufeld, A. Rindi, J.T. Routti and S. Prêtre, Standardizing the fluence-to-dose-equivalent conversion factors for whole-body neutron exposures, in Proc. Symp. on Neutron Monitoring for Radiation Protection Purposes, IAEA, Vienna (1973), Vol. 1, p. 177.
13. G.R. Stevenson, The estimation of dose equivalent from the activation of plastic scintillators, CERN Preprint TIS-RP/107 (1983).
14. K. O'Brien, Neutron spectra in the side-shielding of a large particle accelerator, Health and Safety Laboratory Report, HASL-240 (1971).
15. M. Höfert, A comparison of dose equivalent measurements around a GeV proton accelerator, CERN Internal Report DI/HP/187.
16. M. Höfert and R.C. Raffnsøe, Measurement of absolute absorbed dose and dose-equivalent response for instruments used around high-energy proton accelerators, Nucl. Instrum. Meth. 176, 443 (1980).
17. A.V. Antipov, I.S. Bajshev, V.T. Golovachik, G.I. Krupnyi, V.N. Kustarev, V.N. Lebedev and M. Höfert, Comparison of dose equivalent measurements behind the IHEP accelerator shielding using different methods, Atomic Energy 46, 105 (1979). CERN Translation 78-01 (1978).
18. G. Burger, A. Morhart, P.S. Nagarajan and A. Wittman, Effective dose equivalent and its relationship to operational quantities for neutrons Paper presented at the European Seminar on Radiation Protection Quantities for External Exposure, 13-15 October, Braunschweig, EUR 7101EN (1980).

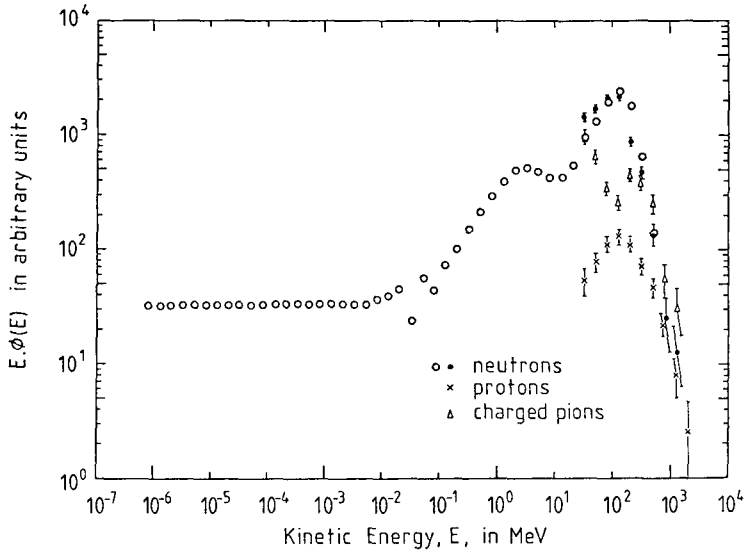

Fig. 1. Theoretical neutron spectrum in the side-shielding of a high-energy proton accelerator (Ref. 14), open circles, compared with high-energy spectra (Ref. 13).

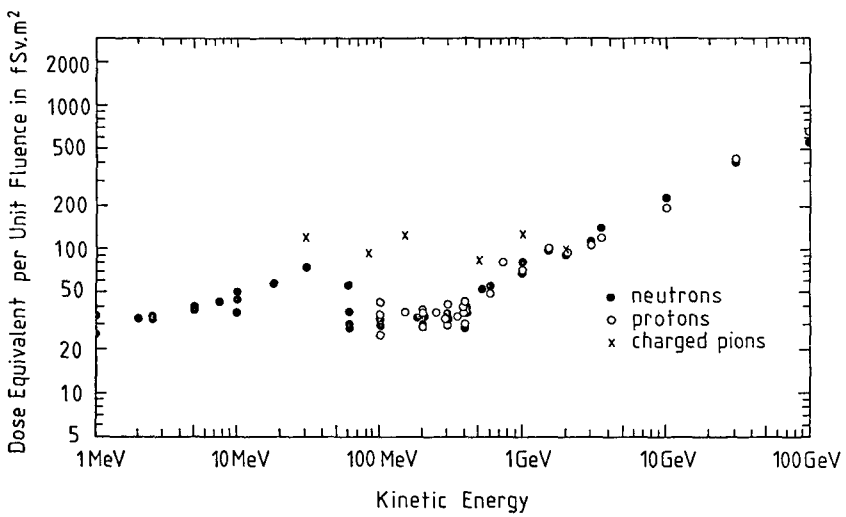

Fig. 2. Dose equivalent at a depth of 1 cm in tissue-equivalent material for monoenergetic neutrons and for protons and pions without the contribution from primary ionization (Ref. 13).

A SKYSHINE BENCHMARK EXPERIMENT

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I. INTRODUCTION

The requirement for the calculation code accurately predicting the skyshine radiation from nuclear facilities has been recently raised with the necessity of shielding design appropriate to the reduction of dose objective to the public. Then many calculation codes have been developed[1,2,3] and benchmark experiments have been performed to make sure of the accuracies for such calculation codes[4,5]. The present report describes a skyshine benchmark experiment with the ^{60}Co source distributed in the reactor pressure vessel wall which was activated by neutrons during reactor operation. The γ-rays were emitted upward, and the energy spectrum and the exposure rate up to 300 m from the source were measured with a NaI(Tℓ) spectrometer and a high pressure ionization chamber, respectively. In addition, the vertical distribution of exposure rate up to 500 m of height was measured with a small ionization chamber. On the other hand, the two-dimensional discrete ordinates transport calculation was performed, and consequently compared with the experimental results.

II. EXPERIMENT

Operation of the JPDR, Japan Power Demonstration Reactor, of JAERI has been stopped for several years, however, a fairly large amount of radioactivities induced by neutrons remains in the reactor pressure vessel wall. The principal nuclide is ^{60}Co and it has enough γ-ray strength to perform the present experiment. The schematic view of the JPDR is given in Fig. 1. Usually, the reactor pressure vessel is filled up with water for radiation shielding. In this experiment, the γ-rays were emitted upward and collimated into about 30-deg. conical beam by lowering the water and taking off the shielding plugs.

The energy spectrum and the exposure rate were measured with a spherical NaI(Tℓ) spectrometer and a high pressure ionization chamber at a height of 1.5 m above the ground. Measuring points were located at 15, 50, 100, 200 and 300 m on a straight line from the center of the reactor core. The sphericity of the NaI(Tℓ) detector provides for a flat

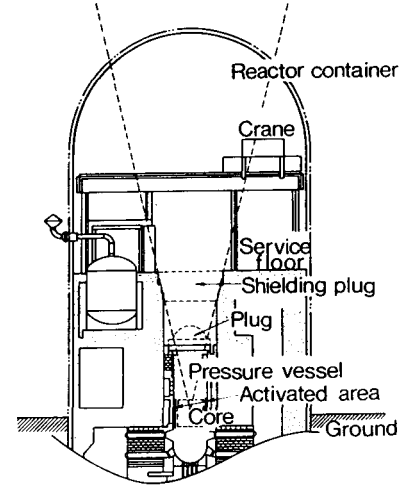

Fig.1 Schematic view of JPDR used as skyshine source

directional sensitivity within 3.5% except for the bottom part. Measured spectra were unfolded with a set of response matrix and converted to the values in exposure rate unit. The high pressure ionization chamber has a spherical stainless steel shell, in which argon gas is filled at 8 atm. The spherical shape provides for the isotropic sensitivity except for the bottom part due to shielding by the electrometer.

A vertical distribution of exposure rate was measured with a small ionization chamber packed with a recorder in a box which was lifted by a 70 m^3 kite - balloon at heights of about 50, 100, 150, 200, 300, 400 and 500 m above a point positioned about 100 m from the reactor core. The ionization chamber has a cylindrical shape and a flat energy response more than 60 keV. Two transits observed exact locations of the ionization chamber in each measurement.

III. CALCULATION

The discrete ordinates transport Sn calculation was performed in two stages with the DOT3.5 code. In the first stage, the photon flux inside the reactor container was calculated and then converted into a virtual anisotropic point source on the service floor after normalization to a measured value in terms of exposure rate. The source area in the reactor pressure vessel was specially divided into small meshes. A set of asymmetric Sn constants (166 angles) was used. In the next stage, the skyshine calculation outside the reactor container was performed with the source defined in the previous stage and a set of symmetric S_8 constants (48 angles) was used. In each stage, the photon energy was divided into 11 groups.

IV. RESULTS

The unfolded spectral shapes have no complicated structure and no significant difference each other for all horizontal locations. The spectrum consists of two parts, one has a peak at ∿70 keV and the other at ∿200 keV. The former becomes increasingly shifted to lower energy and the latter to higher energy being inconspicuous with increasing source-to-detector distance. It can be said from brief spectral analysis that the former is due to the photons multiply scattered by air and the latter to the photons scattered by the media nearby the source; e.g. the reactor container and crane in it. From the comparisons of the calculated and measured spectra, it is shown that the accordance between each spectrum becomes to be good

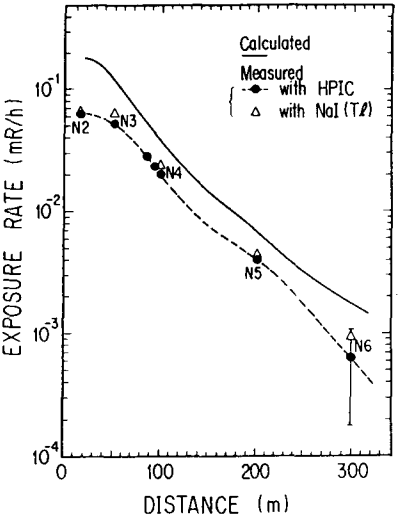

Fig.2 Comparison of calculated and measured horizontal distribution of exposure rate

with increasing source-to-detector distance.

The horizontal distribution of exposure rates decreases in a manner of exponential function with increasing source-to-detector distance in both cases of measurement and calculation, as shown in Fig. 2. From the comparison of the calculated and measured distributions, it is found that the DOT calculation overpredicts the exposure rates by 35 % for the NaI(Tℓ) detector and 50 % for the high pressure ionization chamber. As mentioned before, the virtual point source was used in skyshine calculation instead of the actual source distributed in the reactor pressure vessel wall. Nevertheless, the difference between each result is not so great.

The vertical distributions of exposure rates obtained by measurement and calculation are shown in Fig. 3 as a function of height. The exposure rate decreases with increasing height and its manner is gentle as compared with the horizontal case. From the comparison, the DOT calculation underestimates the exposure rates at all heights more than 100 m. However, the agreement between each result is generally good.

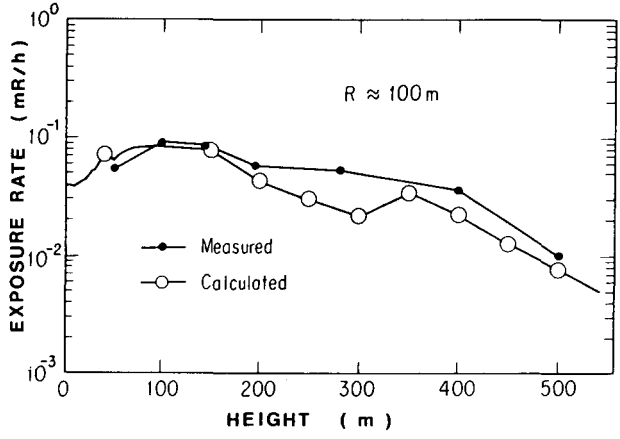

Fig.3 Comparison of calculated and measured vertical distribution of exposure rate. R is distance from reactor axis.

V. CONCLUSION

A skyshine benchmark experiment was carried out in order to make sure of the accuracies for the DOT code and the calculation gave the agreement with the measurement within 50 % in both horizontal and vertical directions. It was shown from this experiment that the DOT calculation predicts appropriately the skyshine exposure rates.

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REFERENCES

(1) M.L. Couchman and G.H. Anno, NUS-TM-NA-42 (November, 1965)
(2) W.A. Rhoades and F.R. Mynatt, ORNL-TM-4280 (September, 1972)
(3) H. Ryufuku, T. Numakunai, S. Miyasaka and K. Minami, JAERI-M-8171 (March, 1979)

(4) R.R. Nason, J.K. Shultis, R.E. Faw and C.E. Clifford, Nucl. Sci. Eng. 79, 404 (1981).
(5) Y. Yamaguchi, H. Ryufuku, Ke. Minami, T. Numakunai and Ka. Minami, Proceedings of Sixth International Conference on Radiation Shielding (May, 1983).

COMPARATIVE STUDY OF RADIATION PROTECTION QUANTITIES FOR NEUTRONS

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INTRODUCTION

Recently, the specified depth dose equivalent H_d has been proposed as the operational dose equivalent quantity for radiation protection measurements for photon, beta and neutron radiation /1/. $H_d(\vartheta)$ can be defined as the dose equivalent at a point at a depth d below the surface of the ICRU spherical phantom. If the point is located on the "reference radius" of the sphere, the angle between the incident monodirectional broad radiation field and the reference radius is termed ϑ.

The aim of this paper is as follows:

a) Partly to present our calculations of H_d as a function of d, ϑ, and the neutron energy E

b) to compare the results with other dose equivalent quantities such as the maximum dose equivalent $\hat{H}$ defined as an energy dependent fluence-to-dose conversion function (FDCF) in ICRU Report 21 /2/, or the effective dose equivalent H_{eff}/3/.

After comparing the energy and directional dependence of H_d and H_{eff} we shall come to the conclusion that H_d is to be questioned as an operational dose equivalent quantity for all exposure situations met with in individual neutron monitoring, since the corresponding ratio of fluence-to-dose conversion factors, $(FDCF)_d/(FDCF)_{eff}$, varies as a function of neutron direction and energy between 3 and 1/336.

c) We therefore propose to extend the concept of H_d by defining an additive dose equivalent quantity

$$H_{sph} = \sum_i g^i H^i \qquad (1)$$

where the upper index i = 1,2,3,.... and where the H^i represent a suitably chosen set of dose equivalent quantities defined within the ICRU sphere. The g^i are suitably chosen constant weighting factors. The ratio of $(FDCF)_{sph}$ to $(FDCF)_{eff}$ varies only between 2 and 1 if the set H comprises five different dose equivalent quantities.

SELECTED RESULTS OF THE H_d CALCULATIONS

The second to fourth columns in the table show $H_d(\vartheta)$ normalized to the fluence of incident neutrons, in units of 10^{-12} Sv cm as a function of neutron energy in eV. A vast body of such data has been computed by means of a recently developed Monte Carlo code and is presented in more detail elsewhere /4/. Here, only the d = 10 mm data are given for ϑ = 0° (frontal or AP incidence), ϑ = 180° (back or PA incidence), and for an isotropic radiation field (ISO).

COMPARISON WITH OTHER DOSE EQUIVALENT QUANTITIES AND DISCUSSION

The remaining columns in the table show $\hat{H}$ as mentioned in Sect. 1 and H_{eff} under the exposure conditions AP (monodirectional broad beam on the front), PA (same on the back), ROT (same with individual continuously rotating about the axis defined by feet and head) and ISO taken from Ref. /5/.

Before comparing and discussing these quantities, calibration procedures for individual dosemeters are briefly recapitulated. The

specified-depth dose equivalent H_d is chosen as an example of an operational dose equivalent quantity in the following. The calibration comprises two steps, as shown in Fig. 1. In step (a) the quantity H_d (normalized to the monitor reading) is measured. In step (b) the ICRU sphere is replaced by the anthropomorphic phantom which together with the individual dosemeter forms the "measuring device". If M is the reading of the individual dosemeter (as normalized to the monitor reading), then the calibration factor is generally obtained from $k = H/M$.

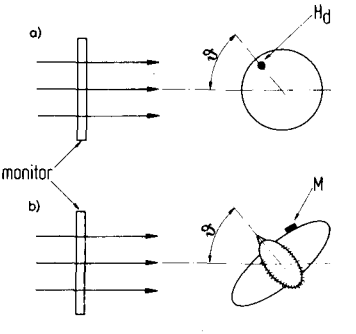

Fig. 1: Experimental arrangement for calibrating individual dosemeters

There are, however, two independent variables which have to be considered: The energy, E, and the direction of incidence, ϑ. An ideal individual dosemeter should satisfy the requirement $k = k(E,\vartheta) = \text{const.}$ for all energies E and all directions ϑ. These calibration problems, particularly if more than one dosemeter is worn by the individual have been discussed in a much more general way elsewhere /6,7/. Here, it is sufficient to emphasize that an operational dose equivalent quantity depends on the energy and the direction of incidence. These dependences for H_{10} are (with respect to the direction only for $\vartheta = 0°$, 180°) given in the second and third column of the table. If H_{10} is to be an adequate operational quantity, it should have a dependence on E and ϑ roughly similar at least to that of H_{eff}.

Considering the energy dependences shown in the table, H_{10} and $\hat{H}$ can be made conservative estimates of H_{eff}-AP by multiplying them by 1.1 and 1.15, respectively. One then obtains a maximal overresponse of 3.16 and 2.81 for H_{10} and $\hat{H}$, respectively. Such factors may be acceptable in routine area monitoring. For the purpose of discussing individual monitoring, one should note that at E =100 keV

$$\frac{H_{10}(0^\circ)}{H_{eff\text{-}AP}} = \frac{67.1}{23.7} = 2.83 \qquad (2a)$$

Regarding the directional dependence now by comparing the second and seventh column of the table, one finds $H_{10}(180°)$ to be very much lower than H_{eff}-PA. For E = 1 MeV one finds

$$\frac{H_{10}(180^\circ)}{H_{eff\text{-}PA}} = \frac{.253}{85.0} = \frac{1}{336} \qquad (2b)$$

Discussing the results of Eqs. (2a, b), H_{10} would give a considerable overall overestimation for exposure of the front side, but a very high underestimation for exposure of the back. We therefore conclude that H_{10} is not an adequate operational dose equivalent quantity for all exposure conditions in individual neutron monitoring.

AN IMPROVED OPERATIONAL QUANTITY H_{sph}

In the preliminary attempt the directional dependence of H_{sph} (see Eq. (1)) was only considered for the directions $\vartheta = 0°$ and $\vartheta = 180°$ of the incident radiation. The reference radius (see Sect. 1) defines a spherical polar coordinate system r,α,β fixed in

the ICRU sphere with the polar angle α ($0 \leq \alpha \leq 180°$) and the azimuthal angle β ($0° \leq \beta \leq 360°$). The preliminary choice of the dose equivalent quantities H^i in Eq. (1) is given by the following volume elements in the ICRU-sphere: H^i corresponds to a point-like volume element at $\alpha = 0°$ and $r = 14$ cm, and this is identical with H_{10} on the reference axis. H^2 is the average in an extended volume element given by 11 cm $\leq r \leq$ 12 cm and $0° \leq \alpha \leq 45°$, H^3 is the average on a thin sperical shell at $r = 14$ cm, H^4 is the average in 11 cm $\leq r \leq$ 12 cm and $135° \leq \alpha \leq 180°$, and H^5 corresponds again to a point-like volume element at $\alpha = 180°$ and $r = 14$ cm. The corresponding factors g^i in Eq. (1) must be so chosen that the energy and directional dependence of H_{sph} match the corresponding dependence of H_{eff}. A preliminary result obtained by a simple trial and error procedure yields $g^1 = 0.13$, $g^2 = 1.14$, $g^3 = 0.16$, $g^4 = 0.57$ and $g^5 = 0.08$. The ratio $H_{sph} : H_{eff}$ at $\vartheta = 0°$, 180° and all energies listed in the table then satisfy the condition

$$1.0 \leq \frac{H_{sph}}{H_{eff}} \leq 2.1 \qquad (3)$$

This preliminary finding appears to be a remarkable progress in comparison with the poor properties of H_d demonstrated in eqs. (2a, b).

CONCLUSION

The preliminary character of the results in Sect. 4 is again emphasized. In particular it can be expected that in future investigations the extended volume elements for some of the H^i can be replaced by suitably located point-like volume elements. Further, linear programming could be used to obtain an optimal matching of H_{sph} and H_{eff}. It can be expected that an operational quantity could be constructed in the ICRU-sphere which not only has an improved energy dependence for area monitoring, but is also suitable for individual monitoring, since the directional dependence is greatly improved. The price to be paid for this is that one would have to deal with a few points distributed over the volume of the ICRU sphere instead of one single point as in the case of H_d.

From the standpoint of neutron dosimetry, the experiment (a) in Fig. 1 cannot be performed, because no instrument is at present available to measure the dose equivalent. Since this experiment must in any case be replaced by Monte Carlo calculations, H_{eff} as calculated approximately in a standardized anthropomorphic phantom, or any related quantity /7/ would represent an excellent reference dose equivalent quantity. However, the dose equivalent distributions in anthropomorphic phantoms are much more difficult to calculate than in a uniform sphere and the likelehood of errors and uncertainties increases. The authors therefore advocate a solution based on eq. (1).

As for photons, H_{eff} has been rejected on the grounds that it is not measurable. Taking this objection into account, it might be expected that an operational quantity based on eq. (1) turns out to be a reasonable compromise.

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REFERENCES

/1/ Burlin, T.E. "Practical Determination of Dose Equivalent", Proceedings of the 7th Int. Congress of Rad. Res., Amsterdam, July 3 - 8, 1983, Paper E1-20, Martinus Nijhoff, Amsterdam, 1983

/2/ ICRP, Publication 21, Pergamon Press, Oxford, New York, Toronto, Sydney, Braunschweig, 1973

/3/ ICRP, Publication 26, Pergamon Press, Oxford, New York, Frankfurt, 1977

/4/ Hollnagel, R; Jahr, R.; Siebert, B.R.L. "Dosimetric Quantities in the ICRU Sphere for Neutron Irradiation with Energies between thermal and 20 MeV", Report PTB-FMRB-101 (1983)

/5/ Burger, G.; Morhart, A.; Nagarajan, P. and Wittmann, A. GSF, München-Neuherberg (FRG), private communication, and 4th Sympos. on Neutron Dosimetry, EUR 7448, Vol. I, p. 33 (1981)

/6/ Siebert, B.R.L.; Hollnagel, R. and Jahr, R. "Theoretical Concept for Measuring Doses from External Radiation Sources in Radiation Protection", Phys. Med. Biol. 28 (1983), p. 521 - 533

/7/ Jahr, R.; Hollnagel, R. and Siebert, B.R.L. "A Conceptual Physical Basis for Monitoring External Radiation", Rad. Protection Dosim. 1 (1981) p. 299 - 304 (Erratum: Rad. Prot. Dos. 2 (1982) 59)

Comparison of Radiation Protection Quantities

E_n	$H_{10}(0^\circ)$	$H_{10}(180^\circ)$	H_{10}-ISO	$\hat{H}$	H_{eff}-AP	H_{eff}-PA	H_{eff}-ROT	H_{eff}-ISO
2.50E+02	8.240	0.055	1.813	10.700	5.480	3.150	3.140	2.450
1.00E+00	10.340	0.123	2.895	10.550	6.940	4.750	4.350	3.330
1.00E+01	7.370	0.100	2.506	10.460	6.980	5.260	4.600	3.510
1.00E+02	6.740	0.182	2.224	10.370	6.740	5.430	4.600	3.500
5.00E+02	5.730	0.117	1.891	10.310	6.290	5.330	4.410	3.350
1.00E+03	5.810	0.146	1.801	10.280	6.110	5.280	4.330	3.280
2.00E+03	5.730	0.146	1.776	9.430	6.210	5.310	4.380	3.320
3.00E+03	6.620	0.195	1.787	9.140	6.280	5.330	4.410	3.350
6.00E+03	6.900	0.117	2.001	9.070	6.380	5.360	4.460	3.390
8.00E+03	7.600	0.125	2.280	9.330	6.430	5.370	4.480	3.410
1.00E+04	8.290	0.212	2.321	9.920	6.460	5.380	4.500	3.420
2.00E+04	13.350	0.107	3.605	16.340	8.990	6.160	5.670	4.370
3.00E+04	19.640	0.159	5.106	22.080	10.890	6.760	6.480	5.040
5.00E+04	32.330	0.206	8.729	32.680	13.900	7.370	7.680	6.030
1.00E+05	67.110	0.185	20.133	57.870	23.700	10.800	12.200	9.660
2.00E+05	128.800	0.283	38.640	99.030	44.670	19.400	22.630	17.900
3.00E+05	164.540	0.200	55.944	135.870	65.950	28.700	33.510	26.400
4.00E+05	221.680	0.186	70.933	168.380	86.790	37.900	44.180	34.700
5.00E+05	272.330	0.200	84.485	198.410	107.000	46.800	54.400	42.700
6.00E+05	257.630	0.264	95.323	230.450	125.720	55.100	63.760	50.000
8.00E+05	331.980	0.268	114.752	284.980	157.440	70.600	81.920	62.900
1.00E+06	375.040	0.253	131.264	326.800	179.000	85.000	94.000	73.300
1.50E+06	338.950	0.903	162.696	376.480	200.770	118.800	118.900	92.500
2.00E+06	352.090	1.626	186.608	396.830	217.800	150.600	140.600	109.000
2.50E+06	352.100	14.203	193.655	405.910	232.000	181.000	160.000	124.000
3.00E+06	402.200	13.850	209.144	409.520	262.100	211.900	184.900	143.000
5.00E+06	439.550	39.267	259.016	408.500	358.000	310.000	270.000	212.000
5.12E+06	451.570	19.990	243.848	408.440	359.500	313.500	273.000	215.000
5.24E+06	360.120	45.798	247.078	408.390	360.890	317.000	276.000	217.000
7.50E+06	439.270	57.723	303.096	407.610	343.800	334.000	292.800	237.600
1.00E+07	492.800	77.121	325.248	408.500	348.000	336.000	302.000	249.000
1.20E+07	528.700	95.953	364.803	412.220	419.900	394.900	356.900	294.000
1.40E+07	561.500	117.469	398.665	416.190	478.200	442.300	400.400	330.000
1.60E+07	575.260	136.967	425.692	420.100	-	-	-	-
1.80E+07	576.570	164.734	438.193	423.840	-	-	-	-
2.00E+07	642.090	181.381	500.830	427.350	-	-	-	-
CF252	350.530	9.680	164.749	340.000	-	-	-	-

ALPHA SOURCE MICRODOSIMETRIC PARAMETERS

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Conventional proportional counters used in microdosimetry are not adapted to alphaparticles microdosimetry, partially because of the distrotion of the E spectrum produced by the interaction of the radiation with the counters' walls. A strictly wall-less proportional counter was therefore developed for the determination of the microdosimetric parameters of alpha-particles. The first results obtained with different alpha-emitters are reported and compared with those obtained with grid counters.

An attempt is also made to develop a theoretical calculation of microdosimetric parameters at cellular and subcellular level.

The measurements and calculations are carried out in view of a comparison with the experimentally determined radiation damage caused to biological macromolecules in eucaryotic cells in culture (DNA strand-breaks, formation of DNA-protein cross-links, etc. ...).

UNIVERSAL BETA-GAMMA-NEUTRON ALBEDO DOSEMETER FOR COMMERCIAL TLD CARDS AND TRACK ETCH DETECTORS

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1. INTRODUCTION

With respect to a broad application of neutron dosemeters in personnel monitoring, there is the need to standardize neutron dosemeters with respect to the detector type, the design of the dosemeter encapsulation, the read-out procedure as well as the calibration technique applied. The universal dosemeter encapsulation recently developed at Karlsruhe [1,2] allows to combine TLD cards with track etch detectors and to measure simultaneously the dose equivalent of beta rays, gamma rays and neutrons. The unified boron-loaded plastic encapsulation is applicable as a two-component albedo dosemeter for most of the commercial TLD systems with automatic read-out (Fig. 1). For a large scale use of TLDs and track etch detectors standardized evaluation techniques are available [3].

2. DOSEMETER DESIGN

The characteristic design of the boron-loaded plastic encapsulation of the size 35 x 54 x 11 mm^3 is the 'beta window' (detector position a) on the side facing the source and the collimating albedo neutron window (detector position i) on the side facing the body (Fig. 2 and Fig. 3). For the TLD systems of interest, the windows are fitted in size and position. The universal encapsulation can be used for two pairs of $^{6}LiF/^{7}LiF$ or $^{6}Li_2{}^{10}B_4O_7/^{7}Li_2{}^{11}B_4O_7$ detectors in TLD cards or holders for a full automatic read-out. The TLD card can be combined with polycarbonate and CR 39

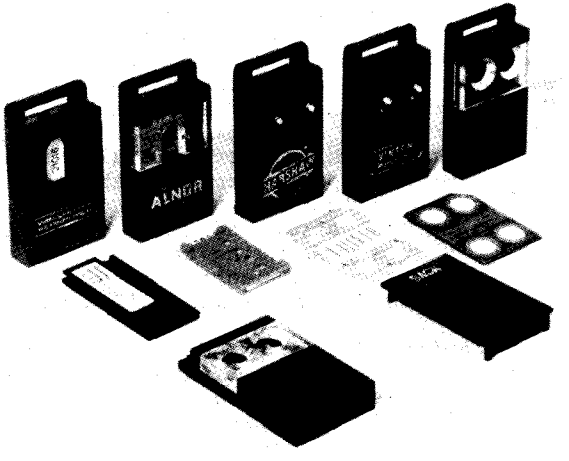

Fig. 1: Universal beta-gamma-neutron albedo dosemeters for different commercial TLD system

track etch detectors and/or (n,α)-converters in contact with track etch detectors instead of TLDs. The neutron insensitive detectors in the positions a and i provide the measurement of skin dose and whole body dose for beta rays and photons, and the neutron sensitive detectors in the position a and i the indication of thermal field and thermal albedo neutrons.

An optional magnetic latch guarantees an authorized opening of the encapsulation only. The universal albedo dosemeter is commercially available for TLD systems of the manufacturers Alnor, Harshaw, Panasonic, Vinten, and Teledyne.

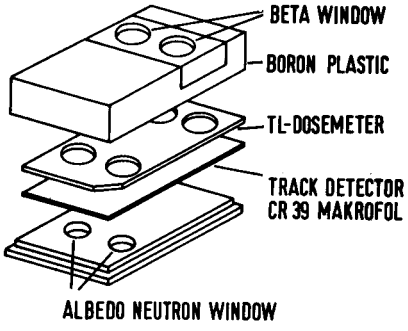

Fig. 2: Combination of TLD's and track etch detectors in the universal dosemeter

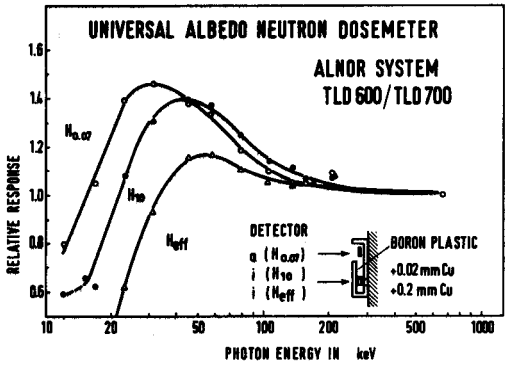

Fig. 3: Photon energy response of the boron-plastic capsule with copper filters

3. PROPERTIES OF THE DOSEMETER

In stray neutron fields the two-component albedo dosemeter has been found to be an appropriate personnel dosemeter which allows to correct itself for location dependent variations of the albedo response using the actual read-out ratio of the albedo (i) and the thermal (a) neutron detector. This technique is verified by experimental results of field calibrations. A computer assisted evaluation technique is routinely applied at Karlsruhe, which performes all data processing including individual detector response factors, the separation of the neutron and gamma dose contributions and the estimation of random uncertainties for the measured neutron and gamma dose equivalent. By using stored calibration curves presenting the response vs. reading ratio for each facility, the apparent neutron dose is multiplied with the actual neutron calibration factor immediately after read-out.

In the dosemeter, neutron track etch detectors are used as a supplement to TLDs in order to indicate separately the neutron dose equivalent due to fast neutrons and in addition to improve the intrinsic correction of the albedo response and thus location dependent changes of the detector response. (n,α) radiators in contact with Makrofol or CR 39 may extent the dose range up to 10 μSv in order to measure low doses or neutrons in high gamma fields. For routine

monitoring, however, only a combination with TLDs offers the dose range of interest.

The photon energy response has been investigated for the Alnor TLD dosemeter with two pairs of TLD600/TLD700 detectors of the size 3 x 3 x 0.9 mm³ inside the boron-loaded plastic encapsulation of 3 mm thickness and additional copper filters (Fig. 3). The dosemeters have been irradiated at the surface of a cylindrical polyethylene phantom of 30 cm diam. The well-known overresponse of LiF detectors of 45 % presented in Fig. 3 for the dose equivalent $H_{0.07}$ in a tissue depth of 0.07 mm can significantly be reduced by additional copper filters which are applied in order to indicate H_{10} or H_{eff}, i.e. the dose equivalent in a tissue depth of 10 mm and the effective dose equivalent, respectively, defined by appropriate conversion factors. In contrast to the indication of $H_{0.07}$ with detector (a), H_{10} has been found to be energy dependent within ± 25 % above 20 keV for detector (i).

The angular response of the neutron detectors in the boron-loaded plastic capsule with an albedo neutron window of 3 mm diam. is presented in Fig. 4a for a distance of 1.25 m from a bare ^{252}Cf source and a height of 1.25 m above floor. Compared to TLDs the reading of track etch detectors decreases rapidly for angles > 30°. The photon angular response is presented in Fig. 4b for various photon energies. The angular response has been found to be higher for low photon energies than for fission neutrons.

The relative standard deviation of the dose measurement calculated from 10 dosemeter readings is presented in Fig. 5 as a function of dose for all neutron detectors of interest. At high track densities the 1 σ-values of electrochemically etched Makrofol polycarbonate and CR 39 manufactured by American Acrylics are practically identical with the counting statistic. The empirical curves presented in Fig. 5 takes into account the correction of the actual background reading of 3 tracks/cm² for Makrofol, 80 tracks/cm² for CR 39 and 0.02 mSv for TLD600. For a bare ^{252}Cf source the lowest detectable

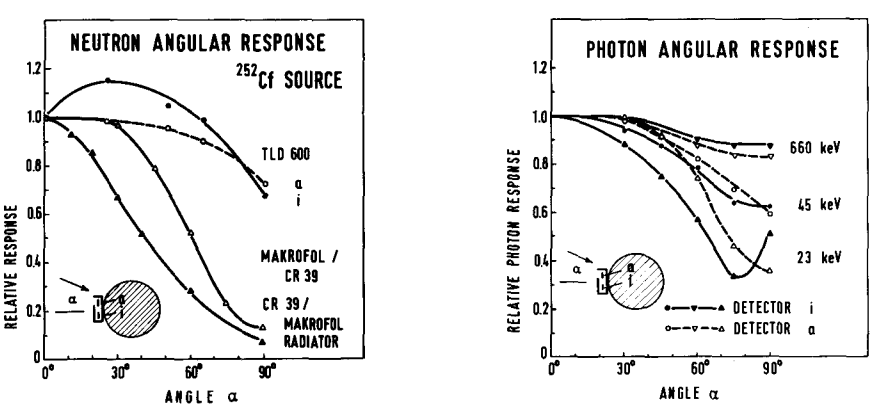

Fig. 4: Neutron and photon angular response of the universal dosemeter

neutron dose (1 σ-value of 50 %) is thus 1 mSv for Makrofol, 0.2 mSv for CR 39, 0.03 mSv for TLD600 and 0.002 mSv for an (n,α) radiator in contact with Makrofol.

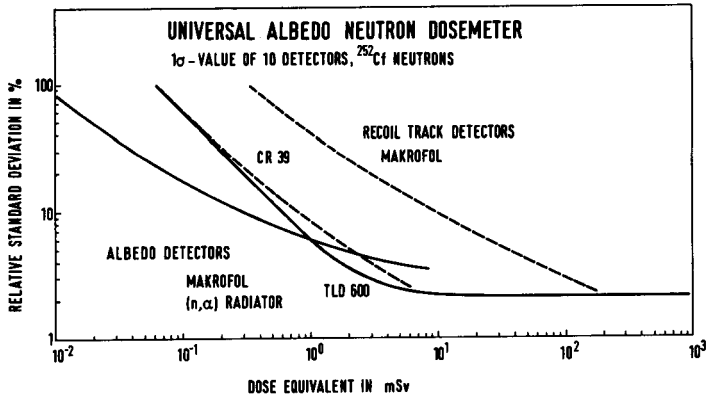

Fig. 5: Relative standard deviation of the dose measurement for different neutron detectors used in the universal albedo neutron dosemeter

4. CONCLUSION

With respect to a large scale use of dosemeters in routine personnel monitoring we standardized the dosemeter encapsulation, the computer assisted evaluation technique for TLDs and the etching and counting technique for track etch detectors. A commercially available high voltage generator and etching cells allows the simultaneous electrochemical etching of 20 foils, i.e. 160 per day. Track densities are counted in a single field of 1 cm² using replices of a microfiche reader.

The technique of calibrating albedo dosemeters in stray neutron fields has been improved and standardized using the single sphere albedo technique [4]. Improvements in this field are still subject of additional efforts mainly the development of an active albedo counter for quick field calibration. Knowing the calibration factor of the reference neutron source or at the location in a stray radiation field, albedo neutron dosemeters allows one to measure the neutron dose equivalent sufficiently accurately [3].

4. REFERENCES

[1] Piesch, E., Jasiak, J., Urban, M., Proc. Int. Conf. Track Etch Detectors, Acapulco, 1983
[2] Piesch, E., Burgkhardt, B., Proc. Int. Conf. Solid State Dosimetry, Ottawa, 1983
[3] Piesch, E., Burgkhardt, B., Proc. 10th DOE Workshop Neutron Dosimetry, Acapulco, 1983
[4] Burgkhardt, B., Piesch, E., 6th IRPA Congress, Berlin, paper Nr. 126,

ANWENDUNG DES PULSE-SHAPE-VERFAHRENS BEI MESSGERÄTEN MIT GROSSFLÄCHENPROPORTIONALZÄHLROHREN ZUR EINWANDFREIEN TRENNUNG UND GLEICHZEITIGEN ANZEIGE DER α- UND β-IMPULSRATE

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Zusammenfassung

Eine verbesserte Trennung bei gleichzeitiger Anzeige und höherem zeitlichen Auflösungsvermögen der α- und β-Impulsrate bei Großflächenproportionalzählrohren ermöglicht das Pulse-Shape-Verfahren gegenüber der bisher üblichen Amplitudendiskriminierung. Ein weiterer Fortschritt wird in einem von der äußeren Gasversorgung unabhängigen, stickstoffbetriebenen Proportionalzähler gesehen.

Zum Pulse-Shape-Verfahren

Um beim Umgang mit ionisierender Strahlung die Sicherheit der Beschäftigten und der Bevölkerung zu gewährleisten, gibt es neben den internationalen Richtlinien eine in jedem Land verbindliche gesetzliche Regelung zur Personen- und Umgebungsüberwachung. Hierfür geeignete Meßgeräte müssen ausreichend empfindlich sein, um die zum Teil sehr hohen Anforderungen des Gesetzgebers erfüllen zu können. Für Kontaminationsmessungen und Bruttoaktivitätsbestimmungen hat sich der Großflächenproportionalzähler durchgesetzt, der eine Trennung der α- von der β-Strahlung ermöglicht, allerdings erst nach Differenzbildung. In der Praxis wurde bisher die von den Teilchen im Zählvolumen abgegebene Gesamtenergie bestimmt. Die von einem α- oder β-Teilchen erzeugten Ladungen wurden gesammelt, d.h. das Zeitintegral über den Zählkammerstrom gebildet. Entsprechend der elektrischen Kapazität der Meßanordnung und der erzeugten Ladungen ergab sich ein Spannungsimpuls, dessen Höhe analysiert wurde. Der prinzipielle Fehler bei diesem Meßverfahren besteht darin, daß vorausgesetzt werden muß, daß jedes α-Teilchen mehr Energie im Zählraumvolumen abgibt als jedes β-Teilchen. Erst dann ist eine Unterscheidung von α- und β-Teilchen anhand der Impulshöhe möglich. Diese Voraussetzung ist jedoch in der Praxis so gut wie nicht erfüllbar. α-Teilchen können durch Selbstabsorption in der Probe, durch Absorption in der Luftschicht zwischen Probe und Zählrohrfenster, sowie im Zählrohrfenster selbst schon vorher den größten Teil ihrer Energie abgegeben haben, so daß ihre Energieabgabe im Detektor letztlich in der gleichen Größenordnung wie die der β-Teilchen liegen kann. In diesem Fall ist eine Trennung mittels einer Amplitudendiskriminierung nicht mehr möglich. Messungen haben ergeben, daß unter gewissen geometrischen Voraussetzungen fast alle aus einer reinen α-Probe (Po-210) stammenden Teilchen fälschlicherweise als β-Teilchen registriert werden.

Ein weiterer Nachteil der bisherigen Meßmethode ist darin zu sehen, daß die α- und β-Teilchen nicht gleichzeitig erfaßt werden können. Bedingt durch das Umschalten der Hochspannung sind zwei Meßschritte und anschließend eine Differenzbildung beider Meßergebnisse notwendig. Wünschenswert ist deshalb ein Meßprinzip, das einerseits wenig abhängig von der im Zähler abgegebenen Energie ist und andererseits simultan α- und β-Teilchen getrennt erfassen kann.

Erreicht man durch konstruktive Maßnahmen, daß die Meßeinrichtung kein integrales Verhalten mehr aufweist, so zeigt sich, daß der zeitliche Verlauf des Kammerstromes mit der sehr unterschiedlichen Ionisationsfähigkeit der α- bzw. β-Teilchen korreliert. α-Teilchen geben auf kürzester Wegstrecke einen großen Teil ihrer Energie ab, während sich die Energieabgabe von β-Teilchen über einen weiten Bereich erstreckt. Die konzentrierte Energieabgabe der α-Teilchen führt zu einem zeitlich rasch anwachsenden Kammerstrom mit entsprechend steilem Kurvenverlauf. Es hat sich gezeigt, daß die Anfangssteigung der α-Stromkurve um den Faktor 2 größer ist, als die der β-Stromkurve.

Die Schwierigkeit bei der Auswertung dieser Tatsache bestand darin, daß sich dieser Unterschied nur innerhalb eines Zeitraums von ungefähr 20 Nanosekunden nach Impulsbeginn bemerkbar macht. Eine entsprechende Schaltung wurde so ausgelegt, daß immer 10 Nanosekunden nach Impulsbeginn der momentane analoge Wert digitalisiert, abgespeichert und nach Ablauf der Meßzeit als α- bzw. β-Zählrate ausgegeben wird.

Die jetzt vorliegende Entwicklung zeigt bei gleichzeitiger Anzeige der α- und β-Impulsrate und höherem zeitlichem Auflösungsvermögen eine um über 1 Zehnerpotenz bessere Trennung unter sonst gleichen Bedingungen als bei Geräten mit Amplitudendiskriminierung. Der Vorteil für den Strahlenschutz - kürzere Meßzeiten bzw. höhere Nachweisempfindlichkeit, Erweiterung des Meßbereichs bei hohen Impulsraten und geringere Meßunsicherheiten, wenn α- und β-Strahlung immer gleichzeitig angezeigt wird - ist offensichtlich.

Stickstoffzähler mit Gasregeneration

Konventionelle Großflächenzähler können entweder im Gasdurchfluß oder als dichte Zähler, mit einer Gasfüllung versehen, betrieben werden. Der Nachteil des Durchflußzählers ist seine Abhängigkeit von einer permanenten und in vielen Fällen sehr aufwendigen Gasversorgung. Beim dichten Zähler mit Gasfüllung bestimmen die Leckrate und die chemische Zerstrahlung seine Lebensdauer. Die Gasmultiplikation eines Proportionalzählers hängt stark von der Reinheit des Zählgases ab. Um geringe Leckraten und damit hohe Lebensdauer zu gewährleisten, braucht ein Zähler mit stationärer Füllung so dicke Fenster, daß α-Strahlung nicht mehr gemessen werden kann. Es ist jedoch wünschenswert, von einer Gasversorgung unabhängig zu sein, ohne auf die Nachweismöglichkeit von α-Strahlen verzichten zu müssen.

Vom Standpunkt der Verfügbarkeit her gesehen, wäre Luft das ideale Zählgas. Die natürliche Umgebungsluft ist als Zählgas jedoch nicht brauchbar. Die ungünstigen elektrochemischen Eigenschaften des Sauerstoffes verbieten den Einsatz von Sauerstoffkonzentrationen in der Höhe des Luftsauerstoffgehaltes. Stickstoff-Sauerstoff-Gemische mit O_2-Konzentrationen bis zu zwei Prozent sind nach unseren Untersuchungen jedoch als Proportionalzählgas geeignet. Die Gasverstärkung bleibt bis zu dieser Konzentration konstant und ist gleich der des reinen Stickstoffs.

Die Verwendung von Stickstoff als Zählgas in einem langlebigen Detektor ist aus folgendem Grund sinnvoll: Undichtigkeiten im System haben das Eindringen von Luft zur Folge. Wenn es gelingt, die Feuchtigkeit und den Sauerstoff der Luft zu entfernen, bewirkt der mit ein-

diffundierende Stickstoff eine Erneuerung des Zählgases und gewährleistet dessen gleiche Zusammensetzung über lange Zeiträume.

Der Stickstoffzähler mit Gasregeneration besteht im Prinzip aus drei Komponenten: einem stickstoffgefüllten Detektor, einer Gasreinigungspatrone und einer Pumpeinrichtung.

Der Detektor, ein Großflächenzähler konventioneller Bauart, ist mit einer doppelseitig aluminisierten Folie geringer Flächendichte ausgerüstet, so daß Alphastrahlen nachgewiesen werden können. Um Verunreinigungen des Zählgases möglichst zu vermeiden, ist der Detektor über O-Ringe abgedichtet.

Aufgrund der geringen Gasverstärkung des Stickstoffs ist eine Elektronik mit möglichst günstigem Signal-Rausch-Verhältnis erforderlich. In der Anordnung werden deshalb rauscharme, ladungsempfindliche Vorverstärker und kapazitätsarme Detektoren verwendet.

Die Gasreinigungspatrone enthält ein Molekularsieb zur Beseitigung von Wasserdampf. Ein chemisch reduzierter Katalysator entfernt auf der Basis der Kupferoxidation den Sauerstoff aus dem Gas. Gasumwälzung (Pumpe, Ventilator) sorgt für den nötigen Gasdurchsatz durch Absorber und Detektor.

Die Lebensdauer des Detektors wird durch seine Leckrate und die Kapazität des Absorbers bestimmt. Ein Dauerbetrieb ist ohne hohen Wartungsaufwand möglich. Nicht möglich ist es bisher, die Pulse-Shape-Diskriminierung auch beim Stickstoffproportionalzähler anzuwenden.

MONTE CARLO CALCULATION OF ACCURATE NaI(Tl) SCINTILLATION DETECTOR RESPONSE FOR GAMMA RAYS AND DETERMINATION OF SPECTRUM-DOSE CONVERSION FUNCTIONS

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1. INTRODUCTION

Sodium-iodide [NaI(Tl)] scintillation detectors are widely used for the purpose of monitoring or evaluating dose rate from environmental gamma rays. For this purpose, the reliable gamma ray response of energy up to 10 MeV is needed. Although a number of calculations and experiments have been carried out already, they do not give adequately thorough information because of inaccuracy and a lack of tests above 3 MeV.

Here, the accurate response of eight types of NaI(Tl) scintillation detector for gamma rays up to 10 MeV was calculated using a Monte Carlo method. Taking into account the detector housing and scintillation efficiency, the calculated results have been found to agree well with the experimental data.

The operation function for spectrum-dose conversion [G(E) function] was determined from the calculated response. The G(E) function, which derives the gamma dose directly from the observed pulse-height spectrum by a simple procedure, has been found a useful dosimetric method for more than 10 years in Japan(1-3). However, the reliable energy range of the G(E) function application has been limited to under 3 MeV, because of the insufficient detector response data. The present research made it possible to evaluate easily and accurately the dose from gamma rays up to 10 MeV .

2. CALCULATION OF RESPONSE FUNCTIONS

Response functions of NaI(Tl) scintillation detectors for gamma rays were calculated using a developed Monte Carlo program, MARTHA. The authors previously showed that the calculated NaI(Tl) detector response was in good agreement with the experimental data, by taking account of: 1) the detector housing around a NaI(Tl) crystal; 2) the scintillation efficiency of the crystal(4).

Consideration of the detector housing causes a decrease of total absorption peak area, and an increase of Compton continuum level in calculated response. Also, depth of the valley between a total absorption peak and a Compton edge is reduced. Consideration of the scintillation efficiency brings positional changes among a total absorption peak, a Compton edge, and escape peaks. Some calculated response functions are compared with the experimental data in Fig.1.

Using the program, MARTHA, response functions of eight types of NaI(Tl) detector (1"ϕx1", 2"ϕx2", 3"ϕx3", 4"ϕx4", 5"ϕx4" cylindrical detectors and 2"ϕ, 3"ϕ, 5"ϕ spherical detectors) are calculated for gamma rays varying in energy from 40 keV to 10 MeV.

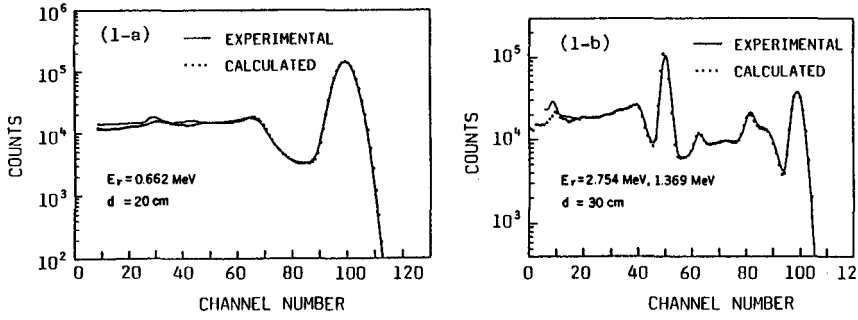

Fig.1. Comparison of calculated and experimental response functions for a 3"φx3" NaI(Tl) detector.

3. DETERMINATION OF G(E) FUNCTIONS

A spectrum-dose conversion function [G(E) function] is defined by the follwing equation,

$$\int_{E_{min}}^{E_{max}} f(E,E_0)G(E)dE = R(E_0), \tag{1}$$

where $f(E,E_0)$ is a pulse-height spectrum per one incident radiation, $R(E_0)$ is dose from a radiation with energy E_0, E_{max} is maximum energy in effective integral range, and E_{min} is minimum energy in effective integral range. Let $F(E) = \sum_j f(E,E_j)$ be the absorbed energy spectrum of a detector for certain gamma ray field, then, the total dose D from the radiation field can be easily given using the following equations,

$$\int_{E_{min}}^{E_{max}} F(E)G(E)dE = \sum_j \int_{E_{min}}^{E_{max}} f(E,E_j)G(E)dE = \sum_j n_j R(E_j) = D, \tag{2}$$

where n_j is the number of radiation with energy E_j.

In order to determine the G(E) functions, the information on $f(E,E_j)$ and $R(E_j)$ are necessary. The calculation of response function, $f(E,E_j)$, was described in sect.2. There are several choices for the value $R(E_j)$, for example, exposure, absorbed dose and dose equivalent can be selected. Here, the output of the G(E) functions is taken to be absorbed dose, and three conditions are simulated in order to estimate the absorbed dose in a human body: 1) the case where radiation equilibrium exists; 2) the case where gamma rays enter a semi-infinite slab of tissue-equivalent material perpendicularly to the surface; 3) the case where gamma rays enter isotropically a 30 cm diameter sphere of tissue-equivalent material(5).

For cases 2) and 3), simulations were carried out using a Monte Carlo calculation, on the assumption that tissue-equivalent material consists of 76.2% O, 10.1% H, 11.1% C and 2.6% N with a density of $1 g/cm^3$(5). For case 1), the absorbed dose can be calculated by a simple equation(6). Fig.2 shows the calculated dose distribution in tissue-equivalent material as a function of tissue depth in case 2). Fig.3 shows the comparison of maximum absorbed dose in tissue-equivalent material among cases 1), 2), and 3).

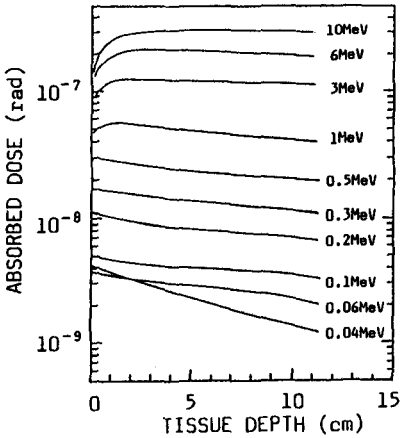

Fig.2. Calculated absorbed dose distribution in a 30 cm sphere of tissue-equivalent material. Intensity of gamma rays is 141.5 photons/cm^2.

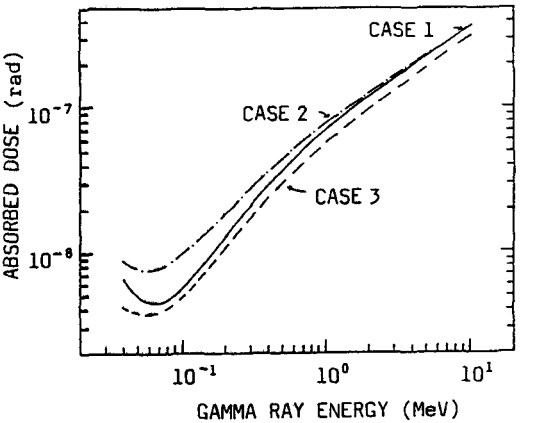

Fig.3. Maximum absorbed dose in tissue-equivalent material for case 1), 2) and 3). Intensity of gamma ray is 141.5 photons/cm^2.

For the derivation of the maximum absorbed dose, the G(E) functions of eight types of NaI(Tl) detector were determined, using polynominal fitting on the calculated results described above(Fig.4). In Fig.5, comparison of G(E) functions for a 3" x3" cylindrical detector was made among cases 1), 2) and 3). Cases 2) and 3) suppose the extreme exposure conditions of a human body to gamma radiations, and a user can select the appropriate one from the three types of G(E) function, according to the usage.

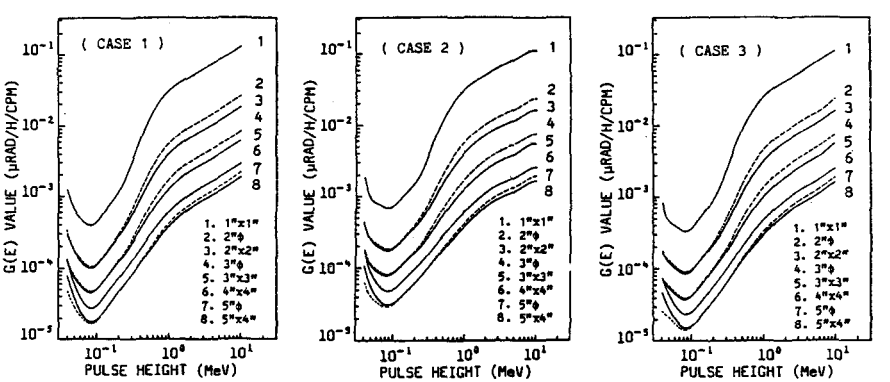

Fig.4. G(E) functions for derivation of maximum absorbed dose in a human body.

4. CONCLUSION

The accurate response functions of eight types of NaI(Tl) detectors, for gamma rays varying in energy from 40 keV to 10 MeV, were calculated making use of the Monte Carlo program, MARTHA. Maximum absorbed dose in a human body was calculated with a Monte Carlo method under three different irradiation conditions. The G(E) functions of eight types of NaI(Tl) detector were determined to derive maximum absorbed dose in a human body under the three different conditions. The dose evaluation using G(E) functions is expected to be applied to more complicated problems.

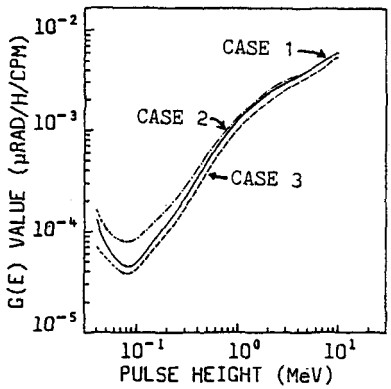

Fig.5. Comparison among three types of G(E) function for a 3"ϕx3" NaI(Tl) detector.

REFERENCES

(1) S. Moriuchi and I. Miyanaga, Health Phys. 12(1966) 541.

(2) S. Moriuchi, "A New Method of Dose Evaluation by Spectrum-Dose Conversion Operator and Determination of the Operator", JAERI 1209(1971).

(3) S. Moriuchi, "A Dosimetric Instrument Based on the Spectrum Weighting Function Method for Environmental Radiation Measurements", JAERI-M 7066(1977).

(4) K. Saito and S. Moriuchi, Nucl. Instr. Meth., 185(1981) 299.

(5) International Commission on Radiation Units and Measurements, ICRU Report 19(1971).

(6) W. K. Sinclair, Radiobiological Dosimetry. In F. H. Attix and E. Tochilin (ed.), Radiation Dosimetry, Vol. III, p.617, New York and London, Academic Press.

ERZEUGUNG HOCHENERGETISCHER PHOTONENBÜNDEL DURCH EINFANG THERMISCHER NEUTRONEN

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Einleitung

Für Strahlenschutzdosimeter, die in Photonenfeldern mit hochenergetischen Anteilen oberhalb von 3 MeV eingesetzt werden, z.B. in Bereichen von Kernkraftwerken, wo die Strahlung des Nuklids N-16 eine wesentliche Rolle spielt, ist eine für diesen Energiebereich geeignete Kalibrierung erforderlich, wenn beträchtliche Meßfehler vermieden werden sollen /1/. Die dazu benötigte Referenzstrahlung läßt sich nach verschiedenen Methoden erzeugen /2,3,4/.

Am Forschungs- und Meßreaktor Braunschweig /5/ wurde nach dem Verfahren des Neutroneneinfangs an Nickel- und Titantargets /1,3/, mit dem sich relativ hohe Dosisleistungen bei guter Bestrahlungsgeometrie erzielen lassen, ein Strahlrohr für Referenzbündel hochenergetischer Photonenstrahlung bis etwa 9 MeV aufgebaut. Ziel der Arbeiten ist es, das Ansprechvermögen von Photonendosimetern, deren Energie-Nenngebrauchsbereich über 3 MeV hinausgeht, zu messen.

Experimenteller Aufbau

Fig. 1 zeigt den Aufbau der Apparatur und deren Anordnung am Reaktor. Das benutzte Strahlrohr S11 mündet in etwa 1,50 m Entfernung vom Südkern des Reaktors in den Graphitblock der thermischen Säule. Dadurch ergibt sich ein sehr niedriger Untergrund an Reaktor-γ-Strahlung und schnellen Neutronen. Das Target besteht aus 6 Nickelscheiben mit einer Gesamtmasse von 6,8 kg bzw. 10 Titanscheiben mit insgesamt 3,7 kg Masse. Beim schieflastigen Normalbetrieb des Reaktors mit 925 kW Leistung im Nordkern und 75 kW im Südkern beträgt die Flußdichte der thermischen Neutronen am Targetort $4{,}2 \cdot 10^9\,cm^{-2}s^{-1}$ und das Cadmiumverhältnis 850:1. Bei Umkehrung der Leistungsverteilung erhöhen sich die thermische Flußdichte und damit die später genannten Werte der γ-Dosisleistungen um etwa den Faktor 12. Durch Bleikollimatoren wird der Durchmesser der Bündel zu 100 mm und der doppelte Öffnungswinkel zu 4^o festgelegt. In das Strahlrohr gestreute thermische Neutronen werden durch einen Stopfen aus Bor-Polyäthylen vollständig unterdrückt. Der Strahlrohreinsatz wird durch eine Dosisaufbauschicht aus 5 cm Plexiglas und einer unmittelbar dahinter angebrachten Ionisationskammer mit 370 cm^3 Nutzvolumen als Monitor abgeschlossen. Zusätzlich zu der in Fig. 1 dargestellten Anordnung können bis zu 30 cm dicke Filterschichten aus Aluminium in das Strahlrohr eingebracht werden, wodurch sich die Strahlenqualität auf Kosten der Gesamtdosisleistung zu höheren Energien hin verändern läßt.

Von einem Abstand von 4 m vom Target an stehen die Photonenbündel für Meßzwecke zur Verfügung.

Meßergebnisse

Photonenbündel, die ohne Zusatzfilterung gemäß Fig. 1 erzeugt werden, lassen sich nach den Meßergebnissen folgendermaßen charakterisieren:

	Nickel-target	Titan-target	Bemerkungen
Homogenität im Strahlquerschnitt	> 95%	> 95%	Messung mit Zählrohr von 1 cm Durchmesser
Ionendosisleistung			Messung mit Stabdosimeter FH39U bei 4,20 m Targetabstand und 75 kW Teilleistung im Südkern
- frei in Luft	12nA/kg (17OmR/h)	22nA/kg (31OmR/h)	
- in Wasser	33nA/kg (46OmR/h)	46nA/kg (64OmR/h)	Maximalwert im Phantom in etwa 3 cm Tiefe
Effektive Energie	7,0 MeV	4,5 MeV	Aus Schwächungsmessungen mit Aluminium ermittelt
Hauptlinien im γ-Spektrum in MeV	9,OO;8,53; 7,82;7,54; 6,83; O,88;O,47	6,75;6,55; 6,41; 1,59;1,38; O,32	Messung mit Ge(Li)-Detektor

Durch Einbringen von Zusatzfiltern aus Aluminium in das Strahlrohr werden die folgenden Änderungen erreicht:

	Nickeltarget				Titantarget			
Filterdicke in cm	0	10	20	30	0	10	20	30
Relativwert der Ionendosisleistung in %	100	48	24	13	100	44	21	10
Effektive Energie in MeV	7,0	7,7	7,9	8,1	4,5	5,5	6,0	6,4

Die"Härtung" der Photonenbündel mit zunehmender Filterdicke, die durch das Ansteigen der effektiven Energien deutlich zum Ausdruck kommt, kann durch γ-Spektrometrie auch direkt belegt werden. Die Frage, ob dabei kontinuierlich verteilte Beiträge aus Compton-Streuprozessen eine wesentliche Rolle spielen, ist zur Zeit noch offen.

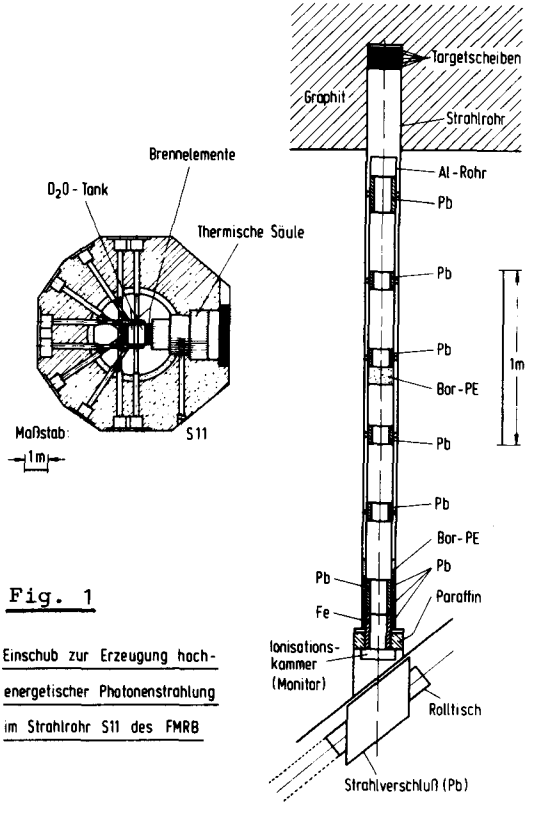

Fig. 1

Einschub zur Erzeugung hochenergetischer Photonenstrahlung im Strahlrohr S11 des FMRB

/1/ F. Bermann, G. Troesch, VIII. Congrès International de la Société Francaise de Radioprotection, Saclay 23-26 Mars 1976, p. 538-565

/2/ J. Beck et al., Atomkernenergie/Kerntechnik 34,1979, p. 57-60

/3/ L. Jarczyk et al., Nucl. Instr. Meth. 13,1961, p. 287-296

/4/ D.W.O. Rogers, Health Physics 45,1983, p. 127-137

/5/ W. Heintz, PTB-Mitteilungen 78,1968, p. 5-11

THE RESPONSE OF SOME PHOTON DOSEMETERS TO SLOW NEUTRONS *)

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INTRODUCTION

The measurement of dose equivalent in mixed neutron-gamma fields is an important task of radiation protection dosimetry, since in practical cases neutrons are always accompanied by photon radiation. Slow neutrons interact with matter mostly through neutron-capture reactions with relatively high cross sections, mostly with the emission of photons or, in some cases, of charged particles. It is to be expected, therefore, that slow neutrons have a significant effect on the reading of a photon dosemeter by interaction with the dosemeter material itself and material in its surroundings, e.g. a phantom or the body on which the dosemeter is worn. To separate these effects from those induced by external photon radiation the thermal-neutron reference beam of the Research Reactor of the PTB in Braunschweig is a most feasible tool because of its low photon component. This paper describes the irradiation facility, measurements of its properties and some investigations of photon dosemeters used in radiation protection.

THE THERMAL-NEUTRON REFERENCE BEAM AT THE PTB

At the Research and Measuring Reactor Braunschweig a thermal-neutron reference beam is installed for irradiation experiments /1/. Neutrons from a beryllium scatterer located at the area of maximum thermal neutron flux density are guided by a collimator to the irradiation position outside the reactor. The accompanying gamma radiation generated by neutron-capture processes in the Be scatterer, its surroundings, and the collimator materials is greatly reduced by inserting into the beam a 20 cm thick filter consisting of four bismuth single crystals. This creates a slow-neutron spectrum with a mean energy of 0.028 eV which was measured with a chopper /2/.

The thermal neutron flux density in the centre of the beam is about 10^6 $cm^{-2}s^{-1}$ at 1 MW reactor power; it is determined by activation of 20 µm thick gold foils. The cadmium ratio for gold foils is about 200 (ratio of activations when irradiated without or with Cd cover) and the fast neutron flux density is below one percent of that of the thermal neutrons.

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In order to estimate the residual gamma-ray component in the beam, measurements were performed with a Geiger-Müller (GM) counter (Valvo, Type ZP 1100) which had been calibrated with photons from a ^{60}Co source in terms of exposure. The GM counter was screened against the thermal neutrons by covering it with a shield of $^{6}Li_2CO_3$.

The contribution of intermediate and fast neutrons to the measured count rate was estimated from the corresponding flux density values using the neutron response of the counter given in /3/; it was found to be negligible. Taking into account the contributions of room background and the activation of the very thin aluminium encapsulation of the Li_2CO_3 shield an upper limit of 1.3 mR/h for the exposure generated by the photons in the thermal-neutron beam could be derived. Assuming a conversion factor of 0.01 Sv/R /4/ this results in a photon dose equivalent of 0.013 mSv/h. The neutron dose equivalent was derived from the flux density in the centre of the beam and using the appropriate spectrum-averaged fluence-to-dose equivalent conversion factor ($1.05\ 10^{-11}$ Sv cm^2 according to ICRP Publ. 21 /5/); the result was 37 mSv/h. Thus the ratio of photon to slow-neutron dose equivalent in the beam was determined to be smaller than 1:2800.

RESPONSE OF PHOTON DOSEMETERS TO THERMAL NEUTRONS

In the following discussion the response is defined as the ratio of the measured value of the dose equivalent to the neutron dose equivalent. The measured value of the dose equivalent is obtained from the indicated value of the dosemeter using the calibration factor for ^{60}Co radiation. The neutron dose equivalent is calculated from the neutron fluence measured free in air in the absence of the dosemeter and the spectrum-averaged fluence-to-dose-equivalent conversion factor as mentioned above.

Geiger-Müller Counter

As the GM counter is a photon dosemeter frequently used in mixed neutron-gamma fields, its response to thermal neutrons was of particular interest. Irradiation of the bare GM counter with slow neutrons creates activation of the counter wall materials, resulting in a count rate contribution due to the decay of the generated radionuclides (about 1.5% of the count rate during irradiation, measured immediately after irradiation). Taking this and the residual beam photons into account, we found a photon dose equivalent of 2.06 mSv/h. The neutron dose equivalent was 37 mSv/h (cf. last section). Thus the response of the GM counter to thermal neutrons, as defined above, is 0.056. This compares sufficiently well with a value of 0.07 derived from data given in /6/; it is two orders of magnitude higher than that for neutrons between 100 keV and 2 MeV /3/, which indicates that in measurements of this type thermal-neutron background ought to be carefully considered.

Film Badges

Various film badges are used to determine the photon dose equivalent for radiation protection purposes. In the Federal Republic of Germany a "photon badge" is officially used /7/; in cases where neutron radiation is expected a "neutron badge" is used /7/, which is similar to the "CERN badge" developed and in use at the CERN, Geneva. All the badges contain a lead filter (0.8 mm thick) for determining the photon dose equivalent for energies above about 100 keV. In the latter two badges a separate determination of the thermal-neutron dose equivalent is possible by means of a cadmium and a tin filter /8,9/. If this is done the measured value of the film behind the lead filter has to be corrected for its response to thermal neutrons in order not to overestimate the photon dose equivalent. A comparative measurement of film dosemeters recently organized by the Commission of the European Communities showed, however, that the influence of thermal neutrons on the reading of a film dosemeter was not generally considered to a sufficient degree /10/.

The response to thermal neutrons of the film under the Pb-filter depends on the irradiation conditions (with or without phantom), on the type of film holder, on the type of film used, and on the emulsion development conditions. It was measured after free-air irradiation as 0.60 for the "photon badge", 0.85 for the "neutron badge", 0.67 for the "CERN badge", all with AGFA softpac film (evaluated with an automatic film evaluation system at the PTB /11/), and as 0.43 for the CERN badge with KODAK Type 2 film together with the routine evaluation procedure used at CERN /12/. When irradiating the film badges on a phantom (cylindrical water-filled phantom, 30 cm in diameter and 60 cm in height), which is the more realistic calibration condition, we get 0.96, 1.3, 1.3 and 0.75, respectively. These values contain additional contributions from neutrons backscattered from the phantom. There is a difference between the three values for the AGFA softpac film and the value for the KODAK type 2 film which may be attributed to different interactions of the neutrons with the emulsion itself (e.g. different content of silver and other components in the emulsion).

Penholder Ionization Chamber and TL-Dosemeter

For regular checks of possible radiation exposure penholder-type ionization chambers are often worn by the personnel. These dosemeters are calibrated in terms of exposure using a ^{137}Cs source. With the aforementioned convention about converting exposure into photon dose equivalent we obtained responses to thermal neutrons of 0.4 when irradiated free in air and 0.8 when irradiated on the phantom.

Another system used in personal monitoring employs thermoluminescence dosemeters as the radiation-sensitive elements. One of these systems in routine use at EIR, Würenlingen, Switzerland /13/ was irradiated in the thermal neutron beam. The reading from the routine evaluation performed at the EIR resulted in response values of 0.47 for free-in-air and 0.83 for on-phantom irradiations.

CONCLUSIONS

The individual dosemeters investigated here (for film badges only the film behind the lead filter) show a response to thermal neutrons between 0.75 and 1.3 if irradiated on a phantom. Within this variation they give a measure for the dose equivalent when irradiated in a mixed ^{60}Co-photon and thermal-neutron field. In cases where a separate neutron dosemeter sensitive to thermal neutrons is worn together with a photon dosemeter, the photon dosemeter reading has to be corrected in order to avoid double measurements of the thermal neutrons. The advantage of the Geiger-Müller counter when used to measure the photon dose component free in air in mixed fast-neutron gamma fields is its very low fast-neutron response /3/. This advantage is strongly reduced, when thermal-neutrons are present as the response to them is 0.056, which is two orders of magnitude higher than that for fast neutrons. As compared to the other dosemeters investigated the Geiger-Müller counter is, however, still the best choice for this purpose.

REFERENCES

/1/ H. Kluge, K. Knauf, PTB-Mitteilungen 87 (1977), 38.

/2/ W.G Alberts, E. Dietz, H. Kluge, K. Knauf, Programme Rad. Prot. Progress Report 1982, EUR 8486 DE/EN/FR (1983), 176.

/3/ S. Guldbakke, R. Jahr, H. Lesiecki, H. Schölermann, Health Phys. 39 (1980), 963.

/4/ H. Reich, G. Bengtsson, Health Phys. 40 (1981), 898.

/5/ Int. Com. Rad. Prot., Data for Protection against Ionizing Radiation from External Sources, ICRP Publ. 21 (1973).

/6/ V.E. Lewis, D.J. Young, Phys. Med. Biol. 22 (1977), 476.

/7/ F. Wachsmann, Die Messung der Personendosis, GSF-Bericht Nr. 76 (1975).

/8/ F.H. Attix, W.C. Roesch, Radiation Dosimetry, Vol. II, Academic Press, New York, San Francisco, London (1966).

/9/ J.A. Dennis, J.W. Smith, S.J. Boot, Proc. IAEA Symp. Neutron Monitoring for Radiological Protection (1967), 557.

/10/ W.G. Alberts, H. Kluge, Proc. Fourth Information Seminar on the European Radiation Protection Dosemeter Intercomparison Programme, Bilthoven, 25 to 27 October 1982, to be published by the Commission of the European Communities.

/11/ P. Ambrosi, J. Böhm, G. Nolte, W. Kolb, Proc. Third Intern. Symposium on Radiological Protection, Inverness (1982), Vol. 2, 655.

/12/ M. Höfert, private communication (1983).

/13/ F. Cartier, C. Wernli, private communication (1983).

PHYSICAL ASPECTS OF USING ROSSI-PROPORTIONAL COUNTERS FOR RADIATION PROTECTION MEASUREMENTS AND PRACTICAL DEVELOPMENT

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Introduction

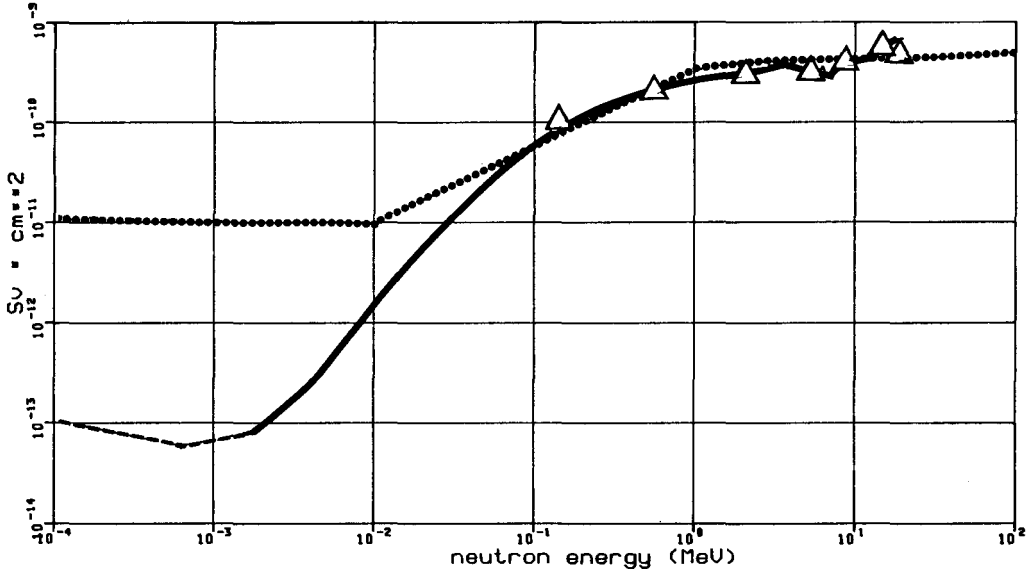
Radiation protection measurements require the determination of dose equivalent, H. This quantity was introduced by ICRU /1/ and ICRP /2/ to account for the fact that the correlation of a given biological effect with absorbed dose will change in general when the type of radiation is varied. Dose equivalent H at a point of interest in tissue is defined as the product of the absorbed dose, D, and the quality factor, Q. The latter factor has values of between 1 and 20 and has been specified as a function of linear energy transfer, LET, because it is assumed that differences in biological effectiveness are related to differences in the microscopic distribution of absorbed dose delivered by the charged particles. The quality factor is unity for X, beta and gamma rays. In practice neutrons constitute the most important radiation for which Q is greater than one. Usually an average quality factor has to be applied because the dose is delivered over a range of values of LET. This holds in particular for complex radiation fields.

Low pressure tissue-equivalent proportional counters ("Rossi-counters") are cavity-chambers and present therefore, similar to ionization chambers, a very accurate method for measuring absorbed dose in most practical situations. The gas amplification of the proportional counter enables to measure in addition the energy deposition for single primary particles yielding energy imparted spectra ("microdosimetric spectra"). These spectra and the LET distribution of the charged particles delivering the dose are closely correlated for a wide range of radiations so that this type of instrument enables the simultaneous determination of absorbed dose and in approximation the average quality factor and thus dose equivalent also in unknown radiation fields. Furthermore, the spectral information can be used to evaluate the contribution of photons and neutrons to dose equivalent.

Physical Aspects

For the purpose of radiation protection dose limitations are stated as limits of dose equivalent in organs or tissue of the body. However, the value of the dose equivalent in an organ can rarely be determined directly and therefore a hierarchy of primary, secondary and derived limited quantities have been introduced. They include measurable or so called operational quantities which are supposed to provide a sufficiently conservative estimate of organ and tissue dose equivalents. The operational quantities thus permit to relate results

of radiation protection monitoring to dose equivalent limits. An example for a quantity of this type is the maximum dose equivalent in a cylindrical phantom for a uniparallel beam (MADE) /2/. The Figure shows the fluence-to-MADE conversion function for neutrons for several decades of neutron energy as given by ICRP publication 21 /2/. For actual measurements of MADE (or other measurable quantities) detectors must have a response function proportional to the conversion function.



Fluence-to-MADE conversion function /2/ (dotted line), calculated response function for a Rossi counter (solid line) and experimental results as function of neutron energy.

The method of determining dose equivalent from Rossi-counter measurements has been described /3,4,5,6/ and is straightforward. Using the computer code for calculating microdosimetric distributions by Caswell and Coyne /7/ the response function of a Rossi counter was determined for monoenergetic neutrons between 1 keV and 20 MeV (below 1 keV the results are very preliminary) (Fig.). The simulated irradiation condition are idealized in the sense that modifications of the radiation field by interactions with the detector material are ignored. In particular, spectrum degradation and photon production in the detector wall are not considered. The Figure includes several results of measurements with Rossi-counters for monoenergetic neutrons between 144 keV and 19 MeV. The experiments were performed at the Physikalisch Technische Bundesanstalt (PTB), Braunschweig and the Gesellschaft für Strahlen- und Umweltforschung (GSF), Munich. The spherical counters used have an inner diameter of 12.7 mm and the thickness of the wall made of A-150 tissue-equivalent (TE) plastic is

2.5 mm. The propane based TE-gas mixture pressure was set at 8.9 kPa (67 torr).

Conversion function and calculated response function agree well above 50-100 keV and also the experimental points do not differ significantly. This agreement shows the principal suitability of this type of detector to determine dose equivalent in this energy range. The agreement between the two functions and the experimental points permit the conclusion that neutron scattering in the phantom or detector wall is not important for the value of MADE above 100 keV. The discrepancy between conversion function and response function at lower energies, however, has to be attributed mainly to the photon contribution to MADE. The photons are produced in capture processes of neutrons degraded in the phantom to thermal energies. In fact, for neutrons below 10 keV about 50% of the conversion function is due to gamma rays from (n,γ) processes /8/.

A real counter, i.e. a counter with finite wall-thickness which implicitly includes the effects of neutron interactions in the wall is expected to have a response function which agrees better below 50 keV than the calculated function in the Figure. This is confirmed by preliminary results of measurements with 2 keV neutrons. The use of additional wall material (300 cm^3 polyethylene) yielded an additional contribution to dose equivalent of about $6\ 10^{-12}$ Sv cm^2 in terms of gamma rays. This value corresponds to about 60% of the conversion function at this energy. These results support the assumption that it is possible to match the response function of Rossi counters to fluence dose equivalent conversion functions for the entire energy range between 1 keV and 20 MeV by choosing the wall thickness appropriately. It is anticipated that a far better proportionality can be obtained than with other types of monitors including REM-counters /9/. Therefore the use of this detector is promising not only in practical radiation protection dosimetry but also as a reference method.

Practical Development

The development of a practical dose equivalent meter based on a Rossi-counter in our laboratory foresees a counter with a diameter of about 5 cm. Equivalent dose rates of interest in radiation protection range from about 0.1 $\mu Sv\ h^{-1}$ to values in excess of $3\ 10^3\ \mu S\ h^{-1}$. For a mixed radiation field with about 50% contribution of photons to absorbed dose pulse frequencies were found to range from about 1 s^{-1} at 0.75 $\mu Sv\ h^{-1}$ to $4\ 10^3\ s^{-1}$ at $3\ 10^3 \mu Sv\ h^{-1}$. This range of dose equivalent rates is of interest for radiation protection. The real count rates depend on the type and spectrum of the radiation field. In a pure photon field they may be two orders of magnitude higher than in a pure neutron field.

For the highest count rates fast pulse processing is required. If a portable battery operated instrument is to be constructed this requirement is opposed to the need to use low power consumption electronic components which in general are relatively slow. Our concept is based on the principle of using a minimum number of channels in the pulse height analysis. This allows an optimum compromise between the two requirements. The digital analogue conversion is achieved by use of comparators /10/. The conversion of the

pulse height information into dose equivalent (rate) and absorbed dose (rate) will be performed by a simplified method permitting single signal processing. A maximum count rate of 10^4 s^{-1} is anticipated.

An instrument based on this concept and covering the above range of count rates and displaying dose equivalent rate and absorbed dose rate can be built with a power consumption of 3 to 4 Watt. This means that the use of rechargeable batteries is reasonable and that portable instruments can be realized. If requirements for the performance of the instrument are further reduced personal dosimeters are feasible /11/. It has already been shown that non portable and portable instruments can be built /12,13,14/.

References

/1/ ICRU Report No. 33, Radiation Quantities and Units (1980)
/2/ ICRP Publication 21 (1973)
/3/ Hartmann, G., Menzel, H.G., Schuhmacher, H., Proc. 4th Symp. Neutron Dosimetry, 225 (1981)
/4/ Hartmann, G., Menzel, H.G., Schuhmacher, H., Krauss, O., Proc. 8th Symp. Microdosimetry, 1117 (1982)
/5/ Booz, J., Poli, A., Proc. 4th Symp. Neutron Dosimetry, 237 (1981)
/6/ Ricourt, A. et al., Proc. 7th Symp. Microdosimetry, 625 (1981)
/7/ Caswell, R.S., Coyne, J.J., Proc. 6th Symp. Microdosimetry, 97 (1976)
/8/ NCRP Report No. 38 (1971)
/9/ Cosack, M., Lesiecki, H., Proc. 4th Symp. Neutron Dosimetry, 407 (1981)
/10/ Charlton, D.E. et al., Proc. 8th Symp. Microdosimetry, 1099 (1982)
/11/ Quam, W., IEEE Transactions on Nuclear Science 29, 637 (1982
/12/ Braby, L.A. et al., Proc. 4th Symp. Microdosimetry, 1075 (1982)
/13/ Nguyen, V.D. et al., Proc. 8th Symp. Microdosimetry, 1087 (1982)
/14/ Lindborg, L., private communication (1983)

Acknowledgement: This work has been financially supported by the Commission of the European Commission.

MIXED RADIATION DOSE EQUIVALENT INDEX METER WITH IMPROVED RESPONSE FOR INTERMEDIATE AND THERMAL NEUTRONS

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Two integral methods for direct determination of dose equivalent index have been so far applied i.e.

i/differential "recombination method.

ii/Twin detector system-TE ionization chamber and organic scintillation probe.

These two methods suffer for serious drawback for unappropriate response to neutrons below 100KeV due to small mass of the detectors.

In order to recognize the possible ways of correcting the response of the measuring systems, the theoretical method of analyzing of the response has been set up. The response of TE ionization chamber for neutrons, using Monte Carlo method, has been calculated.

It has been shown that by alteration of the composition of the walls of TE ionization chamber as well as the composition of gas fillied a chamber the response of the detection system may be approximated to the needed one.

Paper describes the method of calculation and the results for twin TE-organic scintillator system of dose equivalent index determination.

A New Technique to Improve the Accuracy of Albedo Neutron Dosimeter Evaluations*

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Introduction

The calibration factor for albedo neutron dosimeters varies greatly depending upon the energy of the neutrons in the exposure. It has been found that, under some scattering conditions, the calibration factor can be determined by using a ratio of the thermal and fast neutron dose rates. In a previous study, we used the PNR-4 9-in. sphere remmeter to study the relationship between the calibration factor for albedo neutron dosimeters and the thermal neutron component of the dose.[1] We found that the relationship between the percent thermal and the calibration factor varied greatly in operational areas. Therefore, the calibration factor could not be determined from the incident thermal neutrons.

We review here results obtained over an eight-year period at each Lawrence Livermore National Laboratory facility where neutron exposure may occur. When each facility is considered separately, we find there is a stronger relationship between the ratio of the readings of the 9-in. to 3-in. spheres (9/3 ratio) and the "percent thermal" (percent of total neutron dose contributed by thermal neutrons) than we had expected to exist. With this relationship confirmed, we then review our dosimeter readings from personnel and albedo badges and find that the readings are consistent with the use of a calibration factor for the albedo dosimeter which varies with changes in the ratio of the personnel and albedo dosimeter TLD readings. We find significant improvement in our personnel exposure estimates by applying these variable calibration factors in place of the single value used previously. It is still necessary to know in which facility the person was working when exposed; but, for most of the LLNL exposures, only two sets of calibration factors are required.

Results From Field Surveys With Instruments

We plotted results obtained from neutron surveys over an eight-year period to show the 9/3 ratio as a function of "percent thermal." We had plotted the results from each building or facility separately, but found that the results from several of these facilities can be combined for purposes of dosimeter evaluations (Fig. 1). The calibration factor for the albedo neutron dosimeter (right ordinate of Fig. 1) increases as the percent thermal increases.

Not all the results from the buildings or facilities (at LLNL we use building numbers for identification) fell along the solid line seen in Fig. 1. In Fig. 2, we show the results obtained at the Bldg. 231 plutonium storage vault. Inside the vault there is a lower thermal neutron component for a given 9/3 ratio and all the data points fall below the solid line. The data obtained from outside the vault, however, again fall close to the solid line. The low thermal component observed inside this vault results from the large room. When a neutron source is placed in a room, the thermal neutron fluence is fairly constant throughout the room, but the magnitude of this fluence is a function of room size, with smaller rooms having higher fluences. Therefore, in a small storage vault such as the Bldg. 332 vault, the percent thermal at a given 9/3 ratio is higher than the percent thermal in a large storage vault such as Bldg. 231 (Fig. 2).

*This work was performed under the auspices of the U. S. Department of Energy by Lawrence Livermore National Laboratory under contract No. W-7405-ENG-48.

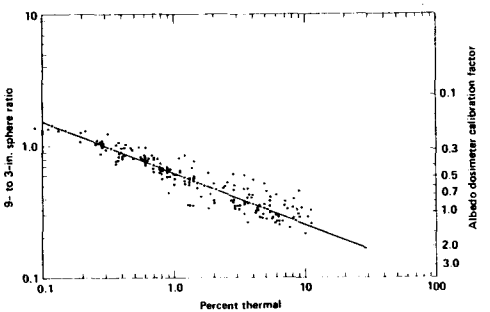

Fig. 1. The ratio of the 9/3-in. spheres as a function of the percent thermal (percent of the total neutron dose rate delivered by thermal neutron) for several LLNL facilities.

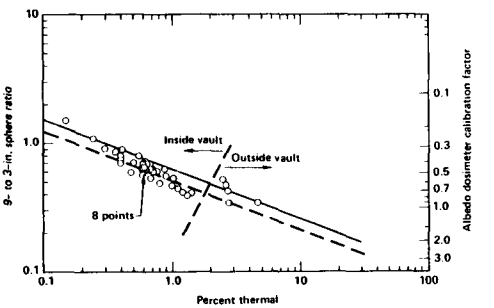

Fig 2. The ratio of the 9/3-in. spheres as a function of the percent thermal at the Bldg. 231 plutonium storage vault.

Figure 3 shows the results for Bldg. 233, which is a neutron-source storage area. The points fall above the solid line taken from Fig. 1, indicating a higher average thermal neutron component than was found at the other facilities. Because the room is large, we would expect the results to fall below the curve, but the neutron sources are stored in concrete, polyethylene, or paraffin shielding which thermalize the neutrons. Each of these shields then becomes a source of thermal neutrons, which causes the observed increase in the thermal neutron component throughout the room.

The thermal neutron component, from a source located outside a building, is lower than if the source were in a room. Figure 4 shows the results obtained at Site 300 in the area around the LINAC (target is outside), where the data points fall below the line.

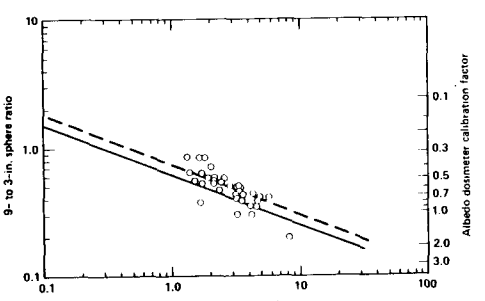

Fig. 3 The ratio of the 9/3-in. spheres as a function of percent thermal at the Bldg. 233 neutron source storage vault.

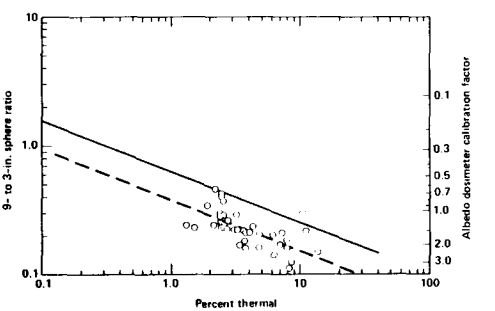

Fig. 4 The ratio of the 9/3-in. spheres as a function of the percent thermal in the vicinity of a LINAC.

We had again confirmed there is no constant relationship between the 9/3 in. sphere ratio (and corresponding albedo calibration factor) and the percent thermal for all exposure conditions. However, we found that, for neutron exposures occurring at all of our facilities (except the reactor), there is a definite relationship and the spread of the data points around the line drawn on each of the figures is fairly small. Therefore, a ratio of the readings of the TLDs in the albedo dosimeter to the readings of the thermal neutron sensitive TLD in the personnel badge can be used to approximately determine the albedo neutron dosimeter calibration factor if the facility in which the exposure occurred is known.

Dosimeter Response

We can calculate the expected TLD readings of the personnel and albedo dosimeters from Figs. 1 through 4. The reading of the TLD 600s in the albedo dosimeter is primarily from the "intermediate energy" albedo neutrons. The reading also contains what normally is a small contribution from the incident intermediate energy neutrons and incident and reflected thermal neutrons. The size of the albedo TLD reading is proportional to the "Albedo Dosimeter Calibration Factor" given on the right-hand scale of the figures. For example, if the albedo calibration factor is one, the reading from the TLD 600 would be 1R for a neutron dose equivalent of 1 Rem.

The neutron reading of the TLD 100 in the personnel badge is more complex. The largest part of the reading, when used in field application, will normally be from the incident and reflected thermal neutrons. It also responds to the incident and albedo intermediate energy neutrons which are detected by the albedo dosimeter. In addition, since it is not surrounded by cadmium like the TLD 600 in the albedo dosimeter, it also responds to the "thermal" albedo neutrons. Its response to these neutrons is $\sim$1.6 times the intermediate energy response of the albedo dosimeter, giving a total sensitivity 2.26 times that of the albedo dosimeter. A further complication is that the response of the TLD 100s is about half the response of TLD 600s to neutrons.

Using the above information, we have calculated the expected relative reading of the TLD 600s in the albedo dosimeter and the TLD 100s in the personnel badge. The lines drawn in Figs. 5 and 6 are based on these calculations and show the albedo calibration factors. Plotted on Fig. 5, are data from badges worn by persons working in one of our facilities where we know the albedo calibration factor that should be applied to the TLD readings. When we compare the known factor determined by observing the workers' actions with those derived from the location of the points on Fig. 5, we find good agreement, confirming the accuracy of this technique.[2]

Results from other buildings are shown in Fig. 6. In most cases, when the calibration factor that existed at the worker's location was known, that point on the figure is in agreement with the calibration factors derived from the lines.

Routine personnel exposures can be evaluated using a table showing the ratio of the albedo dosimeter and bare TLD 100 readings and the corresponding albedo calibration factor for each facility.[2] Since TLD measurements are subject to anomolies and uncertainties in their readings, we require that the observed ratios from the TLDs be within the extremes known to exist in the work area. A procedure for handling anomolies, questionable TLD readings, and readings where only one dosimeter indicates an exposure needs to be developed.

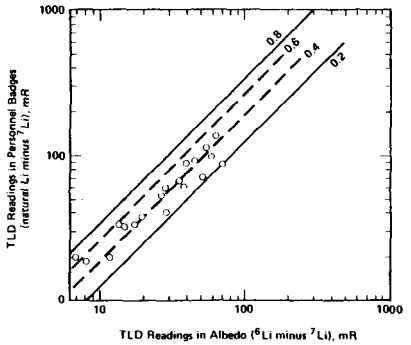

Fig. 5. Results from personnel dosimeters and albedo badges worn by personnel at one of our facilities. The lines correspond to several albedo dosimeter calibration factors.

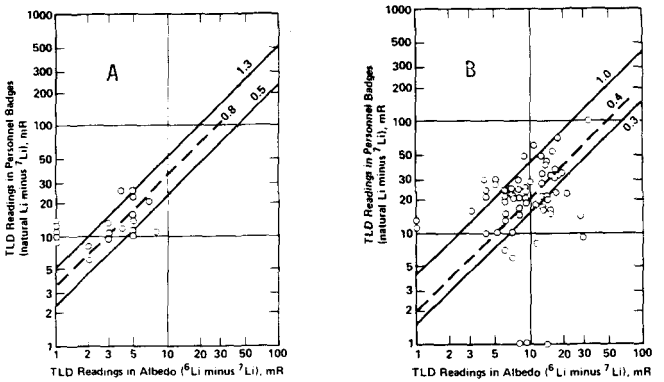

Fig. 6. Results from personnel dosimeters and albedo badges worn by (A) custodians in a radiochemistry building, and (B) personnel at our Bldg. 332 plutonium vault. The lines correspond to several albedo dosimeter calibration factors.

References

1. Hankins, D.E., "The Effect of Energy Dependence on the Evaluations of Albedo Neutron Dosimeters," Proceedings of the Ninth Midyear Topical Symposium on Operational Health Physics, Denver, CO, p. 861 (1976).

2. Hankins, D.E., Evaluation of Albedo Neutron Dosimeters Using the Ratio of TLD Readings in Personnel and Albedo Badges, Lawrence Livermore National Laboratory Report, UCID-in publication (1983).

THE SINGLE SPHERE ALBEDO TECHNIQUE: A REFERENCE INSTRUMENT FOR DOSEMETER CALIBRATION AND ANALYSIS OF STRAY NEUTRON FIELDS

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1. INTRODUCTION

In neutron monitoring there is the need for a standardized technique which may be applied in stray neutron fields in order to establish reference data mainly for the field calibration of neutron dosemeters and to replace the more sophisticated multisphere technique. The single sphere albedo technique originally applied for the calibration of albedo neutron dosemeters [1-3] makes use of thermal neutron detectors in the center of a 30 cm polyethylene sphere and behind boron-plastic shields on the surface of the sphere. Based on well established response functions of four neutron detectors and an on-line computer assisted evaluation technique, the system allows for the estimation of actual field data, i.e. neutron fluence, absorbed dose, dose equivalent as well as the corresponding factors d_{eff}, h_{eff} and Q_{eff} [4,5].

2. TECHNIQUE

The single sphere albedo technique reduces the energy dependence of the 30 cm sphere by using for instance the read-out of TLD600/

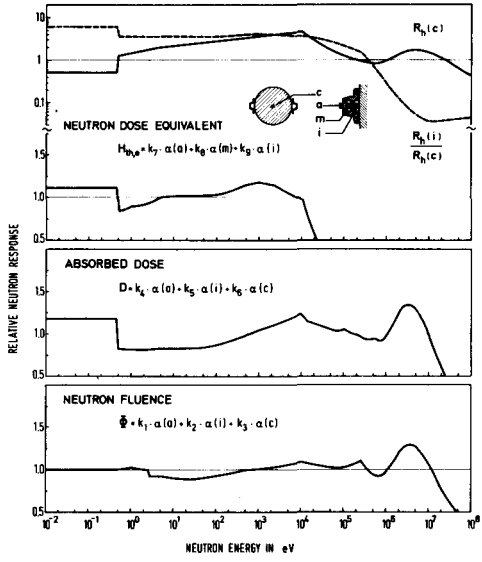


Fig. 1: Dose equivalent response function R_h of the 30 cm sphere detector c and linear combinations using the energy response of the detectors c, m, a, i and the constants k_1 to k_9 for the estimation of neutron fluence and absorbed dose

TLD700 detector pairs in the center of the sphere c and in three positions m, a, i of the boron-plastic encapsulation of the Karlsruhe albedo dosemeter. The detector system enables one to calculate at first the dose contributions of thermal and epithermal neutrons using the readings of detector m and a in front and behind the boron absorber and thus the reading contributions of fast neutrons. With respect to fast neutrons the reading ratio of albedo detector i and sphere detector c is used to establish an energy parameter E_o. The apparent dose contributions of the sphere detector c separated for thermal, epithermal and fast neutrons are then corrected for energy dependence. This technique has been standardized for an application in multi-directional fields by using the reading sum of two dosemeters positioned diametrically at the sphere.

Using experimental and recently calculated fluence response functions $R_\Phi(k)$ of the detectors a, m, i and c, it can be shown in Fig. 1 that the adequate superposition of the detector reading $\alpha(k)$ allows to estimate neutron fluence and absorbed dose practically independent of neutron energy within +30 %/-20 % and the dose equivalent of thermal and epithermal neutrons within ±20 %.

3. STANDARDIZED EVALUATION TECHNIQUE

The on-line computer program described in Table 1 estimates at first the neutron dose reading taking into account the individual response of the detector, the zero dose reading and the separation of the gamma contribution, and corrects then for angular response using the reading ratio of diametrically opposite detectors at the sphere. By splitting up the four detector readings $\alpha(k)$ with k = a, m, i, c in three energy groups of thermal, epithermal and fast neutrons, the set of simultaneous equations are solved. After estimation of the neutron fluence contributions Φ_{th} and Φ_e for thermal and epithermal neutrons, the readings $\alpha_f(k)$ for fast neutrons are derived and the energy parameter E_o reduces the energy dependence of $\alpha_f(c)$ for the estimation of H.

$$\Phi = k_1 \cdot \alpha(a) + k_2 \cdot \alpha(i) + k_3 \cdot \alpha(c) \quad (1)$$

$$D = k_4 \cdot \alpha(a) + k_5 \cdot \alpha(i) + k_6 \cdot \alpha(c) \quad (2)$$

$$H_{th,e} = k_7 \cdot \alpha(a) + k_8 \cdot \alpha(m) + k_9 \cdot \alpha(i) \quad (3)$$

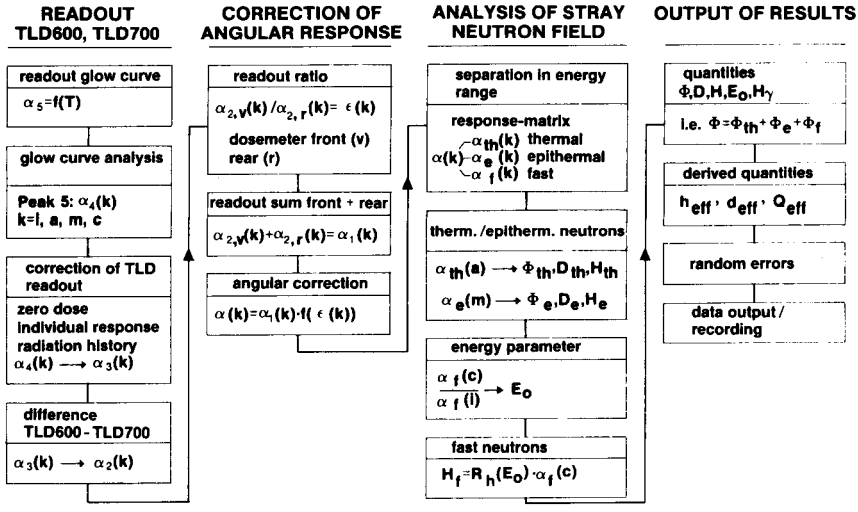
$$H = h_{th} \cdot \Phi_{th} + h_e \cdot \Phi_e + R_{h,E_o} \cdot \alpha_f(c) \quad (4)$$

$$= k_7 \cdot \alpha(a) + k_8 \cdot \alpha(m) + k_9 \cdot \alpha(i) + R_{\alpha,E_o} \cdot \alpha_f(c) \quad (5)$$

with the detector readings $\alpha(k)$ and k = a, m, i, c, the constants k_1 to k_9 found by least square fits and the fluence-to-dose conversion factors d and h given by ICRP 21.

From H, D and Φ the following actual field data can be computed

- the neutron fluence-to-dose equivalent conversion factor $h = H/\Phi$
- the neutron fluence-to-absorbed dose conversion factor $d = D/\Phi$
- the effective quality factor $Q_{eff} = H/D$.



Tab. 1: On-line computer assisted evaluation of TLD detectors using the single sphere albedo technique

4. APPLICATION

With respect to a qualitative interpretation of the stray neutron field, the energy parameter E_O established for neutron energies > 10 keV has been found to be comparable with the mean neutron energy. Values between 50 keV $\leqq E_O \leqq$ 120 keV have been found behind shieldings at reactors and medical linacs (Fig. 2). Experimental field data derived with the single sphere albedo technique have been found to be highly consistant for the various intercomparison experiments at the HPRR in Oak Ridge [6] and agree with calculated data taking into account the neutron spectrum and the detector response functions. The maximum scatter of 14 measurements for each neutron spectrum was within ± 15 %. Because of energy dependence detector i indicates any change of the neutron spectrum in the energy range of intermediate neutrons. The comparison of calculated and measured data shows an excellent agreement for the bare reactor spectrum but small deviations for the shielded reactor spectra.

5. CONCLUSION

Compared to other detector systems based on the reading of four detectors, the main advantages of the single sphere albedo technique is the use of only one sphere, one detector type, one irradiation, one calibration, and the simultaneous measurement of all detectors. After establishing more reliable response functions and a computer assisted evaluation, this technique seems to be most consistant to estimate actual values of Φ, D, H and the corresponding conversion

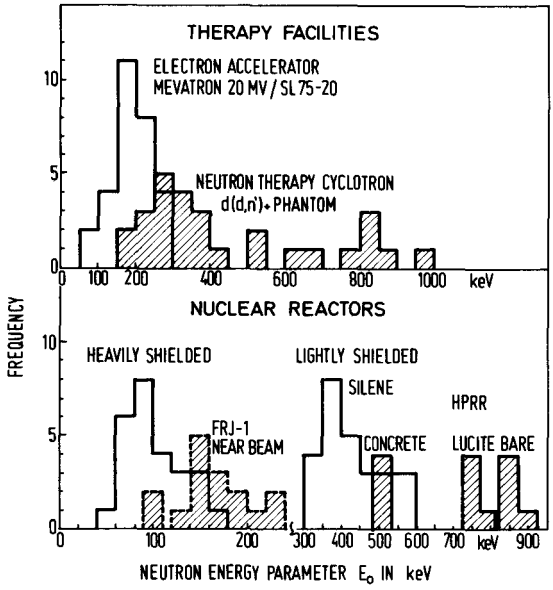


Fig. 2: Frequence distribution of the energy parameter E_o found at reactors and accelerators

factors d_{eff}, h_{eff}, Q_{eff} in the stray neutron field at the location of interest. The technique is more or less independent of any change of the fluence-to-dose conversion factors by applying a new linear fit of the response functions as described in Fig. 1. Because the neutron energy response functions of the TLD600/TLD700 detectors are well established only gamma calibrations are routinely applied. Instead of TLD's, active detectors which simulate the albedo dosemeter type of interest [7] in future will improve this technique for quick field calibrations of albedo dosemeters.

6. REFERENCES

[1] Piesch, E. and Burgkhardt, B., Proc. 5th IRPA Congress Vol. 3, p 121 (1980)
[2] Piesch, E. and Burgkhardt, B., 5th Euratom Symp. Neutron Dos., Vol. I, p 549 (1981)
[3] Piesch, E. and Burgkhardt, B., 8th DOE Workshop Pers. Neutron Dos., PNL-SA-9950, p 111 (1981)
[4] Piesch, E. and Burgkhardt, B., 3rd Symp. Rad. Prot., Inverness (1982)
[5] Piesch, E. and Burgkhardt, B., to be published in Radiation Protection Dosimetry (1984)
[6] Piesch, E., Burgkhardt, B. and Venkataraman, G., Rad. Prot. Dos., Vol. 3, p 25-38 (1982)
[7] Piesch, E., Burgkhardt, B. and Hofmann, I., Report KfK 2847 (1979)

RADIATION CHARACTERISTICS OF DEPTH DOSE EQUIVALENT METERS*

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The dose equivalent at a depth of 10 mm in the ICRU sphere [1] may become the new operational quantity in the monitoring of external exposure. The introduction of this quantity will require a different energy-dependence of the response of dose and doserate meters for X and γ-radiation than is now required by the exposure or air kerma quantities. However, this will not impose major difficulties on the manufacturer of dose equivalent meters because most kinds of the detectors used show an energy dependence of response similar in shape to that of the conversion factor [2, 3, 4] between the dose equivalent at a depth of 10 mm in the ICRU sphere and the air kerma. A number of commercial dose and doserate meters were therefore investigated with respect to their suitability to directly indicate the depth dose equivalent.

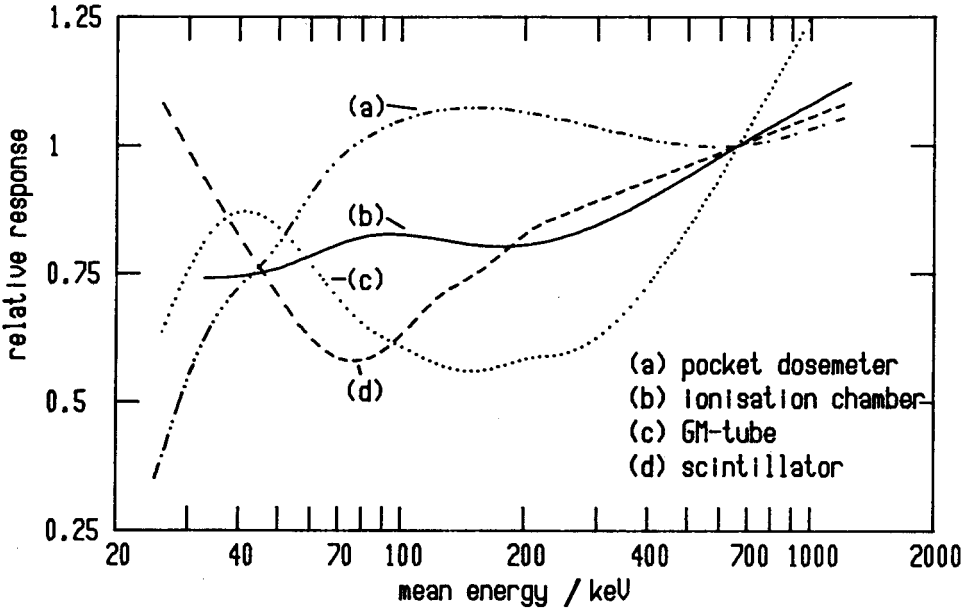

Fig. 1 Energy dependence of response of four commercial dosemeters relative to 662 keV with respect to the 10 mm depth dose equivalent.

Fig. 1 shows the response as a function of photon energy for four commercially available dose and doserate meters with different kinds of detectors. The indicated quantity is the 10 mm depth dose equivalent. It is shown that in all cases there is no insurmountable difficulty in fitting the response of the dosemeter to the new measurement quantity. Even without any change a limit of $\pm$30 % relative to the indication at 662 keV, which is frequently used in present standards, is exceeded only very slightly. Similar results were also

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obtained with nearly all the other dosemeters which were investigated. According to the results shown in fig. 1 pocket dosemeters and ionisation chambers seem to be suitable to indicate the 10 mm depth dose equivalent with only minor changes.

Most of the commercial portable dosemeters, however, use a Geiger-Müller tube as detector because of the advantages they offer, e.g., the instruments are cheap, resistant, and digital indication is easy to perform. Our own investigations were therefore carried out with a Valvo counter-tube ZP 1310. This tube's energy dependence of response with respect to the 10 mm dose equivalent is shown in fig. 2 (curve a). For reducing the maximum of the response the GM tube is usually covered with a filter of Sn or Pb with an additional PMMA ring or an air space to compensate for the strong absorption of these materials at low photon energies. Furthermore the K-absorption edge of the lead gives rise to a distinct minimum of the response in the energy range at about 88 keV.

To obtain a filter material which has a mass attenuation coefficient varying only very slightly in the energy range of interest, 30 keV to about 80 keV, we have investigated various combinations of different materials with K-absorption edges lying in this photon energy range.

Combinations of tin and elements of the lanthanides were under investigation. The computed mass attenuation coefficient for a mixture of tin, cerium and gadolinium with a mass ratio of Sn:Ce:Gd = 1:3:3 is given in fig. 2 as curve (b). It can be seen that the resulting attenuation coefficient differs only by $\pm$40 % from a mean value of 13 cm^2/g, while the coefficient of the elements alone varies in the same energy interval by about a factor of four in the case of gadolinium and a factor of ten in the case of tin.

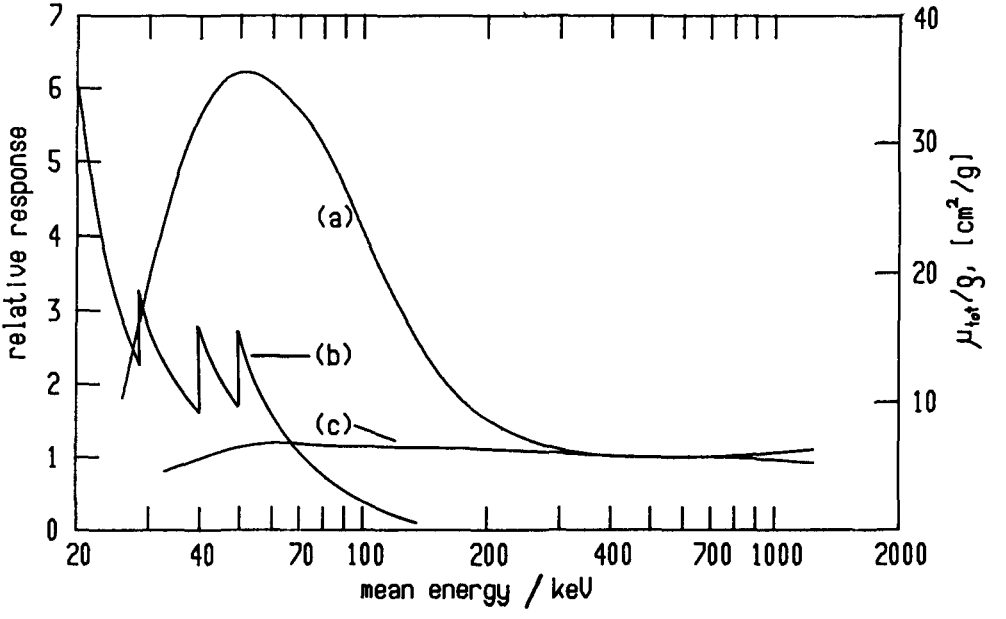

Fig. 2 Energy dependence of response of the GM tube ZP 1310 without any filters (a) and with a filter described below (c). (b) represents the total mass attenuation coefficient of a mixture of Sn:Ce:Gd = 1:3:3, right ordinate.

Most of the pure metals of the lanthanides are difficult to handle and very expensive, so oxides of these elements were used for the filter construction. Preliminary experiments were made with a mixture of Sn-, CeO_2- and Gd_2O_3 powders with mass ratios for the metals of 1:3:3. From this mixture a small cylindrical hull divided into two equal parts was prepared by adding plastic glue to the powders. Between the two parts of the hull various PMMA rings of different thicknesses could be inserted.

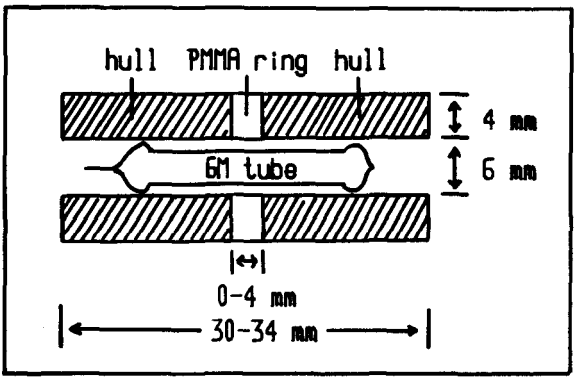

Fig. 3 Filter arrangement for a GM tube with variable PMMA rings.

The complete arrangement is shown in fig. 3. The curve in fig. 2(c) represents the results we have obtained with this filter arrangement and a PMMA ring 3,5 mm thick. The response of the GM tube is drawn as a function of mean photon energy relative to the ^{137}Cs point. It can be seen that a GM tube with the filter combination under discussion has an almost energy independent response and fulfills present requirements over the complete energy range from 30 keV up to 2 MeV by indicating the dose equivalent at a depth of 10 mm in the ICRU sphere.

Conclusions.

A number of dosemeters currently available have been investigated as to their suitability to indicate the 10 mm depth dose equivalent in the ICRU sphere. As a result of these experiments it can be stated that only small modifications of these instruments will be necessary for a direct indication of this quantity. Some of the instruments already fulfill present requirements with respect to the energy dependence of the response.

As an example of a Geiger-Müller counter an energy filter consisting of a mixture of tin and elements of the lanthanides is presented which allows the measurement of the 10 mm depth dose equivalent over a wide range of photon energy.

1 ICRU Report 25. Conceptual for the Determination of Dose Equivalent. ICRU Washington, D.C. (1976).

2 K. Hohlfeld, G. Großwendt: Radiation Protection Dosimetry 1 (1981), 277.

3 IEC/45B (Central Office)43, Februar 1982: Beta, X and Gamma Radiation Dose Equivalent and Dose Equivalent Rate Meters for Use in Radiation Protection.

4 P. J. Dimbylow, T. M. Francis: National Radiological Protection Board, Harwell, Didcot (1979), Report NRPB - R92.

DETERMINATION OF DOSE EQUIVALENT INDEX AND DOSE DISTRIBUTION IN THE ROTATIONAL TISSUE EQUIVALENT SPHERE

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Abstract

The dose equivalent index and dose distribution of the rotational tissue equivalent spherical phantom were determined by exposing the incident parallel beam of photon radiation in the energy ranges 0.22-1.25 Mev.

I. Introduction

The establishment of the recommendation criteria of the ICRP for the maximum absorbed dose index (D_I) and the dose equivalent index (H_I) relating to the principal concept for the radiation protection system shall be decided from the conversion factor in accordance with the actual exposure model. However, only with the current dose equivalent conversion factors couldn't sufficiently covered all the facts of actively involved with radiation works.

Therefore, in the present study, the depth dose distribution for the photon radiations were measured from the spherical phantom of the human tissue equivalent material to meet the active condition by means of changing the speed of the phantom.

II. Experimental

The TLDs used in the present study were LiF-TLD-700, and the energy of the incident photon beams of below 147 kev were obtained from x-rays sources, and the energies of gamma-rays were obtained from Cs-137 (0.66 Mev) and Co-60 (2.66×10^{10}Bq) sources, respectively. In the case of gamma-ray sources, the distances from the source center to the center of the spherical phantom embedded with LiF-TLD-700 were 100 cm and 300 cm, and the exposure dose rates at that position were 310.15 mr/h and 110 mr/h, respectively.

The phantom used were the tissue equivalent material of 30 cm diameter, density of 1g/cm^3 and the composition ratio of four principal elements in soft tissue, proportioned as described by ICRU: O:0.76; C:0.11; H:0.10 and N:0.26. LiF-TLD-700 were embedded in the section of the semi-sphere phantom whose geometrical parameters are the same as shown in Table 1. The spherical phantom were dividened with two semi-spherical phantom sections that were located symmetrically both sides of the central section of the phantom.

The changes in velocity of the phantom during the radiation exposure were as the following two types: the one was a stationary state (0 rpm) that the phantom were fixed on the speed control turn table and the others were rotational states of which the phantom rotates with 7 rpm as a steady velocity and 12 rpm as a fast velocity. The later two velocities were assumed to the equivalent velocities of the actual radiation workers.

Table 1. The Geometrical Parameters of the Tissue Equivalent Semi-Sphere Phantom in Spherical Polar Coordinates.

Section A: X'_{max} = 147 mm

Depth	Z-axis (mm)	X' (mm)	θ	r	φ*
3	15.5	144.5	83.88	145.33	
6	15.5	141.5	83.75	142.35	
9	15.5	138.5	83.61	139.36	
12	15.5	135.5	83.47	136.38	
15	15.5	132.5	83.33	133.40	
20	15.5	127.5	83.07	128.44	
40	15.5	67.5	77.07	69.26	
80	15.5	47.5	71.93	49.96	
120	15.5	27.5	60.59	31.57	

Section B: X'_{max} = 141.5 mm

Depth	Z-axis (mm)	X' (mm)	θ	r	φ*
3	40.5	138.5	73.70	144.30	
6	40.5	135.5	73.76	141.42	
9	40.5	132.5	73.12	139.51	
12	40.5	129.5	72.63	135.69	
15	40.5	132.5	73.12	139.51	
20	40.5	129.5	72.63	135.69	
40	40.5	101.5	68.25	109.28	
80	40.5	61.5	56.63	73.64	
100	40.5	41.5	45.70	57.99	

Section C: X'_{max} = 131.5 mm

Depth	Z-axis (mm)	X' (mm)	θ	r	φ*
3	65.5	128.5	62.99	144.23	
6	65.5	125.5	62.44	141.56	
9	65.5	123.5	61.87	138.91	
12	65.5	119.5	61.27	136.27	
15	65.5	116.5	60.65	133.65	
20	65.5	111.5	59.57	129.32	
40	65.5	91.5	54.40	112.53	
80	65.5	51.5	38.18	83.32	
100	65.5	31.5	25.68	72.68	

Section D; X'_{max} = 117.5 mm

Depth	Z-axis (mm)	X' (mm)	θ	r	φ*
3	89	114	51.02	144.63	
6	89	111	51.28	142.27	
9	90.5	108	50.04	140.91	
12	90.5	105	49.24	138.62	
15	90.5	102	48.42	136.66	
20	90.5	97	46.99	132.66	
30	90.5	77	40.39	118.82	
80	90.5	37	22.24	97.77	

Section E: X'_{max} = 95.0 mm

Depth	Z-axis (mm)	X' (mm)	θ	r	φ*
3	114	92	38.90	146.49	
6	114	89	37.98	144.63	
9	115	86	36.79	143.6	
12	115	83	35.82	141.82	
15	115	80	34.82	140.09	
20	115	75	33.11	137.3	
40	115	55	25.56	127.47	

Section F: X'_{max} = 55 mm

Depth	Z-axis (mm)	X' (mm)	θ	r	φ*
3	139	52	20.51	148.41	
6	139	49	19.42	147.38	
9	139	46	18.31	146.41	
12	139	43	17.79	145.50	
15	139	40	16.05	144.64	
25	140	30	12.09	143.18	

X' : Radius of various depth for each section.
r : Length of radius vector from the origin of the phantom section to the point of the embedded TLDs.
θ : Zenith angle.
φ* : Azimuth angle ($0^{\circ}, \pm 15^{\circ}, \pm 30^{\circ}, \pm 45^{\circ}, \pm 60^{\circ}, \pm 75^{\circ}, \pm 90^{\circ}$).

Each of the section (A-F) as mentioned in Table 1 were dividened appropriate volume elements in order to embedded with LiF-TLD-700, as an example, the section A were classified 132 portions according to the surface and inner part of the phantom. LiF-TLD-700 were initially given the standard annealing i.e., 16 hours at 80°C followed by 1 hour at 400°C prior to the first irradiation for different parallel beams of radiation. Thermoluminescense measurement were made using the Harshow TLD reader system Model-3000 which had been built utilizing heated nitrogen gas as the heating transfer medium.

III. Results

The results of the depth dose distribution in 30 cm diameter tissue equivalent spherical phantom with three different velocities of 0, 7 and 12 rpm for parallel beam of Co-60 and Cs-137 gamma-rays, respectively, were compared in Figures 1 and 2. As shown in the Figures 1 and 2, the distributions of the maximum absorbed dose were distributed within the depth of 10 cm from the surfaces of the phantom.

The more the phantom section reached upper portions, the more the maximum absorbed dose distribution absorbed in the deeper position of the various sections of A to E. These phenomena were occurred all the same throughout the sections, excepts for the outer section F of the phantom. Furthermore, as shown in Figure 2, the maximum absorbed dose distribution of 7 rpm were distributed approximately 10 times greater than that of 12 rpm.

According to the comparison results of the dose equivalent indices obtained by rotation of the phantom as mentioned previously with that of the MIRD-5 phantom, there were not prominent discrepancy occurred in close vicinity of 80 kev, whereas the difference of 20% were occurred in the lower energy range below 80 kev. In accordance with the aspect of the condition of the radiation dose at the phantom of this study compared with parallel and divergence beam. The difference were occurred 2.5 times about 80 kev. However the maximum value of the dose equivalent index and dose equivalent should be happened about 80 kev as shown in Figure 3.

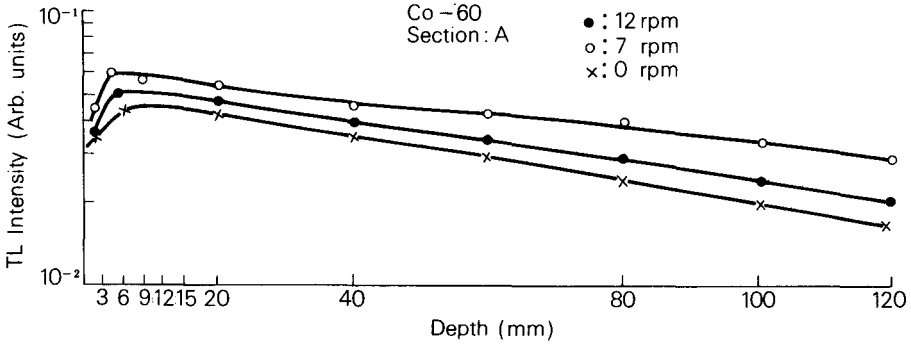

Fig. 1 Depth Dose Distribution Curves for Co-60 Gamma-Rays,

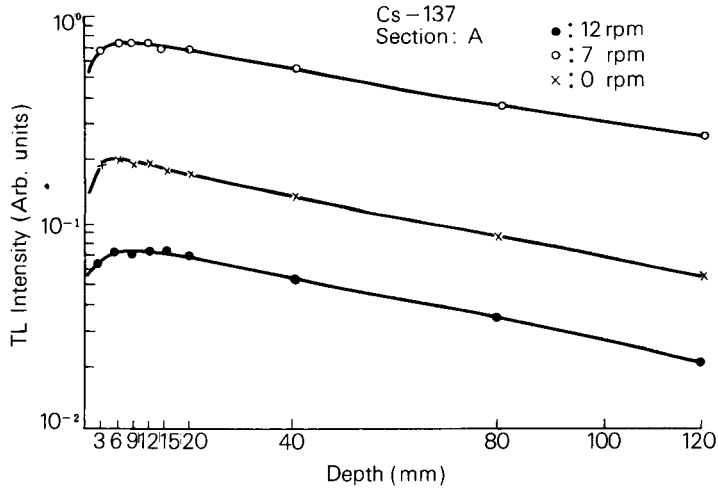

Fig. 2 Depth Dose Distribution Curves for Cs-137 Gamma-Rays.

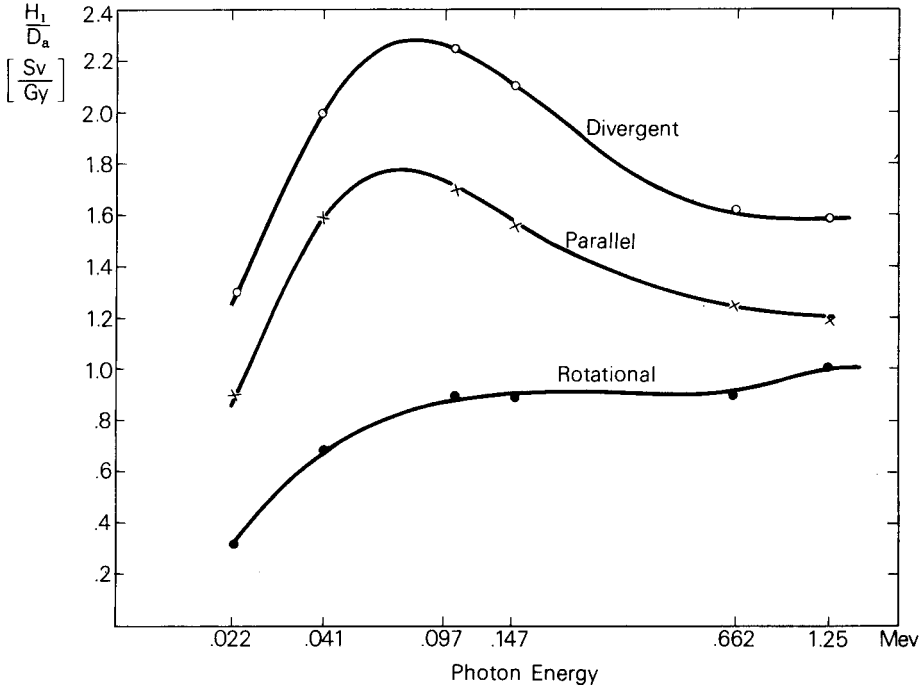

Fig. 3 Dose Equivalent Indices to the Tissue Equivalent Spherical Phantom Normalized to Absorbed Dose in Air as a Function of Photon Energy for Three Different Source Geometries.

A MICROPROCESSOR-BASED RADIATION MONITORING SYSTEM

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ABSTRACT

At TRIUMF intense beams of 500 MeV protons are used to produce secondary beams of mesons for nuclear physics research. The size and nature of the facility are such that about seventy active radiation monitors of various types are deployed at the site and its boundary to measure accelerator beam spills, air activity, residual activity and neutron radiation levels. In order to process and display the monitor readings a monitoring system has been developed which will accept inputs from a variety of detectors. Two microprocessors are used in a CAMAC based data acquisition and display system. One of the microprocessors handles the display of radiation levels together with warning, trip levels and alarms on a colour CRT monitor while the second micro-processor performs the warning and trip logic and passes appropriate flags to the accelerator interlock system. The programs reside in erasable programmable read-only memory (EPROM) to provide protection against arbitrary changes. The system may be interrogated via a keyboard terminal for diagnostic and calibration purposes and there is a provision to defeat the trip function of individual detectors but such defeats are then displayed on the CRT monitor. All information on radiation levels may be accessed by an independent data logging system allowing extensive records to be kept on magnetic tape for periodic analysis. This system is compact, has high reliability and can readily be tailored to site specific needs.

INTRODUCTION

At today's generation of high intensity accelerators used as 'meson factories' the power carried by the particle beams is such that even brief excursions from the operating envelope may have serious consequences in terms of damaged equipment and the attendant exposure to personnel in carrying out necessary repair. It is therefore mandatory that both operating and induced fields be closely and continuously monitored. The higher beam intensities also require more extensive monitoring of the neutron leakage through the primary shielding as well as monitoring of the induced radio-activities in the accelerator building air. These monitoring requirements can lead to a multiplication of detector systems and readouts which cannot always be easily accommodated in the control room. At TRIUMF we have designed a CAMAC-based monitoring system which can accept a variety of signals from radiation detectors. The system processes these signals and generates warning and trip flags for the interlock system and displays radiation levels on a page-selectable colour CRT monitor.

RADIATION DETECTORS

Initially we have included four types of detectors in the system although there are no fundamental difficulties in including other types.

a) Beamspill Monitors/Active Area Monitors

These consist of 50 mm diameter NE102 plastic scintillators mounted on photomultiplier tubes. The signal from these detectors is the anode current suitably amplified. These monitors are used both as beamspill monitors and as area monitors when access is requested to an exclusion area. The required dynamic range

covering five orders of magnitude is achieved by raising or lowering the bias voltage.

b) Beamspill Monitors

These are air ionization chambers and are used in normally inaccessible beamline tunnels near meson production targets where operating fields would produce excessive radiation damage in plastic scintillators. The output is an analogue signal proportional to the beamspill.

c) Neutron Monitors

Neutron monitors of the Anderson-Braun type are deployed in accessible areas outside the primary shielding of the accelerator. The output is in the form of NIM standard pulses at the rate of ~6000 counts/μGy.

d) Air Monitors

The most common species of radionuclides induced in the accelerator air are ^{13}N, ^{15}O, ^{11}C, ^{18}F and ^{41}Ar. To measure exhaust emissions and ambient levels in the experimental halls we use continuous flow air monitors which consist of 50 mm × 50 mm NaI(TL) detectors centered in a 4 litre volume of air shielded by 50 mm of lead on all sides. The output of these detectors is fed to preamplifier-amplifier-discriminator units and the TTL pulses from the discriminator are routed to the monitoring system. The analogue amplifier output is available to periodically measure the gamma ray spectrum of the sampled air.

SYSTEM HARDWARE

The system hardware is contained in two CAMAC crates (Fig. 1). Each crate

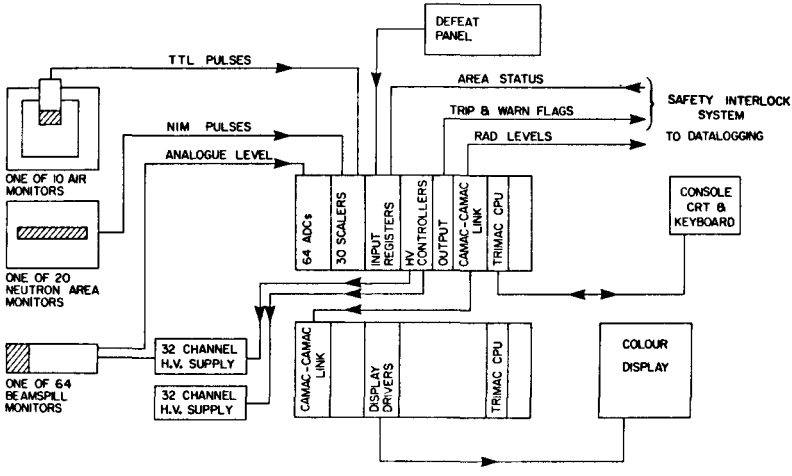

Fig. 1. Simplified schematic of TRIUMF radiation monitoring system.

has its own intelligence in the form of a TRIUMF built micro-processor controller (TRIMAC). The TRIMAC is based on the 8085 micro-processor and is provided with 4K random access memory (RAM) and 48K erasable programmable read only memory (EPROM). One crate is dedicated to accepting and interpreting data and generating warning and trip flags while the other crate performs all the display functions.

a) Inputs

There are three types of input which are accepted by the system. Analogue levels from beamspill monitors are read by two 32 channel 12 bit ADCs, pulses from neutron and air monitors are counted by five six channel scalers, and area status from the interlock system as well as defeat status in the form of contact closures are accepted by three dual 24 bit input registers. A normally locked defeat panel allows overriding of any trip set point associated with a particular monitor.

b) Outputs

Whenever a particular monitor reading exceeds a warning or trip set point a flag is sent to the central safety system[1] which then takes appropriate action (e.g. sounds alarms, shuts down accelerator, etc.) These flags are transmitted via a dual 24 bit output register. Other flags are set for certain fault conditions (e.g. high voltage power supply error). Radiation levels converted to appropriate units are sent via a CAMAC to CAMAC link both to the second CAMAC crate for display purposes and to the central control system for data logging.

SOFTWARE

The software for that part of the system doing the data reading and processing is based on the 'AMX' multitasking executive[2] which allows the scheduling of a number of tasks on a priority basis. Figure 2 shows the overall program organization and a flow chart outlining the 'read and process beamspill monitor' task (RP BSM in Fig. 2) is illustrated in Fig. 3. The cycling time for this task to read and process all 64 beamspill monitors is 250 ms. The program resides in EPROM but tables of scale factors, trip set points, and bias voltage levels are

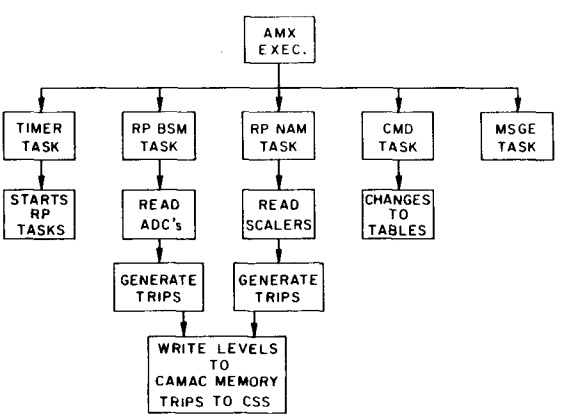

Fig. 2. Program organization for data reading and processing.

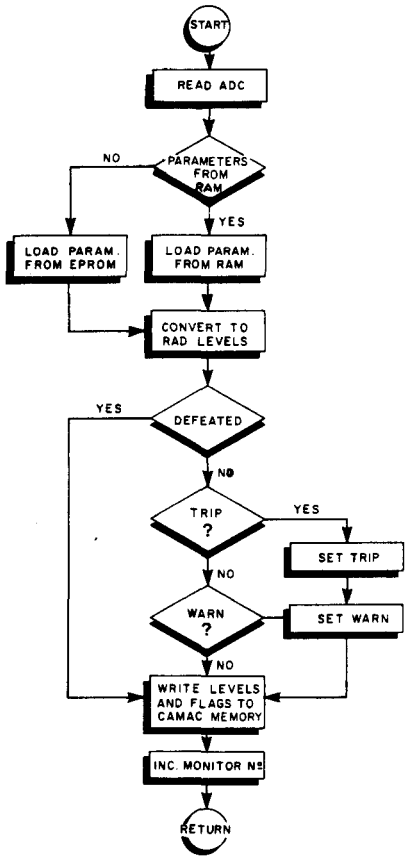

Fig. 3. Flow diagram for beamspill monitor read and process task.

stored in RAM and may be changed by means of the command (CMD) task. A set of default parameters is resident in EPROM.

The bias voltage levels for the beam-spill/area monitors are set depending on the access status of the exclusion areas.

COMMUNICATIONS

The transfer of data from the processing CAMAC crate to the display crate and to the central control system is achieved via CAMAC to CAMAC links. The data is written from an output register to a dual port CAMAC memory. In this way the processing TRIMAC is not kept busy with communications protocols and the data is accessible asynchronously.

Operator input is achieved via a keyboard terminal. The command task accepts a number of valid commands which allow looking at and changing parameters, displaying defeats, resetting to default values and restarting with RAM or EPROM stored parameters. Access to this terminal is subject to strict administrative procedures.

The colour monitor is the only part of the system immediately accessible to the accelerator operators. Radiation levels are displayed in the form of bar graphs with appropriate colour changes when warning or trip set points are exceeded. The monitors are arranged in groups by area and type and a numeric key pad allows the selection of a specific group by entering a 'page' number corresponding to that group.

SUMMARY

The radiation monitoring system we have designed is compact and readily adaptable to process and display data from a variety of detectors. Since it is built from standard CAMAC modules it is possible to take advantage of improvements in that technology (e.g. higher density modules) without rebuilding the entire system. It is especially suitable for accelerator laboratories where the computing expertise is usually readily available.

REFERENCES

1. J. Drozdoff, et al., The Safety Interlock System at TRIUMF, these proceedings.
2. Designed by 'Kadak Products Ltd.' of Vancouver, B.C., Canada.

THE SAFETY INTERLOCK SYSTEM AT TRIUMF

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ABSTRACT

The TRIUMF central safety system is a microprocessor-based interlock system that handles both the personnel access control functions as well as many machine protect functions. The latter have been included in the system because of its high reliability. There are twelve separate exclusion areas each with its own small local controller for monitoring area status and lock-up sequences. All inputs to the central system are in the form of isolated contact closures indicating the safe status. There are at present some 300 such inputs. The interlock conditions reside in erasable programmable read-only memory (EPROM) in the form of a sequence of logic equations in Boolean algebra. Whenever the system senses a change in state of any of the input parameters all the equations are scanned to evaluate the output parameters. Permissives are then generated as +24 volt D.C. levels in accordance with the logic equations. A large, page-selectable CRT monitor displays the status of all input devices as well as the state of the permissives. Operator input is provided via a CRT touchpanel. Changes to the logic equations are verified on an identical system used as a simulator before being burned into a new set of EPROMs. Making a change to the system thus requires only a few minutes of down-time. This system has been in operation for several years with a high degree of reliability.

INTRODUCTION

The central electronics for the TRIUMF safety system consists of two CAMAC crates and related interfacing equipment. Intelligence for both crates is provided by a single Kinetics 3880 microcomputer which reads 24 volt binary inputs, executes a series of logic equations, and generates both internally defined intermediate values and 24 volt outputs. The conditions that cause these outputs to be set essentially define the central safety system; the outputs enable access key release units, drive beamstops, annunciate alarms, and in general, ensure the cyclotron is operating safely. Although the crates operate as a stand-alone system, the use of a CAMAC dual port memory allows the transfer of data to and from the TRIUMF central control system. The status of all input, output, and intermediate value bits is available in this memory to the control system computers which perform the display tasks. Figure 1 shows an overall schematic of the system hardware.

1) Inputs

All inputs to the central safety system are in the form of 24 volt D.C. signals. The 24 volt level is provided by the safety system bus at a central break-out panel and is derived from an uninterruptable power supply. Device status is indicated by an isolated contact closure, the closed contact being interpreted as a logical '1' or safe condition. The system is thus 'fail safe' in the sense that a localized power failure or a cut or disconnected signal cable indicates a logical '0' or unsafe condition. Figure 2 illustrates how the status of two typical contacts reaches the microcomputer via the Kinetics 3471 input registers.

Routing all inputs through a central break-out panel (BOP) also provides the capability of simulating any input should contacts fail or require over-riding. Safe states are simulated by shorting the input with a jumper consisting of a

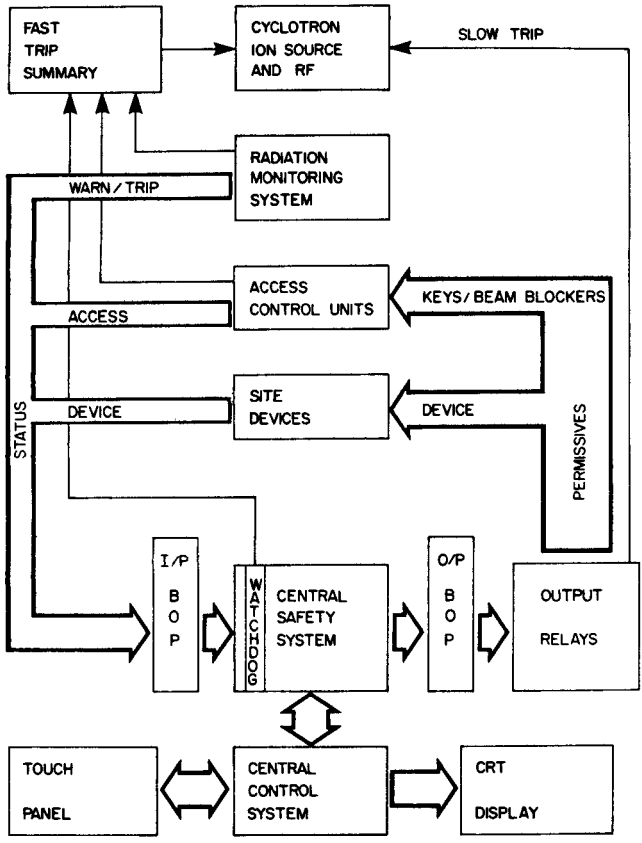

Fig. 1. Overview of the TRIUMF safety system.

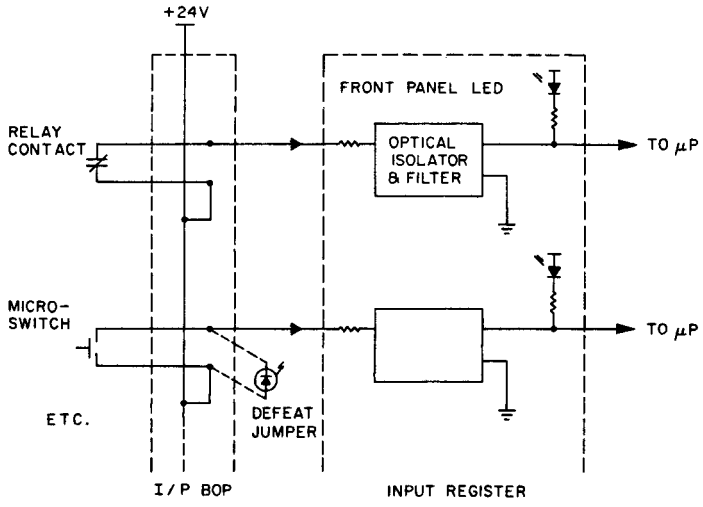

Fig. 2. Typical input configuration for the TRIUMF safety system.

bi-polar LED. In this way the real status of the input is immediately apparent, the LED being 'ON' only if the contact closure is open. A timer inside the locked input panel must be acknowledged once per shift by the operators to force a checking of defeats against a log. Defeating of input signals can only take place over the shift supervisor's signature.

2) Design of Firmware and Logic

The conditions for safe operation of the facility are currently defined by a set of 385 logic equations executed by a Kinetics 3880 microcomputer based on the 8 bit 8080A microprocessor, although each revision typically adds a few more statements. The program is written as a set of equations in Boolean algebra, using mnemonics to represent input and output variables, and the symbols *, + and / to represent the Boolean operators AND, OR, and NOT respectively.

The first section of logic generates a series of internally stored intermediate variables which evaluate expressions repeatedly used in other equations in order to shorten computing time. Much of this section of code is devoted to generating "truly in" (IN AND NOT OUT) and "truly out" (OUT AND NOT IN) values for every device that has two limits. A typical equation defining the cyclotron off condition is:

YCYCOF = YISOF * (MM + RF)

i.e. cyclotron off = ion source off AND either Main Magnet off OR RF system off.

The next section of code uses the intermediate variables and input states to generate hardwired 24 volt outputs. The remaining lines of logic coding create more intermediate values using the output states in the logical expressions. These are used exclusively for display purposes.

The equations are 'stored' in EPROMs located in a Kinetics 3816 memory expansion unit. One single 'base' EPROM resides in the 3880 and executes CAMAC cycles and controls two software timers used in the access key release function. The microcomputer continuously scans all inputs without actually executing the logic equations. When it detects a change of state of any input, however, it runs through all logic equations, executing each once. While a read cycle takes about 5 ms, the time required to execute the logic is 70 ms. On every read cycle the microcomputer strobes a Joerger Watch dog WT module. If the watch dog is not strobed within the specified time (100 ms), a relay contact in the module opens and trips the ion source via a fast direct line. Such direct lines paralleling the computer system are also provided for two other functions; the trips from the radiation monitoring system caused by excessive beamspill and the trips from the emergency push button summaries of the controlled access areas.

3) Outputs

Hardware outputs are generated in eight 32 channel GEC 1617 output registers. These sink current to ground energizing 24 volt relays (Fig. 3) and powering a green LED. 24 volts is sent to a normally open contact of each relay via a 24 volt bus and it is this voltage which is used to turn on status lights, sound alarms and enable site devices. Thus a power failure disables all devices.

4) Communications

The central safety system writes the current status of all inputs, intermediate values, and outputs to a bi-directional CAMAC memory in the central control

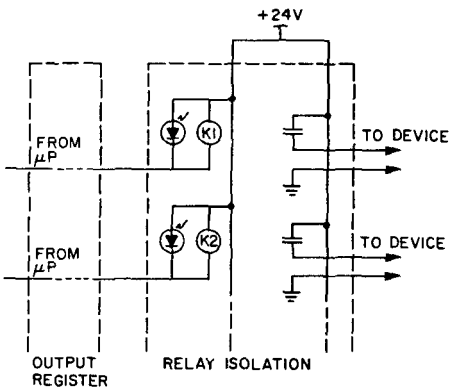

Fig. 3. Typical output configuration for the TRIUMF safety system.

system. The computers of the control system then generate the display on a dedicated CRT monitor. The display is page selectable and there is a general summary page as well as a page for every access control area and major system. Operator control is exercised through a Kinetics 5209 Touchpanel, a device consisting of a 22.5 cm CRT monitor with a touch-sensitive transparent screen. Imbedded in the screen is a 4 × 4 matrix of capacitance switches so that the legend and function of the 16 switches can be changed at will under computer control. Currently any one of 15 existing pages can be called up from an index page. A separate push button must first be pressed to activate any of the touchpanel switches in order to prevent accidental or inadvertent use of the panel. Touchpanel inputs are relayed to the safety system by the central control system but are treated like all other inputs in the logic equations.

5) Altering the Logic

The logic equations written in Boolean algebra are normally edited on the University of B.C. Amdahl V8 computer. We have written a compiler which generates a code composed of macro calls from the Boolean logic equations and replaces input mnemonics with 3880 address codes. The compiled logic is then transferred to a development system, linked with the required CAMAC and timer routines and tested on a simulator. The simulator is virtually identical to the central safety system except that inputs are simulated by a panel of shorting pins and outputs by a panel of LEDs.

Once the program has been debugged it is burned into a set of EPROMs and given a final check on the simulator.

6) Experience and Reliability

The central safety system as described has been operating without substantial interruption since December of 1978. The only regular 'down-times' are approximately 10 minute long periods occurring at intervals of one to two months in order to replace the set of logic equations to accommodate changes required by alterations and additions to the site.

There has been one failure of the central system since it was installed, the failure being in the uninterruptable power supply to the CAMAC crates. Due to the fail safe design no unsafe condition occurred during this failure.

MICROPROCESSOR BASED DIGITAL RADIATION METER/MONITOR

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INTRODUCTION

Digital radiation meters/monitors,which are essentialy digital ratemeters, have been always very atractive for instrumentation designers since digital instruments have advantages in applications when compared with analogue ones (Ref.1).

The integrated electronics allowed the digital ratemeter to be realized in a convenient size with facilities inherent to digital techniques, but not all of them have been included as yet (Ref.2).

By invention of microprocessor in 1971 these problems have been overcome and it was possible to realize not only fundamental tasks but to offer quite new sollutions (Ref.3.4).

The microprocessor, as a component with defined and fixed structure, performs arithmetic and logic operations by a set of instructions i.e. by a given program. Programing depends on the ability of a programer and offers an unlimited variety of different functions including those of interest for digital ratemeters i.e. radiation monitors.

MICROPROCESSOR BASED INSTRUMENT STRUCTURE

The pulse train from radiation detector and the time interval during which the pulses are registered are sources of information for digital radiation meters. Both information are fed into the separate inputs of digital ratemeter which is an input part of a microprocessor based instrument shown in fig.1. The functioning of a digital ratemeter is determined by the program instructions applied on the microprocessor and thus the instrument becomes digital radiation meter. The single measuring cycle duration is determined by preset numbers of pulses in the two programable comparators: for detector pulses and for clock pulses. It should be noted that the predetermined numbers in the programable comparators are set by the microprocessor. The bus line organization is used in the structure, the general purpose microprocessor is assumed, and the operation is evident from fig.1.

In one measuring cycle the digital ratemeter at the output provides the number of detector pulses /N_C/ registered during the measuring time /T_C/ as well as the number of registred clock pulses, defining T_C. These elementary data, Nc and Tc, can be a final result for counting rate i.e. dose rate, or may undergo a further treatment to get more accurate results and to deduce different information. Simultaneously microprocessor can provide information on: tendency and/or velocity of changes, maximal or minimal value measured during the past period, the statistical error of results, integral dose; it can also accomplish comparison with predetermined values and so on.

Thus the microprocessor based radiation meter/monitor is potentialy a multifunctional instrument and it should be designed in that way.

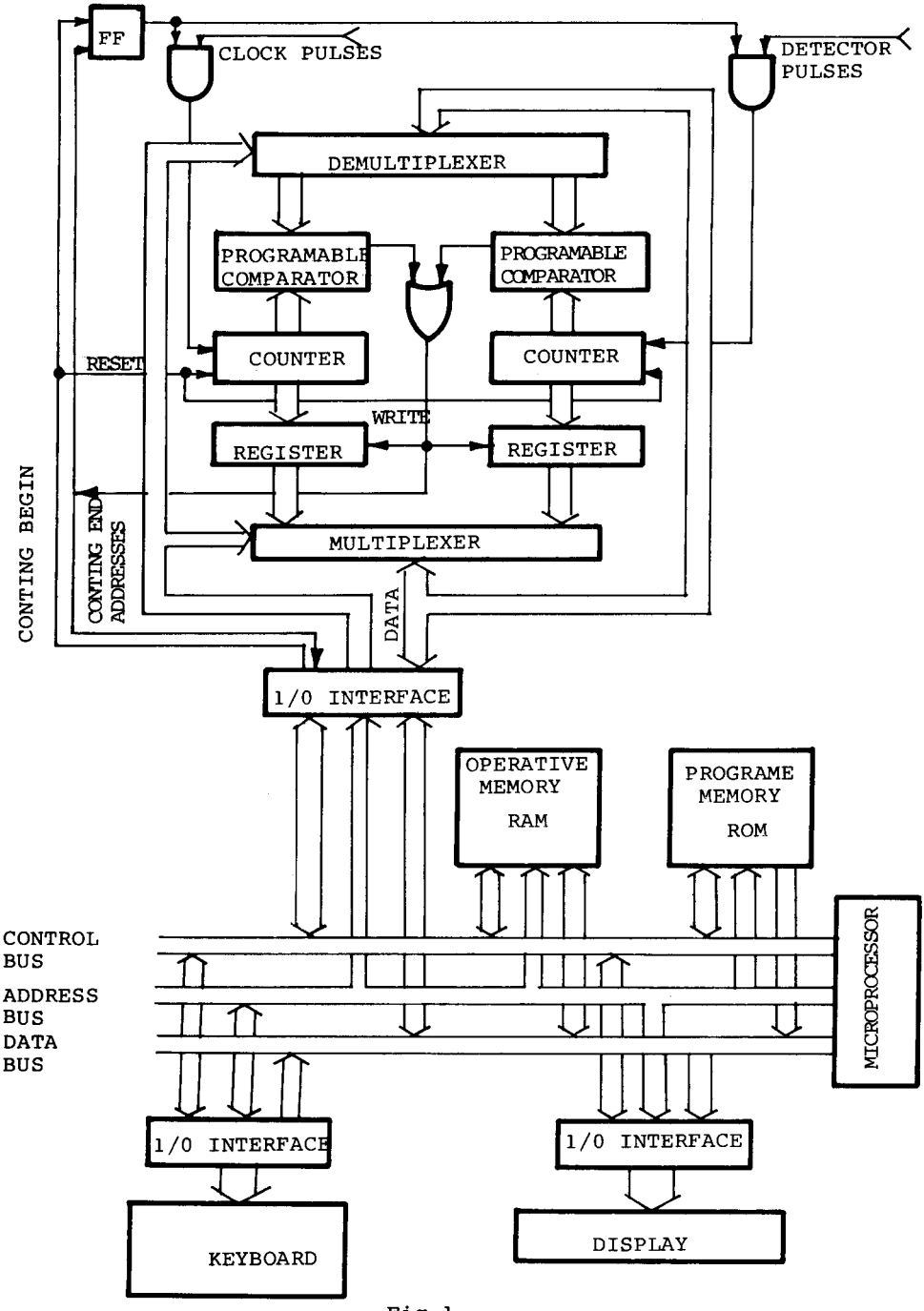
FF
CLOCK PULSES
DETECTOR PULSES
DEMULTIPLEXER
PROGRAMABLE COMPARATOR
PROGRAMABLE COMPARATOR
RESET
COUNTER
COUNTER
WRITE
REGISTER
REGISTER
CONTING BEGIN
CONTING END ADDRESSES
MULTIPLEXER
DATA
1/0 INTERFACE
OPERATIVE MEMORY RAM
PROGRAME MEMORY ROM
CONTROL BUS
ADDRESS BUS
DATA BUS
MICROPROCESSOR
1/0 INTERFACE
1/0 INTERFACE
KEYBOARD
DISPLAY

Fig.1.

INSTRUMENT SOFTWARE

The main consequence of microprocessor presence in the instrument is that the digital radiation meter/monitor functions are not any more accomplished by the hardware but by a convenient program built in the microprocessor based instrument.

The basic monitoring function - dose rate measurement - can already be obtained by a simple algorithm. A direct imitation of an analogue ratemeter with the diode pump circuit, is inconvenient and unreasonable for a digital ratemeter. Meanwhile it is possible to use a very simple algorithm giving the basic result:

a) - after termination of the measuring cycle calculate the partial result: $R_p = N_c/T_c$;

b) - form the sum S_k of k partial resuls: add the n^{th} partial result and substract the $(n-k)^{th}$ partial results;

c) - devide the sum S_k by k, so find the average rate

$$R = \left(\sum_{n-k+1}^{n} R_p \right)/k$$

d) - return to a).

When all single measuring intervals are equal (T_c = const) this algorithm provides linear change of the result for the step change of measured radiation field.The time to reach the final result and steady state depends on k, as well as the error of a result in the steady state.

Other algorithms for the dose rate can also be formed with different characteristics and conveniences (Ref. 5) and generaly one should find the algorithm which would include microprocessor versatility and offer good quality result.

Using a digital ratemeter in this way only, we have not gained too much from a microprocessor. Microprocessor allows us to extend the algorithm and from the same data to calculate additional quantities, allready mentioned above: the accuracy of dose rate, tendency of change, integral doserate, minimal or maximal value in determined measuring period, etc. All these additional quantities improve the measuring qualities of a radiation meter/monitor and may be usefull in practice.

Besides, one should not forget that different types of corrections can be introduced in the algorithm of data treatement: temperature and presure corrections, detector nonlinear characteristics and dead time losses corrections etc.

To have realy multifunctional instrument and to use fully the microprocessor possibilities the algorithm of data treatment might be complex and based on quite new approaches.

The structure given in fig.1. is very flexible, although not the only possible. There the microprocessor controls completely the structure functions, i.e. it is program controled, what is very convenient. This permits the designer to create extremely sophisticated algorithm for acquisition and treatment of data, in order to develop new better solution and to suit the users.

CONCLUSION

It is evidently justified to introduce a microprocessor in the structure of radiation meters/monitors. Its use offers all facilities and conveniences of digital techniques, which are numerous and may be of interest in health physics instrumentation design practice. In such a way the radiation meter/monitor becomes the multifunctional instrument, with many useful characteristics increasing its qualities and influencing measurement methodology and philosophy.

One can expected that the future radiation meters/monitors will mainly be microprocessor based, and the proposed structure is the one that can be used for this purpose conveniently.

REFERENCES

1. C.H.Vincent: Randdom Pulse Trains - Their measurement and Statistical properties
IEE monograph Series 13, Peter Peregrinus Ltd. London, 1973

2. I.D. Vankov, L.P.Dimitrov: Digital Ratemeter,
Nucl.Instr. and Meth., v.188(1981), pp.319-325

3. R.Maushart: Recent Developments in Radiation Protection Instrumentation
XI Reg.Congress of IRPA, Vienna, 20-24 IX.1983

4. A.M.Koturović, R.B.Vukanović: An Outlook on Development and Trend of Microprocessor Based Radiological Instrumentation
XI Reg.Congress of IRPA, Vienna, 20-24 IX.1983

5. A.M.Koturović: The Rate Estimation of Random Pulses Trains by Microprocessor Based Ratemeter
To be presented at: "Measurement and Estimation", 4th Symposium of the IMEKO Tech.Com. on Measurement Theory, Bressanone, Italy, May 9-12 1984

A MODERN AUTOMATIC READ-OUT SYSTEM FOR PHOSPHATE GLASS DOSEMETERS

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1. INTRODUCTION

After the high efforts in developing automatic TLD systems, the producers of RPL systems lost their interest in this field if not stopped their activities in producing readers and glass dosemeters. Germany seems to be the only country where RPL dosemeters are still used in an old fashioned design of 1964 as a replace or support of film dosemeters in routine personnel monitoring [1-3]. RPL systems,not comparable with TLD systems mainly in the lower detection limit are still superior to TLD systems with respect to the simplicity of the read-out procedure, the excellent batch uniformity, the long-term stability and permanent availability of dose information.

The actual interest in RPL dosimetry today is partly based on some how disappointing experiences with commercially available automatic TLD systems [4], but mainly on the advantages of glass dosemeters with respect to an automatic read-out and the opportunity to optimize energy compensation filters in order to separate dose fractions of different energy ranges or to indicate different kind of operational dose quantities of interest practically independent of photon energy above 15 keV.

In the last years the activities of our laboratory have been again focused to the construction of a commercial prototype reader for a fully automatic RPL system. With respect to the RPL system the following objectives have been considered:

- the dosemeter encapsulation, dust-tight and locked by a magnetic latch will be automatically opened within the reader,

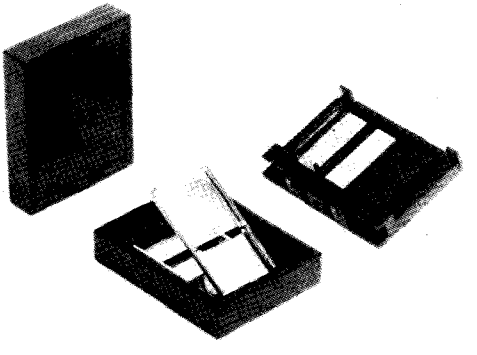

Fig. 1: KfK phosphate glass dosemeter for automatic opening

- two types of magazins are used, one for dosemeters, another one for dosemeter cards, which allows an exchange between used and annealed dosemeter cards,
- with respect to UV excitation the read-out system is applicable for conventional mercury lamp as well as N_2 gas laser,
- two different code systems identify the number of card and capsule,
- the microprocessor controlled evaluation technique allows a flexible change of the read-out procedure,
- the system has been optimized with respect to the dosimetric properties, i.e. reduction of pre-dose, standard deviation, energy and directional dependence.

2. DESCRIPTION OF THE RPL SYSTEM

2.1 Dosemeter

The dosemeter design presented in Fig. 1 makes use of a Toshiba glass D-7 or Schott glass DOS-8 of the size 16 x 16 x 1.5 mm^3 fixed in a stainless steel holder (dosemeter card) bearing the dosemeter number (hole code). In a plastic encapsulation the dosemeter card is shielded from both sides by a flat energy compensation filter consisting of a two-part tin filter of 1 mm in thickness with a plastic part of 3.3 mm width. A magnetic latch avoids an unauthorized opening and a rubber closes the dosemeter dust-tight. The encapsulation is marked with a number and the name of the user.

2.2 Read-out system

The read-out system makes use of a mercury lamp for UV-light excitation all over the glass volume and, on the other hand, only in the center of the glass area 1 x 1.5 mm^2 with an UV-light beam of 1.5 mm Ø. The photomultiplier in front of the glass area 16 x 16 mm^2 registers the RPL light from the glass. The read-out system is thus applicable for common as well as UV-laser beam excitation.

The commercial prototype of a fully automatic RPL system now under construction makes use of a moving grap in the center of the reader which allows the transportation of the dosemeter to all stations at the peripherie. The reader contains a dosemeter magazin for a large scale evaluation as well as a slot for the input of single dosemeters. The stations for the read-out cycle are the following: registration of the dosemeter number on the capsule, automatic opening of the dosemeter, registration of the number of the dosemeter card, UV-excitation and read-out of the dosemeter card in the optical chamber.

In routine operation, the dosemeter card can be stored in the card magazin if a washing step is needed before read-out or if a high reading requires an annealing step before reuse. In the latter case the glass is exchanged by a new annealed one from the card magazin which after pre-dose read-out will be closed in the dosemeter capsule and transported to the dosemeter magazin for reuse. About 300 dosemeters can be measured simultaneously without re-loading the reader.

The complete read-out system is microprocessor controlled. This easily allows changes in the read-out procedure and a comfortable

self-check system for reliable read-out of many dosemeters with a high reading rate. An on-line desk computer is used for the identification of the actual dosemeter capsule and card numbers as well as for the data processing of the pre-dose reading before the monitoring period, the random error of measurement, the dose accumulated in the past monitoring period, the total accumulated dose of a person, as well as dose accumulations required by regulations. Personal data can be stored in a data-file for a group of 10.000 persons.

The modified automatic evaluation allows daily entrance and exit controls by the user himself.

3. DOSIMETRIC PROPERTIES

Using a cylindrical polyethylene phantom of 30 cm diam. the energy dependence of the glass dosemeter (Fig. 2) has been found to be within ± 15 % for photon energies above 15 keV for the measurement of the dose equivalent H_{10} in a tissue depth of 1 cm as well as within ± 15 % above 20 keV for the measurement of the dose equivalent $H_{0.07}$ and H_3 in a tissue depth of 0.07 for the unprotected skin and of 3 mm for the lens of the eye. Free in air, the dosemeter indicates the exposure within ± 20 % above 23 keV. The mean energy and angular response ranging for 0° to 60° (Fig. 3) has been found to be about ± 30 % above 25 keV and thus comparable to TLD dosemeters. The RPL dosemeter system generally allows to apply different calibration factors or other energy compensation filters or other read-out techniques to measure the dose equivalent in different specific depths of tissue or to analyse the radiation field [5,6].

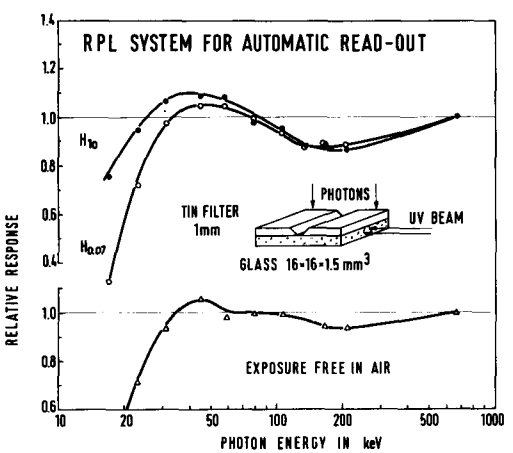

Fig. 2: Energy response of the flat glass dosemeter

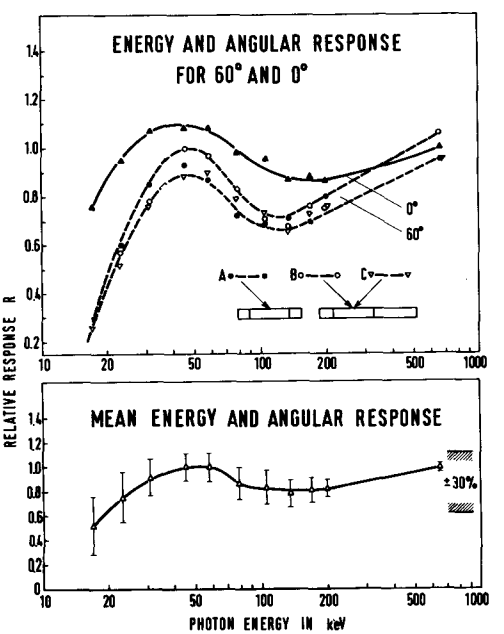

Fig. 3: Energy response for H_{10} and angles of 60° and 0°

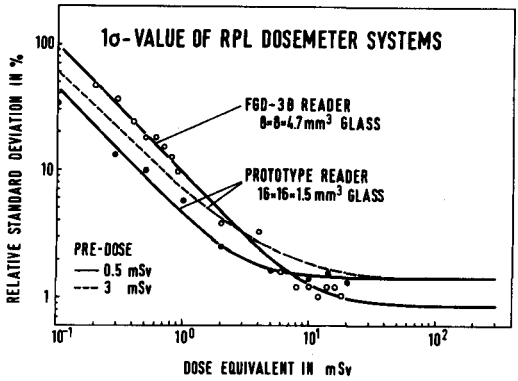

Fig. 4: Relative standard deviation vs. dose

Taking into account the subtraction of the pre-dose namely 0.5 mSv after annealing and 3 mSv after pre-exposure, the random statistic uncertainty of dose measurement is presented in Fig. 4 as a function of dose. The empirical standard deviation calculated from the readings of 10 dosemeters indicates the short-term reproducibility of the read-out system as well as influences of the washing procedure. Thus, a dose of 0.2 mSv can be measured with a standard deviation within ± 20 % for the new RPL system and ± 50 % for the old Toshiba FD-3B reader and the glass size 8 x 8 x 4.7 mm³.

Further characteristics of RPL systems are the capability for repeated measurements, the linearity of the dose range from 0.1 mSv up to 30 Sv and the long-term stability of information (for FD-1 glasses only 10 % fading have been found after 10 years storage at room temperature).

The automatic read-out procedure of a one-element dosemeter offers new aspects in personnel monitoring so far not offered by TLD systems. These are the permanent availability of dose information during excit controlls, i.e. repeated daily read-outs and at the end of the monitoring period subsequent control readings at an independent central laboratory. A pilot test of such a one-element dosemeter with repeated read-outs at the facility and a control read-out at the central laboratory at the end of the monitoring period has been recently performed resulting in an agreement within ± 3 % for the total accumulated dose

REFERENCE

[1] Piesch, E., in Topics in Radiation Dosimetry, Acadmic Press, p 461 - 532, 1972
[2] Piesch, E., Regulla, D.F., Proc. 5th Int. IRPA Conf. 1, p 343, 1980
[3] Regulla, D.F., Piesch, E., One-element personal dosemeter for local and centralized evaluation, Proc. Int. Conf. Solid State Dosimetry, Ottawa 1983
[4] Burgkhardt, B. et.al., Proc. Jahrestagung des Fachverbandes für Strahlenschutz, München, FS-83-30-T, 133, 1983
[5] Röber, H.-G., Burgkhardt, B., Piesch, E., Nucl. Instr. Meth. 175, p 131, 1980
[6] Burgkhardt, B., Piesch, E., Röber, H.-G., see [3]

PERSONNEL NEUTRON DOSIMETRY BY CR-39 PLASTICS WITH CHEMICAL ETCHING, ELECTROCHEMICAL ETCHING AND THEIR COMBINATION.

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INTRODUCTION

One of the major problems in personnel neutron dosimetry is the measurement of neutrons having energy below 500 keV and above the upper limit of the albedo at about 10 keV (Griffith and co-workers, 1979). Since substancial components of neutron dose-equivalent are produced by neutrons with energy within this energy interval (Enders and others, 1981), a dosimeter which efficiently registers these neutrons is urgently needed.

In the last few years great impetus has been taken on the development of a personnel dosimeter based on the proton-sensitive CR-39 track detectors (Cartwright and others, 1978; Benton and others, 1981; Tommasino and others, 1980). In this paper the advantages and limitations of different etching processes for CR-39 detectors are analysed.

Chemical Etching.

When chemical etching is used, the poor visibility for optical counting (specially for small-range particles) makes it difficult to discriminate against artefacts (scratches, dust particles, surface pitting etc.). Cross and co-workers (1982) have reported the large discrepancies encountered between the experimental results and the calculate response at energies below 1 MeV for chemically etched CR-39 detectors. These discrepancies can be mainly attributed to spurious counting due to surface pitting or other background defects, which can not be distinguished from short-range tracks.

Fig. 1 illustrates additional reasons for the poor response of chemically etched detectors to low energy neutrons. In this figure, the formation of proton-recoil tracks is shown schematically at different etching times t_1, t_2..... etc. In the case of chemical etching (upper part of the figure), the tracks a and b (due to 100 keV protons) remain pointed until they are etched to the end of their trajectory. Since prolonged etching is usually required in order to be able to etch high-energy recoils (such as the 2-MeV-proton track, c) both tracks a and b become over-etched and disappear from view. At the end of etching, thick detector layers (5-15 microns) are etched away from the original surface and most of the short-range tracks due to low-energy neutrons are lost by these processes.

Electrochemical Etching with and without Pre-Etching.

The electrochemical etching (ECE) makes it possible to overcome the difficulties of counting low densities of short-range tracks (Tommasino and co-workers, 1981). This type of etching is a two-steps process consisting in the stage of track formation followed by that of tree initiation and propagation.

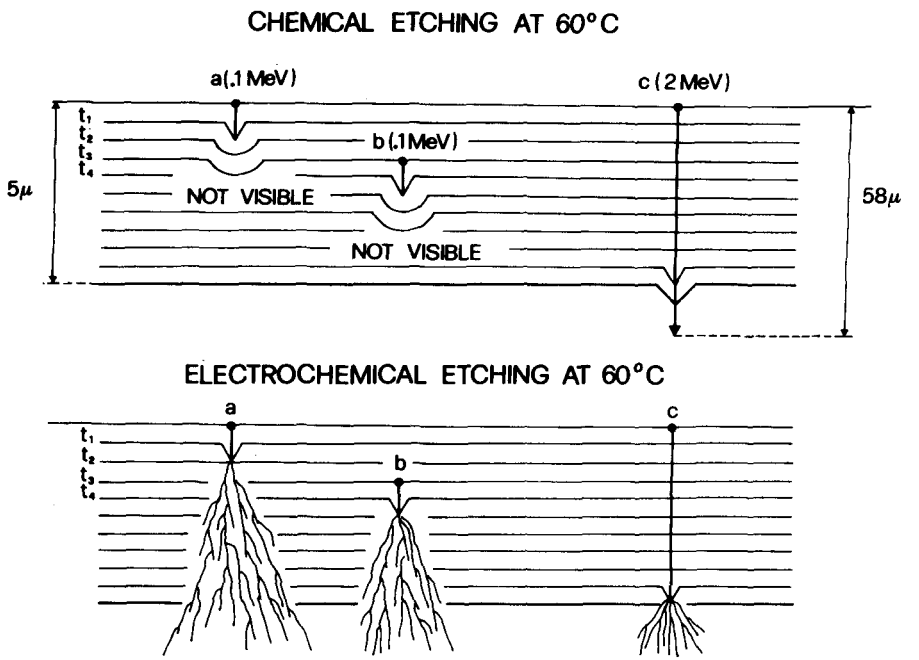

Fig. 1 Schematic diagrams for the formation of tracks both under chemical and electrochemical etching.

In attempts to reduce the background, a temperature of 24 °C is very often used for the electrochemical etching, which then is only capable of producing ECE-spots at tracks previously formed by heavy chemical etching at 60 °C (Espinosa and co-workers, 1981). The electrochemical etching of pre-etched detectors provides excellent registration characteristics for mono-energetic charged particles since they have the same time for track formation. In this case a single pre-etching is suitable to achieve the formation of pointed-shape tracks for all the charged particles, which can be then efficiently registered by ECE processes.

Unfortunately, no single chemical pre-etching exists for the formation of all the proton-recoil tracks induced by neutrons (Cross and others, 1982), which may have energies anywhere in an interval about three orders of magnitude wide (10 keV-15 MeV). Of all the pre-etched tracks, only a small fraction is enlarged by ECE processes since the formation of tree spots occurs preferentially at pointed-shape tracks. The dashed line of Fig. 2 shows the response versus the neutron energy for CR-39 detectors etched electrochemically at 24 °C with 30 kV/cm rms and 2 kH_z after 5 hours pre-etch at 60 °C. This energy response has a shape which can be explained considering that only a small fraction of proton tracks produces ECE spots at low and high neutron energies (Cross and others, 1982). In fact for CR-39 detectors

irradiated to protons and pre-etched under the same above conditions, the most efficient production of ECE spots has occurred for protons with energy between .6 and 2 MeV. In spite of the excellent conditions for automated spot counting and negligible overlapping (Espinosa and others, 1981) these combined etching processes have met with limited success essentially because the shape of the response is not very good for neutron dosimetry.

To exploit all the advantages of ECE processes, the track formation should occur during the ECE process, so that each track will have the opportunity to start treeing once has acquired a pointed shape. To this end a temperature of 60 °C, similar to that for the chemical etching, can be used for the electrochemical etching. During ECE processes at 60 °C, those tracks already treeing continue to enlarge, while more tracks are formed and initiate treeing (as shown in the lower part of Fig. 1). This type of etching can be considered just the converse of the chemical etching at the same temperature, since the low-energy recoil tracks keep enlarging under ECE processes, while they disappear from view or are difficult to recognize under conventional etching.

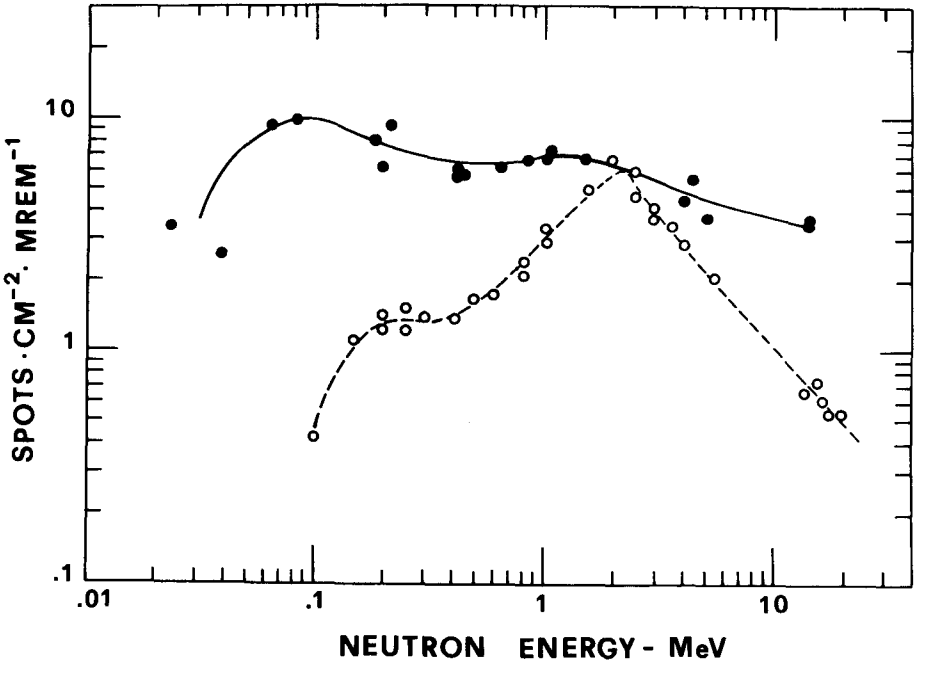

Fig. 2 Neutron energy response of electrochemically etched detectors with (open circles) and without (closed circles) pre-etching.

If the tracks are formed at the beginning of ECE processes, they will grow to a greater extent than those which will be formed at a later time, thus resulting in very different track diameters. When electrical fields of 30 kV/cm rms and 2 kH_z frequency have been used, the rapid growth of ECE spots saturates the detector response at only a few rems of fast neutrons. To reduce the propagation rate of ECE spots, a frequency of 50 H_z has been chosen which, among other things, has the advantage of being the line frequency.

The remarkable improvements of the neutron energy response of CR-39 detectors electrochemically etched for 5 hours at 60 °C with 30 kV/cm rms and 50 H_z are apparent in Fig. 2, where it is possible to compare this response with that obtained by electrochemical etching with 5 hours pre-etch. The little dependence of the response of this dosimeter versus the neutron energy over a very wide energy range (50 keV-15 MeV), together with the simplicity of track counting and automation of ECE spots represent unique and attractive characteristics for personnel neutron monitoring.

REFERENCES

Benton, E.V., Frank, A.D., Osvald, R.A., and Wheeler, R.V. (1981). Proton-recording neutr.dosimeter for personnel monitoring. Health Physics, 40, 801-810.

Cartwright, B.G., Shirk, E.K., and Price, P.B. (1978). A nuclear track recording polymer of unique sensitivity and resolution. Nuclear Instr. Meth., 153, 457-460.

Cross, W.G., Arneja, A., Ing, H. (1982). Background and neutron response of electrochemically etched CR-39. Ninth DOE Workshop on Personnel Neutron Dosimetry, Las Vegas, June 24-25.

Endres, G.W., Aldrich, J.M., Brackenbush, L.W., Faust, L.G., Griffith, R.V., Hankins, D.E. (1981). Neutron dosimetry at commercial nuclear plant. NUREG/CR 1769; PNL 3585

Espinosa Garcia, G., Golzarri, J., Tommasino, L., and Raponi, F.(1981). Electrochemical etching registration and track formation time. Proc. of the 11th Intern. Conf. on Solid State Nuclear Track Detectors, Bristol 7-12 Sept., 241-244.

Griffith, R.V., Hankins, D.E., Gammage, R.B., Tommasino, L., and Wheeler R.V. (1979) Recent developments in personnel neutron dosimetry. A review. Hth Phys., 36, 235.

Tommasino L., Zapparoli, G., Griffith, R.V., Fisher, J.C. (1980). Electrochemical etching of CR-39 foils for personnel fast neutron dosimetry. Proc. 5th Int. Congress of IRPA, Jerusalem, March 9-14, Vol. II, Pag. 205-209.

Tommasino, L., Zapparoli,G., and Griffith, R.V. (1981). Electrochemical etching I Mechanisms. Nuclear Tracks, 4, 191-196.

MESURE DU RADON DANS LES HABITATIONS COMPARAISON ENTRE LES DETECTEURS ACTIFS ET PASSIFS

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INTRODUCTION

La mesure de l'exposition naturelle des populations est une des préoccupations actuelles en radioprotection ; l'exposition interne due aux descendants du radon et du thoron semble y jouer un rôle important.

La radioactivité naturelle de l'air dans les habitations varie avec de nombreux facteurs et une mesure instantanée est imprécise. Il est préférable de faire une mesure en exposant les détecteurs pendant un ou plusieurs mois. Même dans ces conditions il est possible de trouver plus d'un ordre de grandeur de différence d'activité entre les différentes pièces d'une même maison. Ceci oblige à multiplier les points de mesure.

Actuellement deux systèmes de mesures sont utilisés. Un système actif (A) basé sur le principe du dosimètre individuel des mineurs [1] développé par PINEAU [3] et qui permet de mesurer directement l'énergie potentielle α du dépôt actif. Un système passif (P), constitué par des feuilles de nitrate de cellulose (LR 115 de Kodak) nues, sensible non seulement aux descendants actifs, mais également aux précurseurs Rn et Tn ; dans ce cas il s'agit d'une mesure d'activité α. L'étalonnage du détecteur est réalisé dans une ambiance où la teneur en radon est connue : la stabilité du rendement est contrôlée par comptage d'une source α épaisse.

RESULTATS

Le but de cet exposé est de comparer les résultats, les avantages et les inconvénients des deux systèmes. Les informations proviennent de mesures effectuées simultanément au moyen des systèmes A et P, soit en France, soit dans le cadre d'une intercomparaison avec le N.R.P.B. au niveau de la C.C.E.[2].

Les résultats de mesure sont pris en compte si le système actif a fonctionné correctement, et si l'erreur de comptage du détecteur passif est inférieure à 20% (2σ)

- Pour la région parisienne 11 couples de résultats ont été ainsi retenus après une exposition des détecteurs pendant un mois. Les valeurs de la concentration en radon trouvées (Tableau 1) avec les détecteurs P s'échelonnent de 0,3 à 3 $pCi.l^{-1}$. Celles de l'énergie potentielle totale avec le système A de 180 à 1 290 $MeV.l^{-1}$. La valeur que l'on cherche à déterminer, c'est-à-dire celle de l'énergie potentielle α par picocurie ainsi mesurée est en moyenne de $\underline{438} \pm \underline{138}$ $MeV.pCi^{-1}$. Dans la région marseillaise, 7 couples de résultats ont pu être retenus après 20 jours d'exposition. La valeur moyenne de l'énergie potentielle α est de $\underline{447} \pm \underline{103}$ $MeV.pCi^{-1}$ (1 résultat exclu). Pour ces mesures effectuées dans les habitations, le système A montre que l'énergie potentielle du thoron représente 10 à 60 % de l'énergie totale. L'énergie potentielle par picocurie ne semble pas sensible aux variations du pourcentage du Tn. En Bretagne, pour 7 couples de mesures, l'énergie potentielle moyenne par picorurie est de $\underline{545} \pm \underline{240}$ $MeV.pCi^{-1}$.

- Trois types d'essais ont été effectués dans le cadre de l'intercomparaison internationale.

. Le premier type est semblable à celui des mesures habituelles : les détecteurs passifs ont été placés dans un bureau pendant 83 jours. L'estimation de l'énergie potentielle a été faite par le N.R.P.B.. Le rapport trouvé est de $\underline{513}$ $MeV.pCi^{-1}$.

. Dans le second type d'essai, les détecteurs ont été placés dans une chambre étanche contenant seulement du radon et ses descendants sans teneur appréciable en thoron. La comparaison de résultats des détecteurs P, avec les mesures d'énergie potentielle α du N.R.P.B., a été réalisée pour trois équilibres distincts. Lorsque le facteur d'équilibre passe de 1 à 0,11, le rapport passe de 1228 à 217 MeV.pCi^{-1}, et pour un facteur de 0,32 le rapport est de 580 MeV.pCi^{-1}.

. Enfin, une intercomparaison des détecteurs actifs entre eux dans la même enceinte a donné un résultat moyen de 128 mWL, le système A indiquant 114 mWL. Le facteur d'équilibre est de l'ordre de 0,5. Les détecteurs P ont été exposés pendant la durée de l'expérimentation soit 18 heures. Comparé avec le résultat A, le rapport est de 523 $\pm$ 109 MeV.pCi^{-1}.

L'ensemble de ces mesures montre que dans les conditions réelles, un facteur sensiblement constant de l'ordre de 500 MeV.pCi^{-1} permet d'apprécier raisonnablement l'énergie potentielle des descendants présents dans une maison par la simple mesure de l'activité α globale de l'air au moyen de LR 115. Dans les cas extrêmes il est possible que l'erreur soit un peu supérieure à un facteur 2 par rapport à une méthode basée sur l'utilisation d'un détecteur actif.

Certains auteurs utilisent les dosimètres passifs dans un système fermé perméable aux gaz, mais ne laissant pas pénétrer le dépôt actif. Dans ces conditions, la mesure porte essentiellement sur le radon, et le calcul de l'énergie potentielle est effectué en admettant un facteur d'équilibre arbitraire. Dans le système ouvert (type P), tous les α sont mesurés mais l'état d'équilibre est inconnu. Il faut donc choisir un facteur moyen de conversion. La transformation des résultats des systèmes passifs, pour obtenir une énergie potentielle α est donc toujours soumise à un choix, que le système soit ouvert ou fermé.

AVANTAGES ET INCONVENIENTS DES DEUX SYSTEMES

Le tableau 2 montre les avantages et les inconvénients des systèmes A et P. La comparaison devrait être poursuivie pour établir sur une base plus large la validité des résultats des détecteurs passifs par rapport aux détecteurs actifs. On pourrait se fixer l'objet suivant : 90 % des résultats seraient obtenus avec les détecteurs P, qui sont simples et bon marché ; 10 %, obtenus en parallèle avec les détecteurs actifs, permettraient de confirmer ou de préciser certains résultats.

REFERENCES

[1] DUPORT P. et al.
Enregistrement des rayonnements alpha dans le dosimètre individuel et le dosimètre site du Commissariat à l'Energie Atomique.
Solid State Nuclear tracks detector, p. 609-617 Ed. François, Pergamon Press, 1980.

[2] MILES J.C.H., STARES F.J., CLIFF K.D., SINNAEVE J.
Results of a quality assurance exercise for radon and radon decay product measurements.
EUR 8629-EN (1983).

[3] PINEAU J.F.
La dosimétrie individuelle intégrée dans les mines d'uranium et son impact sur les méthodes de prévention.
Radioprotection 17, 2, p. 133-136 (1982).

Tableau 1 - Résultats des Mesures

Origine des Prélèvements	Durée (jours)	Système passif ($pCi.l^{-1}$)	Système actif ($MeV.l^{-1}$)	Energie potentielle ($MeV.pCi^{-1}$)	Energie potentielle Tn/total	Facteur d'équilibre F
		P	A			
	30	0,57	182	319	0,5	
	30	0,66	236	357	0,5	
REGION	30	0,31	238	768	0,4	
PARISIENNE	30	2,25	594	264	0,2	
Habitations	30	0,63	273	433	0,3	
	30	0,50	181	362	0,4	
	30	1,36	502	369	0,2	
	30	2,96	1 292	416	0,2	
	30	0,48	274	570	0,3	
	30	1,70	830	468	0,6	
	30	0,54	254	470	0,5	
	20	0,37	118	318	0,3	
MARSEILLE	20	0,78	892	1 143	0,3	
Habitations	20	1	400	400	0,3	
	20	0,58	227	391	0,4	
	20	0,88	379	430	0,1	
	20	0,40	230	575	0,3	
	20	0,37	210	567	0,2	
	12	4,5	2 810	625	0,2	
	15	2,1	1 560	741	0,3	
BRETAGNE	15	11	7 330	670	0,05	
Habitations	13	7,8	1 360	174	0,5	
	13	34	8 200	241	0,03	
	13	3,2	2 540	794	0,2	
	14	2,8	1 590	569	0,3	
		P	N.R.P.B.			
INTERCOMPARAISON	83	0,76		513		
INTERCOMPARAISON	2	106		1 228		> 0,95
		53		217		0,11
		70		580		0,32
		P	A			
INTERCOMPARAISON	0,75	40		370		∿ 0,5
		30		494		0,5
		28		529		0,5
		27		549		0,5
		22		673		0,5

Tableau 2 - Avantages et inconvénients des deux systèmes

	DETECTEUR PASSIF P LR 115	DETECTEUR ACTIF A
AVANTAGES	- Simplicité - Faible prix - Pas de source d'énergie - Possibilité de multiplier le nombre de points de mesure. - Peut être exposé plusieurs mois - Peu sensible à la poussière - Entretien nul - Faible encombrement	- Mesure directement le paramètre voulu, c'est-à-dire l'énergie potentielle - Richesse des renseignements que l'on peut obtenir sur les familles du radon et du thoron, sur la fraction libre - Peut être exposé plus d'un mois si l'empoussiérage est faible.
INCONVENIENTS	- Mesure des alpha et non de l'énergie potentielle. Hypothèse nécessaire lors de l'interprétation - Sensibilité au rayonnement solaire direct.	- Prix - Matériel plus complexe : maintenance, entretien, mesure de débit. - Encombrement, bruit, nécessité d'une source d'énergie.

SPARK COUNTING OF ALPHA TRACKS ON AN ALUMINIUM OXIDE FILM

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1. INTRODUCTION

Spark counting of etch-pits on polycarbonate film produced by fission fragments due to (n,f) reaction or on cellulose nitrate film produced by α-particles due to (n,α) reaction is known to be very effective for neutron monitoring in the field of gamma-rays near a reactor [1-5], because those films are not sensitive to gamma-rays, which usually accompanies with neutrons in a reactor.

Aluminium oxide neutron detector due to (n,f) reaction with natural uranium was already examined [6]. This time we have tried to use aluminium oxide films as α-particle detectors instead of cellulose nitrate films for above purpose. The merits of this method are that 1) aluminium oxide is good electric insulator. 2) any desired thickness of the film can be prepared. 3) chemical etching of the thin film can be dispensed with.

2. PREPARATION OF ALUMINIUM OXIDE FILM

Aluminium plate (30 x 40 x 0.3 mm^3) was treated with dilute NaOH solution and HNO_3 solution before electrolysis. A nickel plate as a cathod and the aluminium plate as an anode were immersed in 15% sulfuric acid solution for oxidation of aluminium. The solution was kept at 12°C by circulating water fed from a thermo regulator.

Fig.1 shows the relation between electrolytic time and deposited aluminium oxide thickness. The thickness increased with time linearly up to 2 μm. Aluminium oxide film 1 μm ± 10% thick was produced with 30 minutes electrolysis at 6 mA/cm^2 electric current density. After electrolysis the aluminium plate was rinsed with distilled water and kept in boiling water for 30 minutes for pin-hole sealing. The thickness of aluminium oxide film was determined from the weight difference before and after electrolysis.

3. ALPHA-RAY IRRADIATION

The aluminium oxide film was irradiated with α-rays emitted from 500 μg uranium of uranyl nitrate which was set on the film for 40 hours. After irradiation uranyl nitrate was completely washed out.

4. SPARK COUNTING

After α-ray irradiation the aluminium oxide film was attached with an aluminized polycarbonate sheet for spark counting. As shown in Fig.1 high voltage for punching was applied between the aluminized sheet electrode (cathode) and the aluminium plate electrode (anode), which was the base of the aluminium oxide film. After repeated punching sparks sparked pulses were counted with a usual scaler at a lower voltage than the punching one.

5. SPARKING CHARACTERISTICS

The relation between applied voltages and spark (or breakdown) counts was examined for aluminium oxide films of several thicknesses. Spark counts increased as the applied voltage increased as seen in previous report [6]. Passing a plateau it increased abruptly and the aluminium oxide film was broken

down.

Spark counts at the plateau increased as the aluminium oxide film decreased. However, minute scars on surface of the very thin film were liable to cause breakdown of the film, which increased blank spark counts.

The relation between spark counts and circuit constants in Fig.2 were investigated. In spite of large change of capacitance C($20-10^4$pF) number of spark counts changed slightly. This suggests that only the self capacity of aluminium oxide film determines sparking characteristics and is almost independent to C.

Fig. 3 shows curves of the relation between applied voltages and spark counts at several kinds of electric resistances in the circuit (2 kΩ-3 MΩ). As electric resistances in the circuit increased, the plateaus in the curves shifted to high voltage side. Plateaus of spark counts at 2 kΩ and 200 kΩ were a little higher than those at 1 MΩ and 3 MΩ. This difference of spark counts might be spurious counts. This was suggested by replicas of evaporated holes on the aluminized polycarbonate sheet electrode produced by sparks as shown in Fig.4. So electric resistance (R) 1 MΩ was selected for routine spark counting.

6. THE RELATION BETWEEN SPARK COUNTS AND α-PARTICLE IRRADIATION

The relation between spark counts and number of α-particles which entered the aluminium oxide film 1 μm thick was linear in the range of 10^5-10^7 α-particles/7.6 cm^2 as shown in the Fig.5. The sensitivity was approximately 10^{-3}. Minimum detectable limit was about 10^3. This corresponds to 24 hour irradiation using 0.5 μg of uranium.

Thermal and fast neutron detectors of aluminium oxide film are being examined using B(n,α) reaction and recoil protons, respectively.

In case of thermal neutron detection with uranium target using (n,f) reaction spark counts due to α-rays from uranium target should be subtracted.

7. REFFERENCES

[1] S. Prêtre; IAEA-SM-167/14 (1972) 99.
[2] J. Jasiak and E. Piesch; Nucl. Instr. and Meth. 128 (1975) 447.
[3] K. Becker; Health Phys. 16 (1969) 113.
[4] W.G. Cross and H. Ing; Health Phys. 28 (1975) 511.
[5] Y. Nishiwaki, H. Kawai, H. Morishima, T. Koga and Y. Okada; IAEA-SM-167/22 (1973) 117.
[6] Y. Nishiwaki, H. Kawai, T. Koga, H. Morishima and T. Niwa; Nucl. Instr. and Meth. 175 (1980) 248.

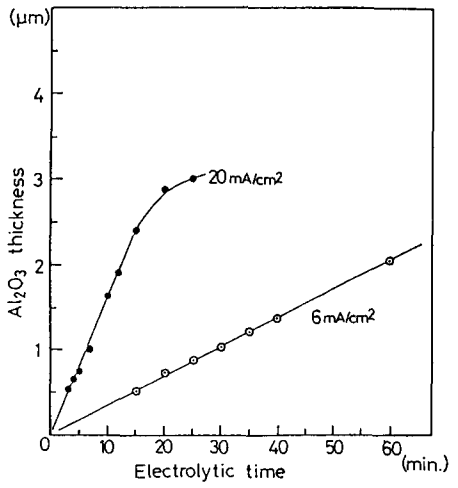


Fig.1
Relation between Al_2O_3 thickness and electrolytic time.

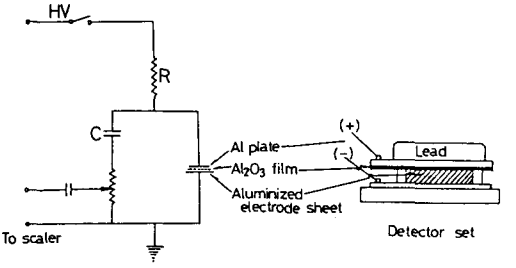


Fig.2 Circuit and arrangement of a spark counter.

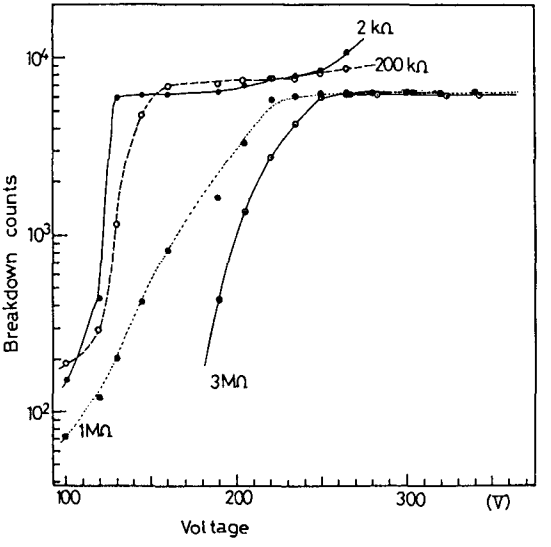
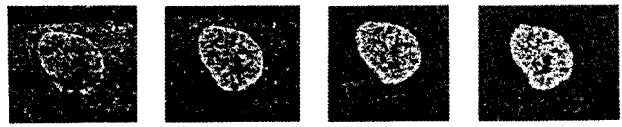


Fig.3 The relation between applied voltages and breakdown counts at different kinds of electric resistances in the circuit.



Resistance:

2 kΩ 200 kΩ 1 MΩ 3 MΩ

Fig. 4 Replicas of evaporated holes on the aluminized polycarbonate sheet electrodes at breakdown counting.

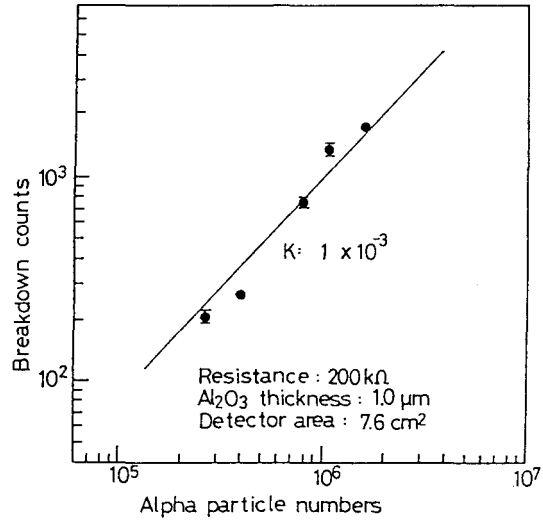


Fig. 5
Relation between alpha particle numbers and breakdown counts.

NEUTRON DOSIMETRY WITH BORON-DOPED CR-39 PLASTIC PLATE

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1. INTRODUCTION

Application of solid state nuclear track detectors for neutron dosimetry has attracted our attention because they have desirable characteristics such as non-fading tracks and non-sensitivity to visible-, UV-, X-, β- and γ-rays. Espesially, the thermosetting plastic of allyl diglycol carbonate, called CR-39, is remarkable as the detector material due to its high sensitivity to fast neutrons (GR81). This is because the plastic has a low threshold for track registration, which depends on energy transfer from heavy charged particles to the material.

Another advantage of the plastic is that it has an optical quality and allows the tracks to grow into very clear etch-pits on a smooth surface. This suggests that the detector, using the plastic, is applicable to an optical automatic track counting system. This induced the authors to study the application of the plastic to thermal and epithermal neutron dosimetry using ^{10}B as a converter. In the present study, aimed at the development of a highly sensitive, simple and safe neutron dosimeter, CR-39 plastic plates are doped with a boron compound (ortho-carborane) and their characteristics are examined.

2. EXPERIMENTAL METHODS AND MATERIALS

Ortho-carborane powder was mixed with allyl diglycol carbonate and diisopropyl peroxy dicarbonate, as polymerizing catalyzer, in a proportion of 0.5 : 96.5 : 3.0. The mixture, in a set of dies, was cured by heating and formed a transparent plate with a thickness of 1.6 millimeters. The plate contained natural boron at a concentration of 0.375% by weight.

The boron-doped plates were wrapped in thin polyethylene sheets to protect their surfaces and were irradiated with neutrons in the heavy water thermal neutron facility of the swimming pool type research reactor (KUR) at the Research Reactor Institute, Kyoto Unversity or in the center stringer of the Argonaut type research reactor (UTR-KINKI) at the Atomic Energy Research Institute, Kinki University. In order to check the influence of epithermal and fast neutrons, part of the plates were covered with cadmium cases during irradiation. The thermal neutron fluence rates were measured with Au-foils which were irradiated in the reactors. In order to check the epithermal neutrons, part of the foils were covered with Cd-cases during irradiation.

Irradiated plates were suspended in an aqueous solution of 30% KOH for etching. The etching solution was kept at 60±0.5°C and agitated by magmixer-driven vanes. After etching, the plates were immedately washed clean in flowing water, and subsequently dried in clean ventilation.

The etch-pits on the plates were counted using an optical microscope or an automatic track-counting system, which includes optical microscope, TV-camera and computer. The number of etch-pits on each Cd-covered plate was subtracted from the corresponding number observed on the bare plate so as to eliminate the influence of epithermal and fast neutrons.

3. EXPERIMENTAL RESULTS

Figure 1(a-d) reveals the transition in number and shape of etch-pits with etching time. Almost all the etch-pits originate in $^{10}B(n,\alpha)^{7}Li$ reactions. From the observation of these figures it was confirmed that the number and size of etch-pits increased with etching time during the early period of ethcing. A quantitative relation between the etching time and the number of etch-pits per unit area (density of etch-pits) is shown in Fig.2. When the etching condition and time are kept constant, the density of etch-pits P (cm^{-2}) is proportional to the irradiated thermal neutron fluence Φ (cm^{-2}). Generally P can be expressed as

$$P = \int_0^\infty K(E)\Phi_E dE \ , \qquad (1)$$

where Φ_E ($cm^{-2}\cdot eV^{-1}$) is the differential distribution of the neutron fluence with respect to neutron energy E (eV), and the proportional constant $K(E)$ is termed the "complex sensitivity" or simply "sensitivity" which has no dimension. From Fig.2 the sensitivities for thermal neutron are $(2.3\pm0.2)\times10^{-4}$ and $(4.2\pm0.2)\times10^{-4}$ for 8 and 16 hrs in etching time, respectively. The range of etch-pit density at which counting can be done with reasonable accuracy is estimated to be about 10^{3}-$10^{6}cm^{-2}$.

On the other hand, sensitivity $K(E)$ is derived theoretically to be expressed as

$$K(E) = \tau\{\frac{1}{2}R\cos^2\theta_C + (1-\sin\theta_C)l\}\sigma(E)$$

$$= k\sigma(E) \ , \qquad (2)$$

where τ is the ^{10}B-concentration in the plate (atoms·cm^{-3}), R is the mean effective range of α- and ^{7}Li-particles in the plate (cm), θ_C is the critical angle for etch-pit formation, l is the bulk etching (cm), $\sigma(E)$ is the microscopic cross section of $^{10}B(n,\alpha)^{7}Li$ reaction (cm^2), and k is termed the "prime sensitivity" which has a dimension of cm^{-2} (TS78). Assuming that θ_C is 0°, the sensitivities are calculated to be 2.2×10^{-4} and 4.0×10^{-4} for 8 and 16 hrs in etching time, respectively, for the condition shown in Fig.2. The experimental values agree well with the theoretical values. As shown in equation (2), sensitivity $K(E)$ is in proportion to the microscopic cross section of $^{10}B(n,\alpha)^{7}Li$ reaction $\sigma(E)$, which is inversely proportional to the square root of the neutron energy in the neutron energy region from about 0.001eV to 0.1MeV.

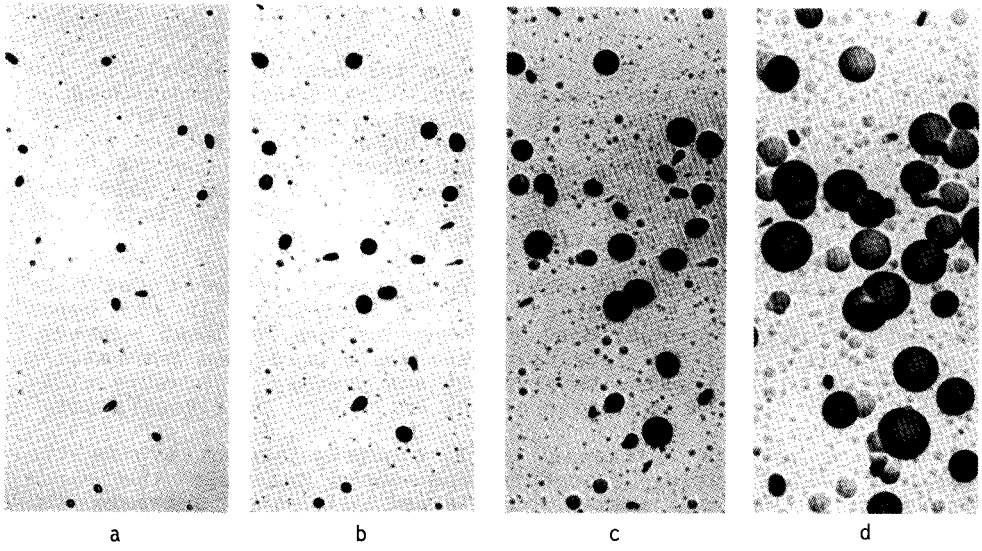

Fig.1 Growth of etch-pits on the boron-doped CR-39 plate. Etching conditions : 30% KOH, 60°C. Etching time : a 2hrs, b 4hrs, c 8hrs, d 16hrs.

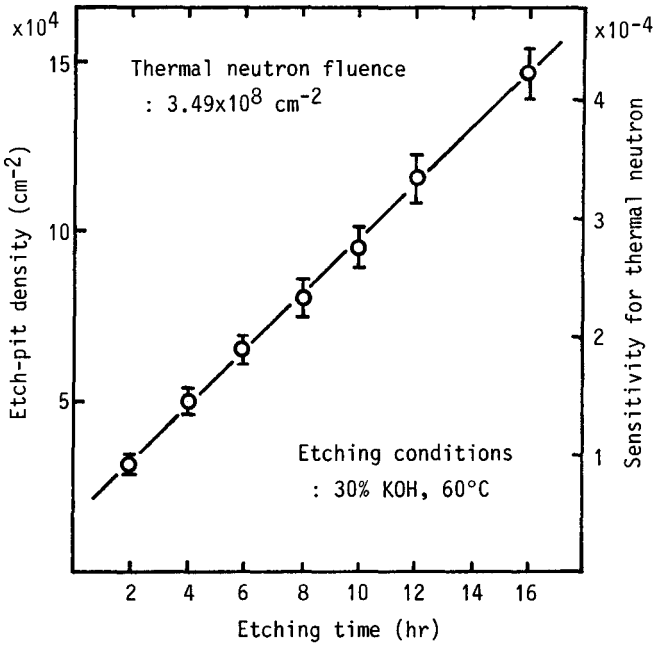

Fig.2 Increase of etch-pit density and sensitivity for thermal neutron with etching time.

4. DISCUSSION

The maximum sensitivity obtained in the present experiment was 4.2×10^{-4}, using 0.375% boron-doped plate. The value approximated to the sensitivity of a boron-doped cellulose nitrate film (TS82). If the lower and upper limits of the density of etch-pits are set at 10^3 and $10^6 cm^{-2}$, the corresponding significant ranges for the thermal neutron dose equivalent would be about 2.5×10^{-5} and $2.5 \times 10^{-2} Sv$, respectively. This detection limit is good enough for individual monitoring intended for radiation workers.

The maximum sensitivity obtained in this experiment could be enhanced five times by raising the ^{10}B enrichment in the boron compound to 100%. Occupational exposure is low for most workers. Accurate dosimetry of low level exposure would give them information for improving their working environment and process. It would ultimately do much for the reduction of the collective dose equivalent.

The use of the automatic track counting system avoids the tedious visual counting procedure and makes possible the rapid processing of a large number of plates. It also improves the objectivity of the counting.

Bare boron-doped, Cd-covered boron-doped and boron-free plates could be used in cases where thermal, epithermal and fast neutrons are significant and the energy spectrum is unknown. The coexistence of γ-rays etc., does not disturb the estimation of the neutron dose. There is no possibility of underestimating the dose equivalent due to fading. Etched and unetched plates can be preserved as material evidence for the necessary period of time. Another advantage of the plate is that it is easy to handle because the detector and converter are incorporated. Furthermore, the isotope ^{10}B is non-radioactive and thus perfectly safe.

All the foregoing merits of the boron-doped CR-39 plate should justify its wide adoption as a suitable individual neutron dosimeter.

ACKNOWLEDGMENT

The authors wish to express their indebtedness to Mr.O.Murata, of Sola Optical Japan Ltd., for preparration of the boron-doped CR-39 plates.

REFERENCES

GR81 Griffith,R.V., Thorngate,J.H., Davidson,K.J., Rueppel,D.W., Fisher,J.C., Tommasino,L. and Zapparolli,G. : Rad. Prot. Dosimetry 1, 61 (1981).

TS78 Tsuruta,T. and Sakamoto,M. : J. Nuel. Sci. Technol. 15, 602 (1978).

TS82 Tsuruta,T. and Takagaki,M. : Health Physics 43, 705 (1982).

EVALUATION OF RADON CONCENTRATION AND F-VALUE IN NATURAL ENVIRONMENT WITH TRACK ETCH DETECTORS

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In order to estimate precisely the collective lung dose due to 222-Rn and its daughters in natural environment, we must evaluate not only mean radon concentrations but also mean f-values (fraction of unattached RaA atoms) at many locations.

In the present work, we developed a device for measuring natural level radon concentration by electrically collecting radon daughters on a cellulose nitrate (CN) film enclosed in a vessel. Characteristics of the device, such as the dependency of collection efficiency of radon daughters on the humidity, track etch efficiency of CN film for alpha -particles, etc., were studied carefully.

It was verified both from experiment and theory that the amounts of deposited RaA atom upon a CN film exposed to natural air have a linear relationship with the product of f-value and radon concentration in the air. Based on the results, we also developed a technique for estimating mean f-value from the amounts of deposited RaA atom upon the CN film put on the outer wall of the vessel described above.

The present methods utilized conveniently and economically, and allows measurements of mean radon concentration of 100 pCi/m^3 and mean f-values after exposure time of more than 1 - 2 months.

RADON AND RADON DAUGHTERS INDOORS, PROBLEMS AT THE DETERMINATION OF THE ANNUAL AVERAGE

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Regulatory limits are often based on the annual average radon daughter concentration. For evident reasons this annual average can not be actually measured, in practice it is necessary to rely on shorter-term measurements of radon and/or radon daughters which are used as a basis for an estimate of the annual average. It is thus important to know the relation between different kinds of measurements used in practice and the annual average.

In order to get a quantitative basis for assessing different types of measurements in dwellings, a research program is under way at this institute.

Simultaneous measurements of radon, radon daughters and air change rates have been made and these experimental F-factors as a function of the air change rates have been compared with theoretical values. Longterm registrations of the radon concentration have been made and simultanious longterm registrations of radon and radon daughters during a whole year are planned for some dwellings. At the same time other methods will be used. The results are treated to simulate various time periods compared with the annual average.

DESIGN AND CALIBRATION OF A MODIFIED CONIFUGE

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The analysis of the internal contamination problems requires the knowledge of the composition of aerosols, both concerning their size distribution within a given respirable rate and their relative concentration.

Taking into account the operational conditions at work places, where the presence of atmospheric contaminants may be expected, the sampling instrument must be have its own operating advantages towards a systematic campaign in order to avoid the burden of complicated dissassembling techniques or the internal contamination of the device. For this purpose a modified version of a centrifugal aerosol precipitator was adapted for its use as a size spectrometer for non-gaseous contaminants in low concentrations.

The design of the instrument and its operating principles are discussed, as well as details of the modifications introduced.

One of the modifications is concerned with the utilization of the grills of an electronic microscope for sample collection, located in external grill holders for calibration purposes.

The second modification is based on the utilization of object holders for sample collection, located on special pressure-mounted devices, with external access for obtaining the mass concentration of the contaminant without disassembling the instrument and facilitating its operation during the sampling campaign.

The calibration procedures and the deposition pattern are analyzed and a graphic representation is given of the relation between the deposition point and the equivalent aerodynamic diameter for one rotational speed.

The use of the instrument for assessing the distribution of air-borne radioactivity of the mass distribution as a function of the aerodynamic diameter is discussed.

DEVELOPMENT OF MONITORING TECHNIQUE BASED ON SPECTRUM ANALYSIS OF ENVIRONMENTAL GAMMA-RAYS IN THE VICINITY OF A NUCLEAR FACILITY

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1. INTRODUCTION

The rapid increase in the utilization of a nuclear reactor for peaceful purposes requires continuous efforts to protect populations around a nuclear facility from radiation exposure. The requirement calls for intensified efforts to improve methods for radiation monitoring even under a normal operation condition of a reactor.

From this point of view, an analytical monitoring technique had developed for identification of artificial radionuclides and assessment of their radiation exposures in the natural environment(1). Since this technique was based on an analysis of the energy spectrum of the natural environmental gamma-rays, it was quite possible to estimate effective dose equivalents, recommended by the ICRP published in 1977, due to artificial radionuclides.

In the present study, a simple estimation method for exposure due to gaseous effluents from the stack has derived from the analysis monitoring data, and an effective dose equivalent for Japanese adults is determined on the basis of the energy spectrum of exposure caused by gamma-rays from ^{41}Ar radionuclide in the vicinity of the Kyoto University Reactor(KUR).

2. EXPERIMENTAL

The monitoring system for analysis of environmental gamma-rays was composed of a NaI(Tl) scintillation spectrometer, a pulse-height analyzer and a microcomputer system for processing accumulated data. The detector which was held at about 1.2 m from the ground level was concealed with a protection case of 5 cm thick plastic foam to protect given functions against severe atmospheric conditions. The monitored place was located at about 180 m to the north of the stack of 35 m in height. The continuous gamma-ray monitoring was performed in Feburary of 1982.

The pulse-height distributions which were recorded every 1 hour in a magnetic tape were unfolded by the iterative technique using a response matrix, and the true gamma-ray energy spectra were analyzed into their main components, i.e. photons from ^{40}K, ^{238}U-series, ^{232}Th-series and others, by the microcomputer system. Gamma-ray energy spectra from the gaseous effluents were determined by subtracting the natural environmental gamma-ray energy spectra from the measured ones. The terrestrial gamma-ray energy spectra were calculated using the energy and directional distributions of photon flux densities from the isotropic sources of U, Th and K in the soil(2) and gamma-ray fields due to ^{222}Rn daughters on the ground-air interface were determined on the basis of the results in reference 3. The exposure rates were calculated from the well-known formula(4) and the effective dose equivalent rates were determined from the following equation,

$$H_t = \sum_i C(E_i) \cdot f(E_i) \qquad (1)$$

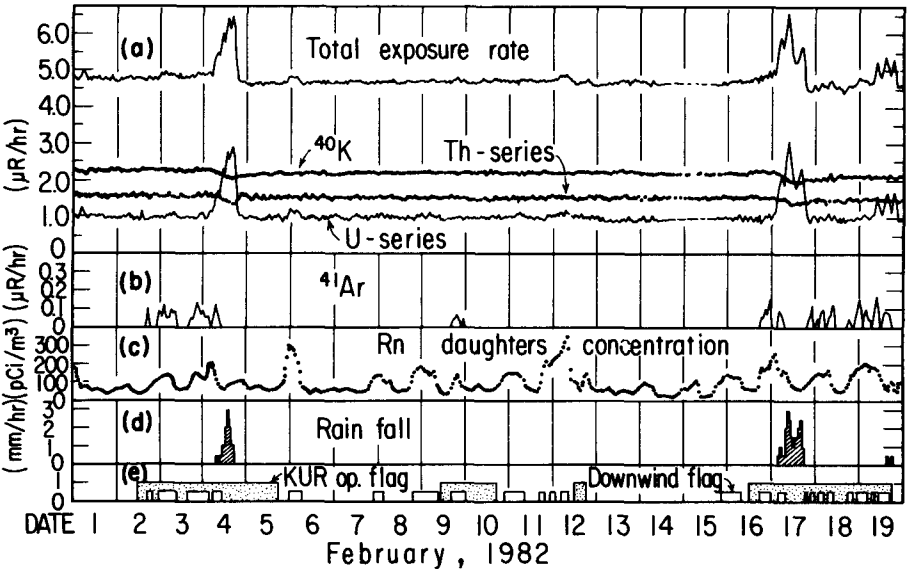

Fig.1 Time histories of total exposure rate and its components, ^{222}Rn daughters concentration, rainfall incidence, KUR operating and wind direction flags.

where C(Ei) is an exposure(0.87 Gy in air) to effective dose equivalent(Sv) conversion factor for an adult Japanese given as a function of photon energy(5) and f(Ei) is an energy spectrum of exposure rate calculated from the photon flux density.

3. RESULTS and DISCUSSION

Sample analysis data for 19 days are illustrated in Fig.1 with ^{222}Rn daughters concentration(c), rainfall incidence(d), wind direction and reactor operation flags(e). The flag of 0.5 means the monitoring place was at downwind from the stack, and that of 1.0 means the KUR was operating at the power of 5 MW. Downwind width adopted in this work were ± 3 points on a 16-points compass. Mean speeds of southerly and northerly winds during reactor operation were 2.2 m/sec and 3.3 m/sec, respectively.

The time variation of the total exposure rate and its components reveals us that the short term variation of the exposure rate can be mainly attributed to the precipitation washout and the variation of the ^{222}Rn daughters concentration in the atmosphere. The exposure rate of about 2.0 μR/hr is increased by the rainfall rate of about 3 mm/hr in Fig.1(d) and the increase of about 0.25 μR/hr is observed owing to the airborne radioactivities of 250∿300 pCi/m^3 in Fig.1(c). On the other hand, the long term variation depending on the moisture content in the soil after rainfall incidence in Fig.1(d) is clearly observed in the time variation of ^{40}K and ^{232}Th-series exposure rates.

Plume exposure rates due to unscattered gamma-rays from ^{41}Ar radionuclide released at the rate of 6 x 10^{-2} Ci/hr is shown in

Fig.1(b). The accuracy of the plume exposure rate was minutely discussed in ref.1 together with the arguments of minimum detectable amount of exposure rate. It is evident from the figure that they were observed when the monitoring place was at the downwind from the stack, furthermore, they were accompanied with the increase of ^{222}Rn daughters concentration. It seems possible from this fact that the movement of ^{41}Ar in the air was mainly controlled by the same meteorological factors as those regulating airborne radioactivities in the natural environment. A relation between a plume exposure rate(E_p) and an inverse of wind velocity(I_v) is shown in Fig.2 Vertical bars represent errors included in the analysis method. Though there are a few data around higher 1/v values, a general tendency of them seems to be shown by a linear relation of $Ep(\mu R/hr) = 0.14\ I_v$ (for release rate of 6×10^{-2}Ci/hr). This will be available for assessment of the long term plume exposure rate around a nuclear facility, for example, the relation will be rewritten by the formula,

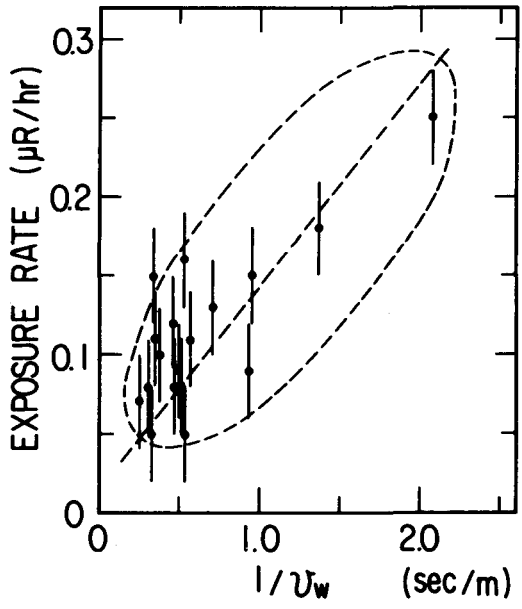

Fig. 2 Relation between a plume exposure rate and an inverse of wind velocity.

$$Ep(\mu R/year) = (2.2 \pm 1.1) \cdot Q \cdot I_v \cdot P_d \qquad (2)$$

when the total amount of gaseous ^{41}Ar(Q Ci/year) and the inverse of the mean wind velocity(I_v) and the downwind probability(P_d) for a year are given. The exposure rates due to scattered photons(Es) were determined by using the build-up factors(BFs) calculated for the finite line source of ^{41}Ar at an effective stack height of 45 m from the ground level.

Effective dose equivalent(H_t) due to ^{41}Ar were determined by the following equation,

$$H_t = C(1.29) \cdot E_p + C(0.25) \cdot (BF - 1.0) \cdot E_p \qquad (3)$$

where C(1.29) and C(0.25) are exposure to effective dose equivalent (EEDE) conversion factors of Japanese adults for photon energies of 1.29 and 0.25 MeV, respectively(5). Some examples of exposure rates and effective dose equivalent rates are shown in table 1. EEDE conversion factors for isotropic irradiation given in ref.5 were chosen because the exposure caused by the gaseous effluents was considered to be almost isotropical. The last column in table 1 also denotes the deep dose equivalent rate index(H_d) that was recommended by the ICRU published in 1976. Calculations of indices were carried out in the same manner as was adopted in case of the effective dose equivalent rate. The exposure to dose equivalent index conversion

Table 1. Exposure rates, effective dose equivalent rates and dose equivalent rate indices due to plume ^{41}Ar from the KUR stack.

Date	ER (μR/day)	EDER (10^{-2}μSv/day) Male	EDER (10^{-2}μSv/day) Female	DERI (10^{-2}μSv/day) Hd
Feb. 3	1.66	1.43 (0.86)*	1.34 (0.81)*	1.73 (1.05)*
4	0.82	0.70 (0.85)	0.67 (0.82)	0.85 (1.05)
9	0.39	0.34 (0.87)	0.32 (0.81)	0.41 (1.05)
18	1.00	0.85 (0.85)	0.81 (0.81)	1.03 (1.03)

* : Values in the parentheses are ratios to the exposure rates.
ER : Exposure rate.
EDER : Effective dose equivalent rate.
DERI : Dose equivalent rate index (Hd is the deep dose equivalent rate index recommended by the ICRU).

factors which were given in ref.6 were used in the calculation in stead of EEDE conversion factors $C(E_i)$ in the formula (3).

It is evident that the effective dose equivalent rate was the smallest in the three types of quantity information for radiation protection and the ratio of effective dose equivalent or dose equivalent index to exposure was almost constant in spite of the different BF for the line source of ^{41}Ar. The mean values of these ratios are quite similar to those obtained in the natural environment (5). This fact suggests that the unique conversion factor from exposure to effective dose equivalent will be employable for estimation of radiation exposure due to background gamma-rays and photons from ^{41}Ar released artificially.

4. CONCLUSIONS

From the investigations described in the present report, conclusions are as follows,

1. The short term variation in the time series of exposure rate could be separated successfully from the long term ones using the analytical gamma-ray monitoring technique.

2. General tendency of the relation between the plume ^{41}Ar exposure rate and an inverse of wind velocity was confirmed as was applicable to estimation of the long term plume exposure around the KUR site.

3. The effective dose equivalent rate was determined on the basis of the energy spectrum of the plume exposure rate, and it was concluded that the same conversion factor, from exposure to effective dose equivalent, as that given by the natural environmental gamma-rays could be available for estimation of effective dose equivalent due to artificial radionuclide of ^{41}Ar.

References

(1) I.URABE et al., to be published.
(2) S.MINATO, J. Nucl. Sci. Technol., 8 342 (1971).
(3) S.MINATO, J. Japan Health Phys. Soc., 15 19 (1980).
(4) S.MINATO and M.KAWANO, J. Geophys. Res., 75 5825 (1970).
(5) I.URABE et al., J. Japan Health Phys. Soc., 18 (1983).
(6) K.HOHLFELD and B.GROSSWENDT, Radiat. Prot. Dosm., 1 277 (1981).

INVESTIGATION OF THE OVERALL MEASUREMENT UNCERTAINTY OF DOSE (EQUIVALENT) RATE METERS USED IN RADIATION PROTECTION

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Radiation protection monitoring instruments are calibrated under specified conditions with all influence quantities (e.g. environmental conditions, radiation field geometry and radiation energy) fixed at or corrected for their reference values. In national and international standards the relative intrinsic error of instruments monitoring external exposure is required not to exceed given limits. Influence quantities within their rated range of use shall not change the indication of the instruments beyond specified amounts (1, 2). On the other hand international bodies of radiation protection as e.g. ICRP (3) and IAEA (4) postulate that the uncertainty of practical measurements should be within a factor of 2 (95 % confidence level) of the true value or within ± 50 %, respectively. Up to now, there have been not any comprehensive investigation on wether and to what extend these different and independently developed requirements comply with each other.

Considerable information on the behavior of radiation protection instrument was acquired during type tests carried out for the approval of radiation protection dosimeters for verification by the German "Eichbehörden". The investigation of the properties of monitoring instruments intended to be used in measuring new operational quantities contributed additional knowledge (5).

Hitherto there have not been any recognized method for determining how the effects of different uncorrelated influence quantities are superimposed. At reference conditions the error of an instrument is called "intrinsic error", the response r_o at a reference value of an influence quantity is related with the instument's reading M and the conventionally true value of the measuring quantity H (e.g. dose equivalent) by $r_o = (M/H)_o$. The response when only one influence quantity assumes any value within a given range while all the other influence quantities remain at their reference values is called r_i. The effect of the influence quantity i on the change in relative response can be expressed by

$$f_i = \frac{r_i - r_o}{r_o} \cdot 100 \text{ in \%}$$

In fig. 1 an example of the composition of errors due to various influence quantities is given. At the left hand the algebraical addition of the $(f_i)_{max}$ is shown, where $(f_i)_{max}$ is the stated maximum allowed limit of variation due to one influence quantity. The error in operating an instrument must remain within the total spread of the left hand part of fig. 1 for any combination of influence quantity values allowed by the standard. In the case of dose meters and dose rate meters the addition of the maximum deviations (expressed as the f_i) due to all influences on the instruments response as proposed in ref. (6) for an other kind of instrumentation would lead to unrealistically large uncertainties.

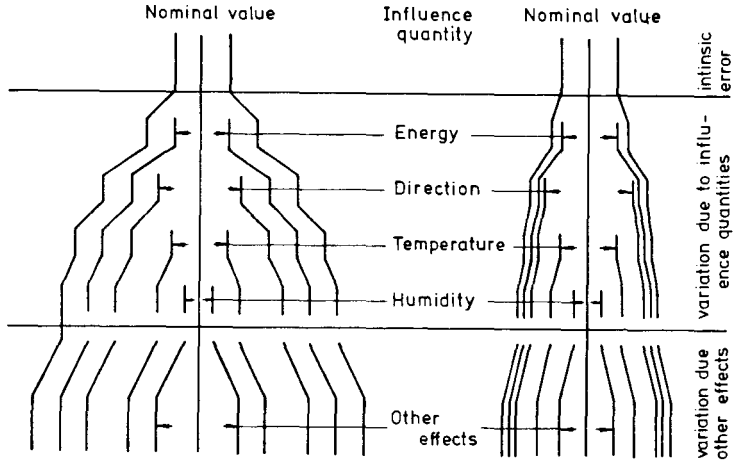

Figure 1. Functioning error. Example of the composition of errors due to the effects of influence quantities.

Influence-quanty / Detector		Energy	Direction	Dose rate	Temperature	Pressure	Humidity	EMI
Requirements		±30	±20	±20	±20	± 5	±20	
Ionisation-chamber	1	-27	+17	-	10	0	0	-
	2	-25	- 7	-14	-10	0	- 5	-
	3	-24	- 3	-	+16	0	- 1	-
	4	+24	18	-19	1	- 1	+ 1	-
Geiger-Müller counter	1	-30	4	- 3	- 3	+ 3	+ 8	-
	2	+35	27	-	+ 5	0	+15	-
	3	+30	-21	-	-11	0	6	-
	4	+30	-12	-	5	- 1	+ 4	-
Scintillator	1	+17	-10	-12	+13	0	-16	-
	2	-33	+ 2	- 7	- 7	1	6	+ 5
	3	-30	- 3	-	-14	0	-12	< 2

Table 1. Measured maximum variation of relative response f_i when one influence quantity was changed within its rated range. The dosemeters have been selected to be typical ones for the kind of dosemeters used (influence quantity EMI = electromagnetic inferences). The requirements relate to ref. (2).

Table 2. Combined error of relative response due to the variation of influence quantities within their valid range of use.

Detector \ Error in %		Intrinsic Error	$\sqrt{\Sigma f_i^2}$	F_1	F_2
Ionisation-chamber	1	+ 20	33	159	52
	2	+ 18	32	143	55
	3	- 10	29	145	65
	4	+ 20	36	149	65
Geiger-Müller counter	1	- 15	32	168	65
	2	- 15	50	189	65
	3	+ 16	39	190	59
	4	- 14	33	156	79
Scintillator	1	- 5	31	154	58
	2	- 7	35	141	54
	3	- 8	35	124	46

In table 1 measured maximum values of f_i for each influence quantity are given. Typical instruments were selected with three different kinds of detectors. Using the detailed information on how the f_i vary within the rated range of the influence quantities the values of F_1 and F_2 shown in table 2 have been calculated. F_1 and F_2 are defined as follows

$$F_1 = \text{Max} \prod_{i=1}^{n} (1 + f_i/100) \cdot 100$$

$$F_2 = \text{Min} \prod_{i=1}^{n} (1 + f_i/100) \cdot 100 \; .$$

Values of F_1 and F_2 given in table 2 refer to that combination of influence quantity values which result in the most extreme deviation of the indication from that at reference conditions.

As this is a very inprobable case, the variation of response is combined in such a way as to take the statistical nature of occurence of the different conditions into account. The problem how to sum up uncertainties has been discussed since long time (7).

Employing the usual summation of the squares of the uncertainties

$$F = \sqrt{\sum_{i=1}^{n} f_i^{\,2}}$$

together with largest measured values of f_i the values of row 3 of table 2 are found. With this procedure under the assumptions given

in ref. (8) the confidence level can be quoted as 92 %. The corresponding uncertainty for the confidence level 95 % is obtained by multiplication with a factor of 1.13 which can be neglected for the conclusions to be drawn from this investigation.

Table 2 demonstrates that the overestimation F_1 never exceeds a factor of 2 and that the underestimation F_2 is confined to a factor of 0.5 (confidence level 100 % because of the definition of F_1 and F_2). At a confidence level of 95 % all but one instrument type tested had a total uncertainty within a factor of 1.5. It can be concluded that radiation protection instruments fulfilling each requirement stated in various standards are well within the overall limit of uncertainties acceptable in routine individual monitoring and monitoring of the workplace for external radiation. The uncertainty discussed does not include uncertainties in deriving tissue or organ dose equivalents from the measurement results.

Special attention should be devoted to the fact that the angular dependence of response will normally be subjected to requirements which are not sufficient to guarantee the indicated value showing an acceptable deviation from the true value (8).

References

(1) IEC 45B (CO)43, Februar 1982: Beta, X and Gamma Radiation Dose Equivalent and Dose Rate Meters for Use in Radiation Protection.

(2) DIN 6818 - Strahlenschutzdosimeter Teil 2 bis 5, 1980. Beuth Verlag, Berlin und Köln.

(3) ICRP Publication 36 (1982). General Principles of Monitoring for Radiation Protection of Workers. Pergamon Press, Oxford, New York, Toronto, Sydney, Paris, Frankfurt.

(4) IAEA Safety Series Nr. 14 (1980): Basic Requirements for Personnal Monitoring. IAEA, Vienna.

(5) Selbach, H. J., Hohlfeld, K., Kramer, H. M. (1983): Radiation Characteristics of Depth Dose Equivalent Meters. This Conference, session 14.3.

(6) IEC 45 (CO)170, March 1983: Complementary Instrumentation for Counting Ratemeters. Characteristics and Test Methods.

(7) Wagner, S. (1969): Zur Behandlung systematischer Fehler bei der Angabe von Meßunsicherheiten. PTB-Mitteilungen 79, S. 343.

(8) Reich, H. (1978): Evaluation of Measuring Results, Statement of Uncertainty in Dosimeter Calibration. PTB-Dos-3; Physikalisch-Technische Bundesanstalt Braunschweig.

(9) Grosswendt, B., Hohlfeld, K. (1982): Angular Dependence of Specified Depth Dose Equivalent Quantities in the ICRU Sphere for Photon Radiation. RPD, 3, p. 169-174.

MESUREUR DE L'EQUIVALENT DE DOSE ET DU FACTEUR DE QUALITE EN CHAMP MIXTE NEUTRON ET GAMMA (C.I.R.C.E.)

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INTRODUCTION

La plupart des dosimètres couramment utilisés en radioprotection convertissent la réponse du détecteur en équivalent de dose par un simple étalonnage dans un spectre connu qui sera supposé analogue aux spectres des champs de rayonnement à mesurer.

Il est évident que cette méthode avantageuse ne puisse convenir que pour un nombre limité de situations. L'I.C.R.U. et l'I.C.R.P. [1] recommandent des valeurs du facteur de qualité Q(L) déterminées en fonction du transfert linéique d'énergie L, ceci afin de proposer une définition aussi simple que possible de ce facteur. Dans ce cas, la détermination de l'équivalent de dose H par la relation :

$$H = \int_0^{\infty} D(L).Q(L)dL$$

nécessite la connaissance préalable de la distribution de la dose absorbée D(L) en fonction de L.

Le but de cette communication est de présenter une unité compacte d'acquisition et de traitement d'information, pouvant donner en une seule mesure la dose absorbée, l'équivalent de dose et le facteur de qualité d'un champ mixte de radiations photonique et neutronique.

PRINCIPE DE LA METHODE

Le principe [2] utilisé dans la détermination de l'équivalent de dose et du facteur de qualité est basé sur deux idées essentielles portant d'une part sur la substitution du transfert linéique d'énergie L par l'énergie linéique y, $L = \frac{8}{9} y$, et d'autre part sur le traitement impulsion par impulsion au fur et à mesure de la production d'événements de dépôts d'énergie au niveau du capteur.

Cette substitution permet d'éviter les méthodes complexes de déconvolution traditionnellement utilisées pour obtenir une distribution des transferts linéiques d'énergie à partir d'une distribution des impulsions issues d'un compteur proportionnel.

A chaque événement i de dépôt d'énergie ε_i communiquée au milieu, correspond une hauteur h_i d'impulsion créée par un compteur proportionnel sphérique équivalent au tissu.

Dans le cas d'une amplification linéaire on a $\varepsilon_i = C h_i$ (1)
où C est le facteur d'étalonnage.

L'énergie linéique y_i est égale à $\varepsilon_i/(2d/3)$, en utilisant la relation (1) on obtient : $y_i = C h_i / (2d/3)$ (2)
où d est le diamètre de la taille simulée par le compteur. Le transfert linéique d'énergie L_i correspondant à l'événement i est égal alors à :

$$L_i = \frac{4}{3} \frac{C}{d} h_i \quad (3)$$

Par conséquent, la valeur $Q(L_i)$ du facteur de qualité est attribuée à l'événement i, selon la fonction Q(L) définie par l'I.C.R.P. et l'I.C.R.U.

La dose absorbée D est l'énergie moyenne communiquée au milieu par unité de masse. La meilleure estimation de D est égale alors au quotient de la somme de

toutes les énergies communiquées par la masse m du volume considéré $D = \frac{1}{m} \Sigma \varepsilon_i$ où l'indice i concerne tous les événements de dépôt d'énergie produits dans le compteur.

La dose absorbée D est déduite en fonction des hauteurs h_i des impulsions par les relations (1) et (4) : $D \simeq \frac{C}{m} \sum_i h_i$ (5)

Par analogie, l'équivalent de dose H est égal à : $H = \frac{C}{m} \sum_i h_i \, Q(L_i)$ (6)
Et le facteur de qualité moyen $\overline{Q}$ est déterminé par la relation

$$\overline{Q} = \frac{H}{D}$$

PERFORMANCES

Des mesures ont été effectuées pour la détermination du facteur de qualité dans des faisceaux de neutrons monoénergétiques.

La figure 1 montre la comparaison entre les valeurs de $\overline{Q}$ mesurées avec le compteur simulant une taille de cible de 2 µm et celles calculées (4) relatives aux tailles 0,5 µm et 2 µm.

Les valeurs de $\overline{Q}$ calculées et mesurées pour la taille simulée de 2 µm sont sous-estimées par rapport à celles recommandées par l'I.C.R.P. 21 [5] calculées pour la même taille simulée. Théoriquement, lorsque la pression de remplissage du compteur diminue, on obtiendra de meilleurs résultats, car, l'effet des particules arrêtées dans le volume gazeux ou de celles qui y prennent naissance devient moins important. La perturbation due à cet effet est dans tous les cas plus prépondérante que celle due à la substitution de L par $\frac{8}{9}$ y [4].

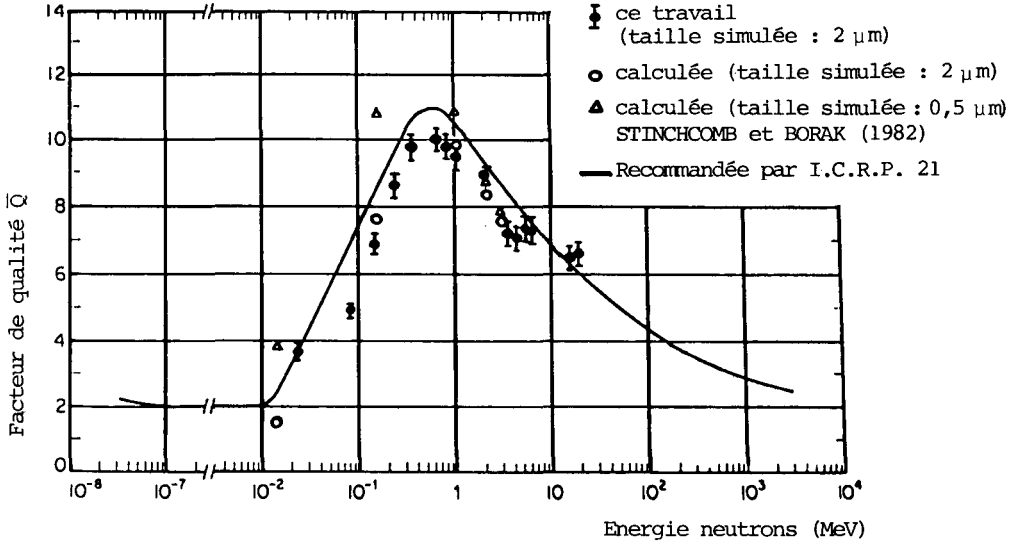

Figure 1 - Facteur de qualité effectif $\overline{Q}$ en fonction de l'énergie neutron

CONCLUSION

Avec une pression gazeuse du compteur simulant une cible de 2 µm et la substitution de L par $\frac{8}{9}$ y, les mesures semblent encore acceptables jusqu'à l'énergie des neutrons de 15 MeV, l'écart reste inférieur à 15 % par rapport aux valeurs recommandées.

Les résultats expérimentaux confirment les prévisions théoriques pour des neutrons inférieurs à 4 MeV. Le choix d'un compteur de 5 cm de diamètre semble être un bon compromis pour obtenir une sensibilité suffisante sans avoir des effets importants dus aux particules arrêtées dans le volume gazeux ou qui y prennent naissance.

Les mesures faites avec CIRCE et présentées dans ce travail montrent une possibilité de répondre au besoin de déterminer à la fois le facteur de qualité, l'équivalent de dose et la dose absorbée par une seule et unique mesure en radioprotection des champs mixtes de neutrons et de photons.

Sans aucune modification sur la structure matérielle, CIRCE peut encore fonctionner dans le cas où la fonction Q(L) serait remplacée par une autre, par exemple, q(y) proposée par ROSSI [6].

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REFERENCES

[1] I.C.R.P. and I.C.R.U.
Report of the RBE Committee to the International Commission on Radiological Units and Measurements,
Health Phys. 9, 357 (1963).

[2] NGUYEN V.D., LUCCIONI C., CHUITON R., CHAPUIS J.C., RICOURT A., PARMENTIER N.
Compteur intégrateur de rayonnement complexe,
Congrès commun Italo-Français de Radioprotection, Florence (Italie) 1983.

[3] RICOURT A., POSNY F., SOULIE R., CHEMTOB M., NGUYEN V.D.
Possibilités d'utilisation des techniques microdosimétriques pour la détermination de l'équivalent de dose.
7th Symposium on Microdosimetry, p. 625-636, Oxford (U.K.) (1980).

[4] STINCHCOMB T.G., BORAK T.B.
Neutron quality parameters versus energy below 4 MeV from microdosimetric calculations
Radiat. Res., 93, 1 (1971).

[5] I.C.R.P. 21 (1971).

[6] ROSSI H.H.
A proposal for revision of the quality factor
Rad. and Environm. Biophys., 14 (1977).

EXPERIMENTAL DETERMINATION OF COSMIC-RAY RESPONSE OF TL DOSIMETERS FOR ENVIRONMENTAL USE

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1. INTRODUCTION

Thermoluminescence dosimeters (TLD) are widely used for environmental monitoring because of their small size and ease of handling, but it is known that the results obtained from TL dosimeters are slightly decreased compared with the results obtained by other dosimetric instruments, for example an ionization chamber. The reason is considered to be the low response characteristics of TL dosimeters to cosmic-ray hard component(1,2).

In order to clarify these problems we have carried out some experiments to evaluate TLD response to cosmic rays under different cosmic-ray intensities, using a 15 liter ionization chamber and a 3" dia. spherical NaI(Tl) scintillation spectrometer in comparison with TL disometers.

2. GENERAL DESCRIPTION OF TL DOSIMETERS

The test TL dosimeters are commercial ones which package two elements in the one TL dosimeter, encapsulating TLD powder in glass. Their energy responses are improved to within 10 % in the range from 40 keV to 3 MeV, by putting a metal filter within the packaging case. The types of TL phosphors used were Mg_2SiO_4:Tb (Kyokko MSO-L) and $CaSO_4$:Tm (National UD-200S). The relevant information is shown in Table 1.

Table 1 Type of TL Dosimeter and Reader.

TL Dosimeter		TLD Reader
Type Name of TLD	Phosphor Material	
Kyokko MSO-L*	Mg_2SiO_4:Tb	Kyokko-1200
National UD-200S*	$CaSO_4$:Tm	National UD-502A

* Packaging two elements in the one TL dosimeter.

The TL dosimeters were calibrated with photons from a Ra-226 gamma ray source, and the beam of photons was irradiated perpendicularly to the TL dosimeter axis. The exposure rate value in calibration was obtained by calculation according to the inverse square law, and the scattered fraction, evaluated by a shadow shielding technique, was subtracted.

3. INSTRUMENTS FOR EXPOSURE RATE MEASUREMENT

For comparison with TLD measurements a 15 liter spherical ionization chamber

and a 3" dia. spherical NaI(Tl) scintillation spectrometer were used as instruments for the standard measurement of exposure rate. The exposure rate at the measuring point was evaluated from ionization chamber measurements, and was also, calculated directly from pulse height spectrum data by Spectrum-Dose conversion method using G(E) function(3,4). Furthermore, from pulse height spectrum data the absorbed energy in NaI(Tl) scintillator was integrated in the range from 3 MeV to 60 MeV and used as an interconnecting parameter for evaluation of TL response to muons which constitute a large part of cosmic-ray charged particles.

The contribution of alpha-ray background in the ionization chamber was 0.80 (μR/h), and that of K-40 background of the NaI(Tl) scintillation counter was less than 0.01 (μR/h). Their contributions were subtracted in a dose evaluation process.

4. RESPONSE TO HARD COMPONENT OF COSMIC RAY

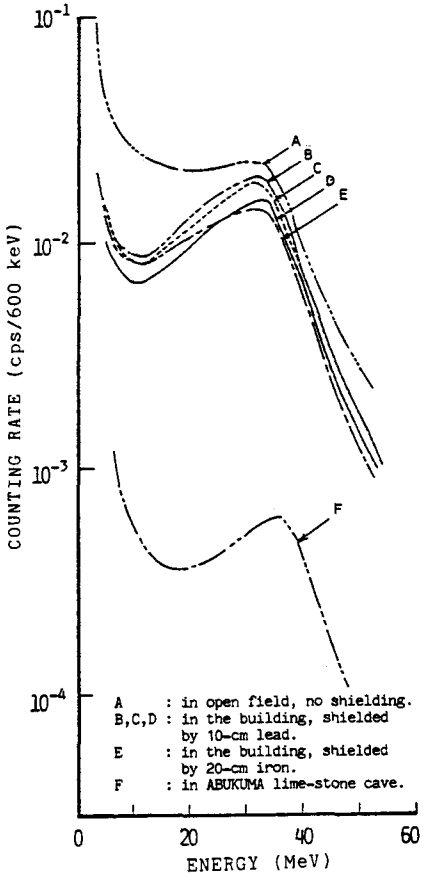

Fig.1 Absorbed Energy Spectra Measured by 3"Ø Spherical NaI(Tl) Scintillation Spectrometer at Various Points.

To evaluate self-irradiation background of TL dosimeters and the response to the cosmic-ray hard component, measurements were carried out at several sites, in the open ground, in buildings and in a lime-stone cave. Two sets of MSO-L and UD-200S, which contain the seven or more TL dosimeters, were stored in a 10-cm thick lead box in order to eliminate the contribution due to natural gamma radiations and cosmic-ray soft component.

Fig.1 shows the cosmic-ray intensity at various experimental sites, which was measured by a 3" dia. spherical NaI(Tl) scintillation spectrometer. Judging from shapes of the spectrum and the similarity as shown in Fig.1 the cosmic-ray components filtered with 10-cm thick lead consist of muons actually. The cosmic-ray intensity in a lime-stone cave was reduced to only about 4 % of the ground level value.

The TL dosimeters were exposed for 67 days. Fig.2 shows correlation curves between values of TLD measurement and total absorbed energy in 3" dia. spherical NaI(Tl) scintillator, and the regression equation was obtained by least square fitting as follows,

$$R_{TLD} = a \cdot E_{abs} + b, \qquad (1)$$

where R_{TLD} is TLD reading (μR/h), and E_{abs} is the absorbed energy (MeV/sec) in 3" dia. spherical NaI(Tl) scintillator, and the constants a, b are 0.090 ($\mu R \cdot h^{-1}/MeV \cdot sec^{-1}$) and 0.76 ($\mu$R/h) respectively for Mg_2SiO_4:Tb and 0.100 and 1.13 for $CaSO_4$:Tm. From these results it is noticeable that TLD responses

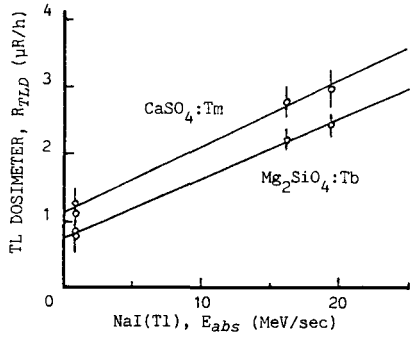

Fig.2 TLD Responses to Cosmic-Ray Hard Component under the Different Shielding Condition.

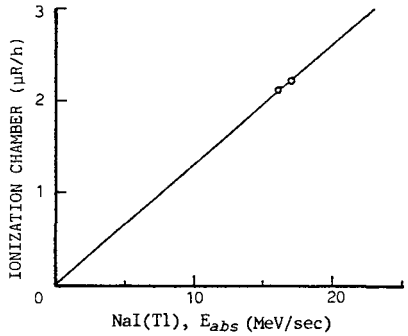

Fig.3 Relation of Responses between Ionization Chamber and 3"φ Spherical NaI(Tl) Scintillator to Cosmic-Ray Hard Component.

between MSO-L and UD-200S to the hard component of cosmic-ray charged particles differs by about 10 %. The value of b corresponds to the self-irradiation background and differs between MSO-L and UD-200S too.

On the other hand, the relation between exposure rate measured by an ionization chamber and absorbed energy in NaI(Tl) scintillator is obtained as shown in Fig.3. The value 20.6 MeV/sec of cosmic-ray hard component in tne open ground was obtained by extrapolation to zero of lead absorber thikness. Using this value and the factor 0.131 ($\mu R \cdot h^{-1}/MeV \cdot sec^{-1}$) from the ionization chamber measurement, the TL responses to the hard component in open field were calculated 1.85 ($\mu R/h$) for MSO-L and 2.07 ($\mu R/h$) for UD-200S. As we can consider the muon ionizations constitute about 79 % of the total ionizations, the TL dosimeters indicate the low response to the hard component, of 0.68 for MSO-L and 0.77 for UD-200S, compared to the response of an ionization chamber.

5. RESPONSE TO ALL COSMIC-RAY COMPONENTS

The further experiments in open field was conducted in order to obtain the response of MSO-L to the total cosmic-ray components of the hard and the soft under the unshielded conditions. In this experiment UD-200S was not examined.

The contribution from natural gamma radiations must be subtracted from total value of TLD measurements to obtain the net response to cosmic-ray components. In this experiment a NaI(Tl) scintillation DBM exposure rate meter with a flat energy response was used as an exposure monitor(4,5), and to reduce the contribution from direct gamma radiations from the ground the concentric concave lead shielding was applied to TL dosimeters and a DBM monitor respectively.

The total exposure rate of gamma radiations was evaluated from continuous monitoring data. The net response to all cosmic-ray components was obtained by subtracting this value and the self-irradiation from the total value of TLD measurements exposed during 74 days.

In MSO-L the total value from cosmic rays was evaluated 2.74 ($\mu R/h$) equivalent. This value corresponds to 81 % of that of an ionization chamber. On the other hand, using the previous results on the hard component the contribution

from cosmic-ray soft component in open field is estimated to be 0.90 (μR/h) equivalent in MSO-L. This value is a little high compared to that obtained by an ionization chamber.

6. CONCLUSION

All results were summarized in Table 2 with the results of an ionization chamber. As the results show, the TL response to cosmic-ray hard component is much reduced compared with that of an ionization chamber, and, moreover, is different between MSO-L and UD-200S, but TL response of MSO-L to the cosmic-ray soft component was nearly comparable with an ionization chamber.

Table 2 Experimental Results of Dosimeter Responses to Cosmic-Ray Soft and Hard Components.

TL DOSIMETER	COSMIC RAYS (μR/h)			RATIO
	Hard Component	Soft Component	Total	TLD/IC
MSO-L (Mg_2SiO_4:Tb)	1.85	0.90	2.74	0.81
UD-200S ($CaSO_4$:Tm)	2.07	——	(2.96)*	(0.87)*
IONIZATION CHAMBER	2.70	0.70	3.40	——

* Assuming the soft component 0.90 (μR/h).

The total response of TL dosimeters to cosmic-ray charged particles was assumed 2.74 (μR/h) for MSO-L and 2.96 for UD-200S respectively, under the cosmic-ray field of 3.40 (μR/h). The low TL responses are surely affected by low LET values of incoming charged partilces, and the difference between MSO-L and UD-200S will be caused probably by properties of phosphors and the structural and material configuration of dosimeter packaging, but the cause or mechanism of these phenomena has yet to be sufficiently clarified. This problem must be taken into account for absolute measurement of environmental radiations.

REFERENCES

1. Wayne M. Lowder and Gail de Planque, HASL-313 (1977).
2. Keran O'Brien, Intern. J. Appl. Radiation and Isotopes, Vol.29, pp735-739 (1978).
3. S. Moriuchi and I. Miyanaga, Health Phys., Vol.12, pp541-551 (1966).
4. S. Moriuchi, JAERI-M 7066 (1977) (in Japanese).
5. S. Moriuchi, K. Imai and I. Miyanaga, J. Nucl. Sci. Technol., Vol.17, pp710-720 (1980).

FIBER OPTIC SENSORS FOR ENVIRONMENTAL SURVEILLANCE

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ABSTRACT

The development of synthetic glasses of highest quality for the production of fiber optic waveguides creates new potential applications in dosimetry. The impact of radiation on glass is shortly described. The principles of the measurement equipment used to detect the radiation-induced changes of the optical properties of fibers are discussed. The sensitivity and dose range achievable with commercial optic fibers available today are given. A comparison of the specific benefits and limitations of optic fiber waveguides with those of other detectors shows their wide range of applications in adverse environments and in nuclear instrumentation. Suggestions are made for applications using the advantages of fiber detectors. Information about the costs and availability of fiberoptic dosimeters is given to facilitate the discussion between equipment developers and end users.

Effects of Radiation in Glass and Their Measurement

There are several physical processes induced by radiation in glass: radiation-induced absorption, thermo-, cathodo- and radiophotoluminescence and the change of refractive index. Up to now, only the radiation-induced absorption and thermoluminescence are used for optic fiber dosimetry and therefore we will limit the discussion to these topics. Radiation-induced absorption is caused by centers within the bandgap of the silicon dioxide (SiO_2) produced by the absorbed dose. The damage may be annealed by low energy photons or phonons interacting with these centers. This effect can possibly decrease the long-term stability of fiber dosimeters but on the other hand can be used to erase the stored information and thereby reset the dosimeter. Fig. 1 shows the radiation-induced attenuation of a lead-glass fiber as function of the measurement wavelength. The large sensitivity-difference at different wavelengths can be used to select different measurement ranges on just one fiber detector.

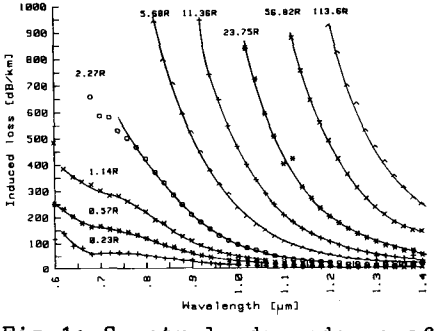

Fig.1: Spectral dependence of radiation induced loss of a high sensitive lead-glass fiber.

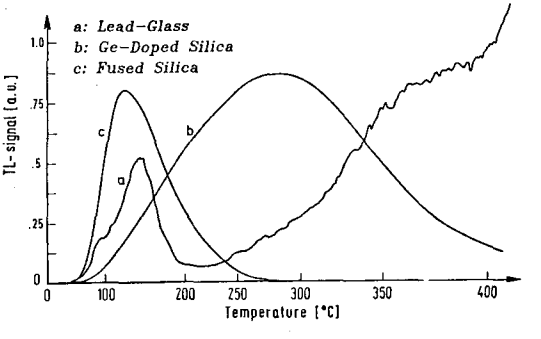

Fig.2: TL-signals of differently doped glass fibers.

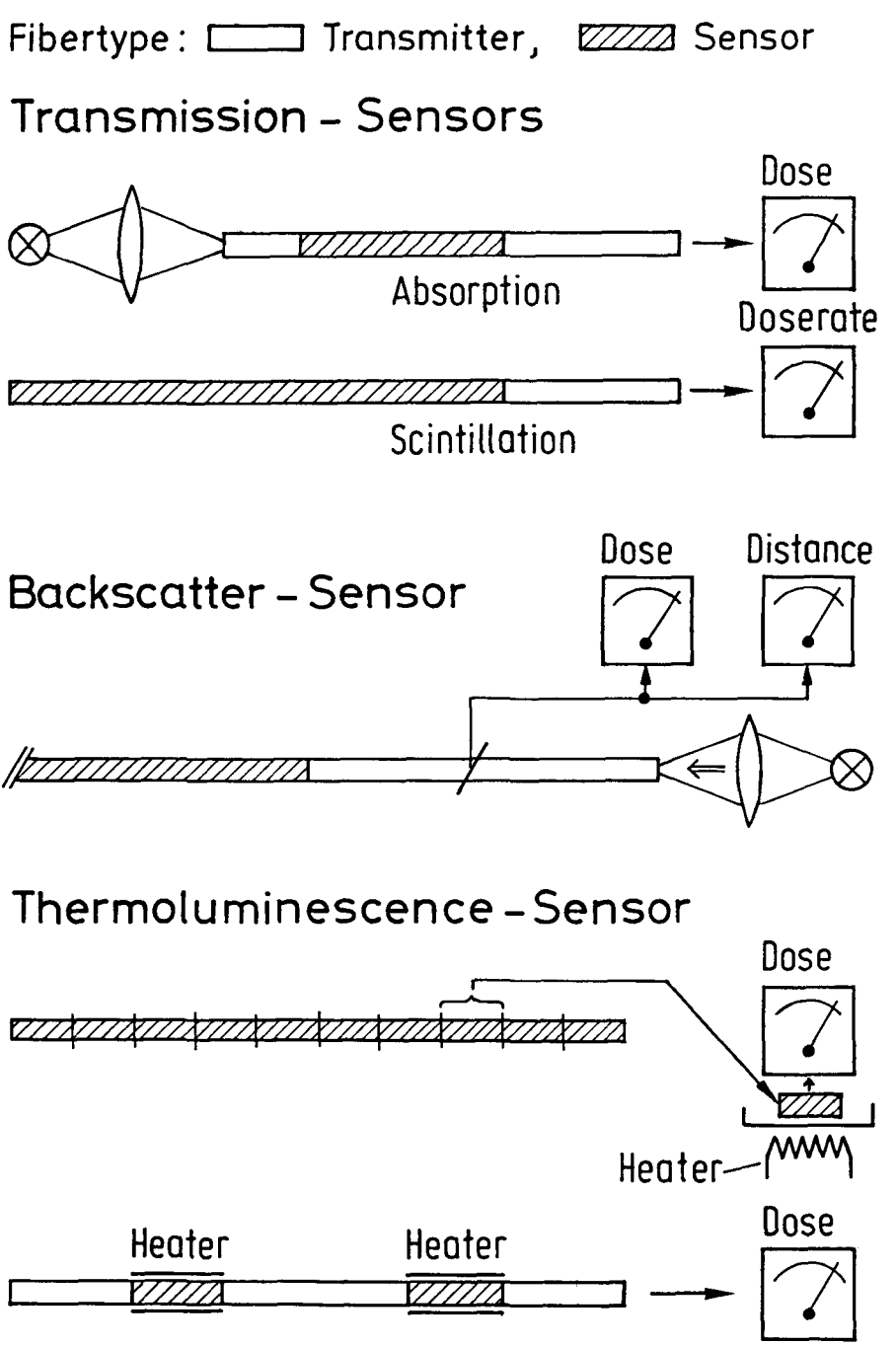

Fig.3: Schematics of fiber optic radiation detectors

Electrons freed by ionizing radiation and subsequently captured by shallow centers will be emitted to the conduction band by heat treatment. During heat-up, part of them recombine radiatively and visible light can be detected (Thermo-Luminescence). Fig. 2 shows the TL-glow curves of lead-glass (a), Ge-doped silica (b), and pure silica fibers (c) after X-ray irradiation.

In fig. 3, basic measurement setups are shown. In the case of transmission type sensors, the optical absorption or scintillation is evaluated directly from the light leaving the fiber end face. The absorption measurement can be performed with a simple detector, a lock-in amplifier system or just the human eye, depending on the required measurement range, sensitivity and accuracy.

Optical fiber waveguides made of scintillating materials feed the radiation-induced light pulses into an optical pulse counter, thus measuring the dose rate to which they are exposed. The connection of the scintillating fiber to the optical detector is done by a low loss transmitter fiber (1-10 dB/km).

The principle of the backscatter sensor is based on radiation-induced absorption, too. It gives information about the dose and distance on the fiber by use of the OTDR (Optical Time Domain Reflectometry) technique (see also fig.4). A short light pulse is launched into the fiber and partly scattered back (Rayleigh scattering). The shape of the backscattered signal contains the desired informations. The slope of the backscatter signal yields the local optical loss in dB/km, and the time axis can be converted to a space coordinate by multiplying by the group velocity of the light pulse. The dose is evaluated by introducing the absorbance per unit dose as used in integrating absorption dosimeters.

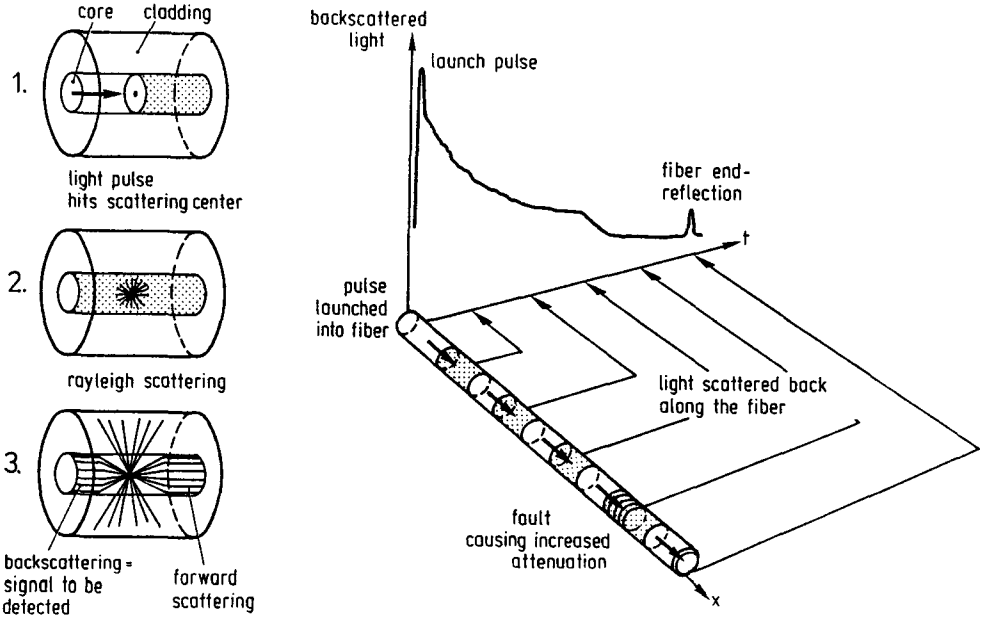

Fig.4: Schematic of absorption dosimetry by the backscatter technique

Advantages and Properties of Fiber Optic Dosimeters

Fiber optic waveguides have very small dimensions (10 to 200 um diameter) and usable lengths from cm to km. They are produced in high purity and uniformity by processes originating from semiconductor technology; they are highly resistive against chemical or thermal stress, and have no EMC-problems. They include both sensing and data transmitting in the same element and no power supply for recording is necessary. Compared with conventional detectors, the electronic equipment has not to be placed in radiation exposed areas.
Scintillation fiber optic detectors are far more rugged than scintillator- photomultiplier combinations. TL-dosimeters can be batch-calibrated and have a good reproducibility compared to the difficult handling of chip dosimeters.

The sensitivity of fiber optic dosimeters is highly dependent on the fiber material. The sensitivity of a lead-glass absorption-detector varies by a factor of 200 by changing the recording wavelength from 1.4 to 0.6 um. With an measurement accuracy of .1 dB and a sensor length of 10m exposure levels in the mSv to kSv range can be measured. Annealing of this fiber at room temperature occurs at a rate of 0.3 %/h. The sensitivity of Ge-doped fibers is much smaller than that of lead-glassfiber so that this type of fiber can be used as detector at exposure levels hundred times larger than lead-glass. This fiber reveals a TL-peak at 270 ^{o}C indicating a better stability than that of lead-glass fibers where the peak occurs at 150 ^{o}C. The spatially resolving limits of OTDR are at about 1-10 m for some km length using presently commercial available semiconductor lasers. The stored dosimetric information can be erased by the application of extreme high power laser light or temperatures above 400^{o}C in the case of TL-measurements. One drawback of this type of sensormaterial is the differently energy dependent response compared with tissue due to the higher Z.

Applications of Fiber Optic Dosimeters

The properties of fibers facilitate some applications, otherwise obtainable only with much more effort: A filamentary array spans over a large area and detects local radiation fields, with remote measurement electronics in environmental surveillance applications(i). The analyzer can be coupled to several fibers in the same detector cable with different sensitivities using optical switches. There is no need for carrying each detector separately to a central station(ii). Concentrating the fiber in a close volume increases radiation-induced attenuation proportional to the fiber length irradiated and thus increases the effective sensitivity (iii).

The above mentioned results were obtained with selected commercial telecommunication-fibers in the laboratory. The development of special fibers for dosimetry and of fiber optic dosimeters has recently started in cooperation with Felten und Guilleaume Energietechnik. The today costs of fibers are about 4 $ per m and the prices of the required electronics are comparable to conventional TLD systems.

The fiber-TL-dosimeter either uses conventional TLD readers, or the fiber is heated up locally in small capillaries. In the second case, the detector can remain installed and the light is transmitted via fiber to the detector.

Development of an Operational Multicomponent Personnel Neutron Dosimeter/Spectrometer DOSPEC*

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Introduction

The disadvantages of single component dosimeters such as albedos, NTA film, fission track and recoil track detectors have been discussed for many years.[1-2] Poor energy response, lack of sensitivity, high cost, use of fissionable radiators, fading and other problems can be attributed to one or more of the detectors currently used for operational dosimetry. A number of workers[3-5] have suggested using multiple detectors in personnel dosimeters. In this way, the advantages of each detector element would, to some degree, offset the disadvantages of the others.

The composite dosimeter concept has not received wide acceptance, in part because the use of additional detectors implies increased sample processing and cost. However, the albedo detector is one element commonly proposed for most multicomponent dosimeters. Noting that the TLD albedo is very sensitive and readily automated, we have developed a multicomponent dosimeter that uses the albedo detector both to provide the measurement of low energy neutrons and as a screening element. The track detector components -- CR-39 and polycarbonate -- need only be processed if the TLD indicates that there has been an exposure to neutrons. Since the three components each have significantly different energy responses,[4] the DOSimeter can act as a crude SPECtrometer, thus the name DOSPEC.

DOSPEC Description

The DOSPEC components are contained in a cadmium box originally used for the Hankins albedo dosimeter (Fig. 1). In addition to the TLD, we use three pieces of CR-39 and three of polycarbonate. Originally, the commercial cellulose nitrate--LR115--was included as a fourth component. However, we found that it has marginal sensitivity for neutrons of interest to us (less than 10 MeV) and the results had unacceptably high uncertainties.

The albedo component (TLD 600 and 700 chips) is processed using hot gas readout techniques. If the results indicate a neutron exposure, we etch the CR-39 and polycarbonate. The CR-39 is chemically pre-etched for five hours, then electrochemically etched on one side for five hours. We etch the polycarbonate electrochemically on both sides for five hours following a one-hour exposure ultraviolet light to enhance the normal photo-oxidation process. Processing details are provided elsewhere.[6]

Optical track counting, even when the tracks are enlarged by electrochemical etching, is a tedious and subjective process. We have adopted the use of bacterial colony counter[8] with a microscope and external TV camera to reduce counting time. The counter is now being interfaced to a desk top computer (Fig. 2) as a step toward eventual system automation. The responses from three components, plus those from control dosimeters exposed to either ^{252}Cf or PuBe sources, are used as input for an 800 step programmable pocket calculator code. The code calculates fluence values (spectrum) in energy bands thermal to 0.1 MeV, 0.1 to 1.5 MeV and above 1.5 MeV, as well as dose equivalent values in the same energy bands. A crude printer plot is also available on request.

*This work was performed under the auspices of the U. S. Department of Energy by Lawrence Livermore National Laboratory under contract No. W-7405-ENG-48.

DOSPEC Experience

As a result of a low incidence of neutron exposures at our laboratory, we have had little opportunity to evaluate DOSPEC operationally. However, we have participated in the 1982 CEC/ORNL Personnel Neutron Dosimetry Intercomparison. The dose equivalent results are presented in Table 1. It is important to note that the dose equivalent values were determined without making any corrections that require prior knowledge of the spectra. The largest error for administered dose equivalent values over 1.20 mSv (120 mrem) was 45%. This is in marked contrast to the experience of intercomparison participants using single component dosimeters.[10] This experience is, however, more consistent with others who have used the combination detector approach.[10] A summary of the spectral distributions determined by DOSPEC compared with reference values is presented in Table 2.

Summary

DOSPEC has been in operational use for over two years, but a low worker exposure history has limited the experience with system performance. However, participation in the CEC/ORNL intercomparison provided results that support the multiple detector concept and validate its ability to estimate the spectrum.

Table 1. DOSPEC Results Reported for the 1982 CEC-ORNL Personnel Neutron Dosimetry Comparison

Neutron Field Description	Reference Value - (mSv)	DOSPEC Value - (mSv)
HPRR-no shield	0.62	0.67
	11.1	9.3
HPRR-13 cm steel	0.64	0.51
	11.0	8.01
HPRR-20 cm concrete	0.48	0.43
	9.43	7.38
HPRR-12 cm Lucite	0.59	1.32
	11.0	11.8
0.57 MeV	0.70	0.98
	8.37	6.68
1.2 MeV	6.00	6.56
	1.50	2.17
5.3 MeV	6.50	7.33
	3.99	5.10
15.0 MeV	13.1	18.2
	1.05	1.31
^{252}Cf-15 cm D_2O	10.9	11.6
	1.20	1.91

Table 2. DOSPEC Spectral Distribution Comparison for 1982 CEC-ORNL Personnel Neutron Dosimetry Intercomparison Fields.

Neutron Field Description	Fluence per Energy Band, Normalized to Unity			
	Thermal - 0.1 MeV	0.1 - 1.5 MeV	>1.5 MeV	
HPRR-no shield	0.295	0.459	0.246	D
	0.143	0.568	0.288	R(a)
HPRR-13 cm steel	0.478	0.388	0.134	D
	0.119	0.798	0.0833	R(a)
HPRR-20 cm concrete	0.718	0.181	0.101	D
	0.646	0.226	0.128	R(a)
HPRR-12 cm Lucite	0.681	0.193	0.126	D
	0.734	0.141	0.125	R(a)
0.57 MeV	0.000	1.000	0.000	D
	0.000	1.000	0.000	R
1.2 MeV	0.000	0.956	0.044	D
	0.000	1.000	0.000	R
5.3 MeV	0.000	0.318	0.682	D
	0.000	0.000	1.000	R
15.0 MeV	0.101	0.000	0.899	D
	0.000	0.000	1.000	R
^{252}Cf-15 cm D_2O	0.807	0.116	0.077	D
	0.721	0.149	0.130	R(b)

D-DOSPEC R-Reference Values: (a) - Calculated from data in [8]
(b) - Calculated from data in [9]

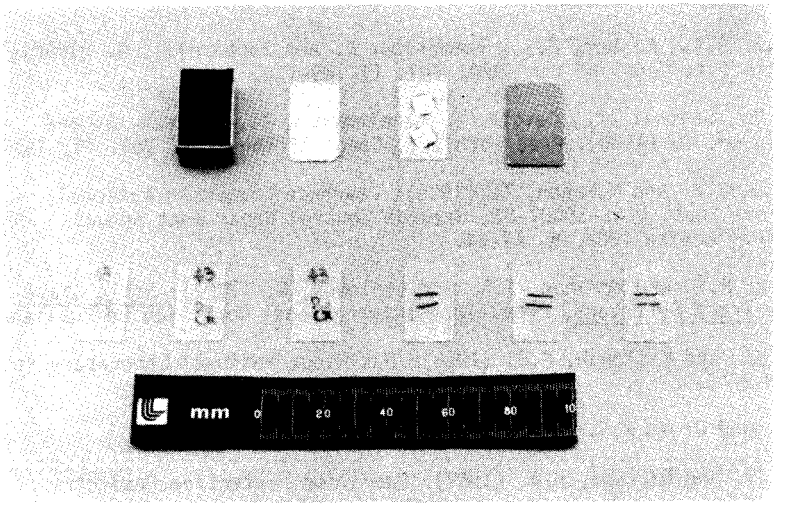

Figure 1. DOSPEC components with cadmium container.

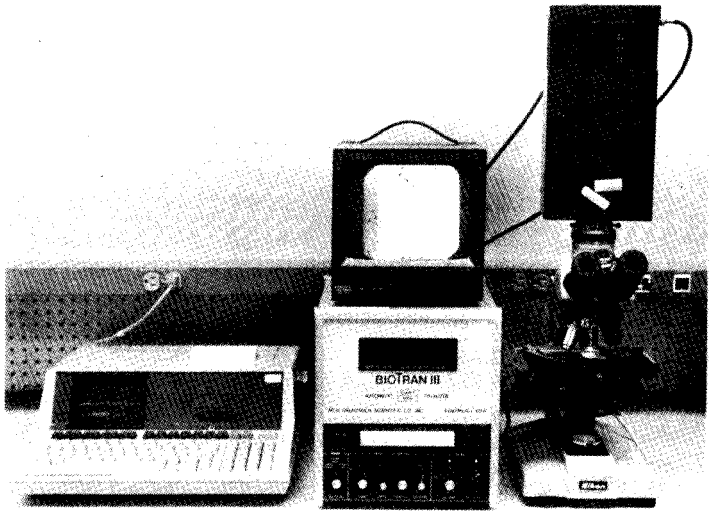

Figure 2. Bacterial colony counter for measurement of track density on CR-39 and polycarbonate, together with desk top computer.

References

1. Sayed, A.M. and Piesch, E. (1974): Kernforschungszentrum, Report No. KFK-2032.

2. Griffith, R.V., Hankins, D.E., Gammage, R.B., Tommasino, L. and Wheeler, R.V. (1979): Health Physics, 36 p. 235-260.

3. Distenfeld, C.H. (1973): Fourth AEC Workshop on Personnel Neutron Dosimetry, Rept. No. BNWL-1777, Battelle-Pacific Northwest Laboratories, p. 20, Richland, Washington.

4. Griffith, R.V., Fisher, J.S., Tommasino, L. and Zapparolli, G. (1980): Proc. 5th Int. Cong. of the IRPA, Vol. II 169-172.

5. Eisen, Y., Sharnai, Y., Oavdia, E. Karpinovich, Z., Faerman, S. and Schlesinger, T. (1980): Proc. 5th Int. Cong. of the IRPA, Vol. II, 157-160.

6. Griffith, R.V. and McMahon, T. (1983): Lawrence Livermore National Laboratory Rept. UCRL-50007-82, Hazards Control Department Annual Technology Review 1982, p. 17-24.

7. Griffith, R.V. and McMahon, T.A. and Espinosa, G. (1983): Solid State Nuclear Track Detectors, Proc. 12th International Conf. (To Be Published).

8. Sims, C.S. and Killough, G.G. (1981): Oak Ridge National Laboratory Report ORNL/TM-7748.

9. Ing, H. and Cross W.G. (1983): Health Physics (In Press).

10. Sims, C.S. and Dickson, H.W. (1984): Radiation Protection Dosimetry (To Be Published).

RADIATION PROTECTION STANDARDS FOR THE 1990s

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National Radiological Protection Board

The recommendations of the International Commission on Radiological Protection have formed the basis for most national and international regulations in this field. The most recent revision of the basic standards was Publication 26, issued in 1977, and it is timely to review how the ideas in that publication are standing up to the test of practical implementation and to consider where ICRP might concentrate in its next revision.

The Strengths and Weaknesses of the ICRP Standards

The principal strengths of the ICRP Standards set out in Publication 26 are that they provide a coherent self-consistent system of limiting exposure, the biological bases are clearly set out, the objectives are defined, and the method of achieving them is explained.

The international framework of the Commission's work is a strength but also a weakness. The recommendations should be, and in practice have proved to be, broadly acceptable all over the world. The basic aim, the level of protection sought, has been to keep the occupational risk of death and comparable other effects below a value of about 1 in 10,000 per year. This choice is a social judgement and the validity of a single social judgement covering the whole world must always be open to argument.

At a more detailed level, there is an interaction between the choice of the dose limits and the expected dose distribution over individuals. Other parts of the dose limitation system influence the distribution and could thus influence the choice of the dose limits. In any case, the dose limits alone would not give sufficient protection. This fact, together with the complexity of the overall system, makes it easy to attack the dose limits, and thus to undermine the whole protection system.

Problems for National Authorities

The problems for national authorities will depend on the national tradition of regulatory controls and only a few points can be dealt with here.

The choice of a dose limit

In 1977, for the first time, ICRP explained in some detail the basis of its social judgements. The Commission decided that it would be sensible to make its recommendations on the policy (with inevitable simplifications) that industries using radiation should not be more dangerous in terms of risk to life than those other industries thought of across the world as being reasonably safe. The figure selected by the Commission of 100 accidental deaths per million workers per year as being representative of more or less safe industries still seems a reasonable international compromise figure, but it will be open to challenge in relation to an individual country.

Alternative approaches can be adopted at the national level. The problem facing national authorities is whether to accept the ICRP limits with the advantages of uniformity between countries or to adopt the more complicated and protracted process of negotiating limits of its own, with the consequent

confusion in international discussions and agreements.

The optimisation of protection

It is logically inescapable that there needs to be some system of keeping all exposures as low as can reasonably be achieved if the exposure is to an agent - any agent - which does not have a threshold of exposure below which there will be no adverse effects. National techniques for dealing with such exposures vary very widely. The United Kingdom uses phrases like "so far as reasonably practicable", which are, in effect, very similar to the ICRP phrase "as low as reasonable achievable". The United States of America has some basic legislation which apparently has the effect of aiming for zero exposure or at least requiring the use of the best available technology (regardless of cost?) to achieve reductions. The practical difficulty for both operators and regulators is that of deciding how much more effort to direct to reducing exposures which are already low, but which may yet carry some residual risk. ICRP and other bodies working on the problems of radiation have provided only a framework. National authorities have now to solve their own problems including that of judging the trade-off between, for example, worker and public doses or doses at different levels and different times.

Collective effective dose equivalent commitment (CEDEC)

One of the least endearing of the Commission's creations for the common practitioner is "CEDEC". This multiple integral is rooted in the dual belief that a large number of small doses are equivalent to one large dose and that a dose now is equivalent to a dose at any time. Since the quantity was introduced to help with decisions and it is clear that those who make decisions hold neither belief, CEDEC should be disaggregated.

Committed effective dose equivalent

The complexity of effective dose equivalent with the implied need to determine organ and tissue doses throughout the body has also been seen as a problem for national authorities. In fact, it is not. For external exposure, the results of personal dosemeters or of suitably designed and calibrated field instruments are quite adequate for control and regulatory purposes. For internal exposures, the quantity has been of enormous help to ICRP Committee 2 in formulating annual limits of intake (ALIs) and derived air concentrations, both of which can be used directly without the need for the user to invoke effective dose equivalent at all.

One bone of contention has resulted from the impact on traditional monitoring methods for long-lived, long-retained nuclides, such as plutonium-239, of the ICRP emphasis on ALIs. The use of the ALI contrasts sharply with those monitoring techniques which have traditionally led to results expressed as fractions of the 'maximum permissible body burden'. Taken alone, monitoring techniques based on the maximum permissible body burden allowed many years worth of maximum permissible intake to be received in a single year. Publication 26 has merely added emphasis to the Commission's long-standing objection to this process of buying up future years.

We realise that there is a genuine practical difficulty, caused by the fact that individual monitoring techniques cannot always detect an intake over a year of one ALI. Reliance has then to be placed on air sampling or other forms of environmental measurement and individual monitoring can be used only as a longer term confirmatory test. The problems are greatly exacerbated if the national regulatory (or medical) approach requires a worker to be withdrawn from his normal duties for the rest of the year if he has been subject to

higher doses or environmental concentrations over the year than the annual limits permit. Difficulties of execution should not , however, overwhelm rightness of intention.

Areas of Development for the 1990s

It takes several years for national authorities to reflect any significant changes in ICRP recommendations. The Commission must therefore maintain a long-term stable position. It must not, however, ignore changes in scientific information or in social attitudes, nor must it be seen to lag behind national authorities. The early 1990s should see the completion of the next major review, and if necessary revision, of the Commission's main recommendations. We will now speculate on some factors which might underlie this review.

The scope of the recommendations

Since Publication 26 much effort has gone into clarifying that the ICRP system of dose limitation - but not the numerical dose limits - applies to natural radiation. The only source of irradiation remaining partly outside the system is medical use. The Commission may well consider how control of exposures to all sources of radiation can be more clearly unified.

Scientific information

Two trends seem likely. Risk estimates at high doses and high dose-rates may well rise as more data come forward from existing human studies. Secondly, increasing understanding of mechanisms may allow better use to be made of the non-linearity of the dose effect relationship for low LET radiations. For practical reasons, it will still be necessary to assume a linear relationship over a wide but limited range of doses, but a lower slope may be appropriate.

The combined result could lead to a recognition that the present limits are more conservative than we thought for low LET radiation but that a larger value of quality factor may be needed for high LET radiation if we wish to retain that conservatism for all qualities of radiation. The logical step of increasing the dose limits at the same time is not likely to find favour.

The present dosimetry system depends on the approximations that Q adequately reflects the different radiation qualities regardless of the irradiated tissue and that the tissue weighting factor adequately reflects the differences in organ sensitivity regardless of the type and quality of the radiation. If these assumptions were shown to be so inaccurate as to be untenable, the implications would be considerable.

Metabolic data will have improved but a complete revision of Publication 30 seems very unlikely. If the new data bear on nuclides of genuine importance, an amending statement would be appropriate. There is also a need for clear guidance on ALIs for members of the public. At the time of writing, a statement of the basis of this guidance is in press.

The control period

There will certainly be pressure for a dose control related to periods longer than a year. This is not a simple proposition. Clearly the control period for radiation dose should not include extended periods of other employment subject to other risks. Lifetime dose limits are thus improper and periods much over a year need to be defined with great care. The system could easily become bureaucratically complex.

The optimisation of protection

The identity of the terms "optimisation of protection" and "as low as reasonably achievable, economic and social factors being taken into account" will have to be emphasised and the policy may need reviewing. For example, is the correct point of choice the minimum of the total detriment plus cost curve (if all the quantities have been estimated without bias), or is that point only the minimum level of protection? If the latter, how much more safety should be added?

One of the most difficult problems will be balancing the interests of the individual and the community. The present emphasis on collective dose is seen by some as a distortion of priorities. Further publications will need to be more explanatory and less dogmatic on issues of this kind. One problem concerns the methods of optimising protection against events which may not happen.

The practical techniques will need more attention. The Commission will need to reconsider its present emphasis on cost benefit analysis which is only one of the available techniques and perhaps not the most useful.

Consultation

The Commission will increasingly have to consider how best to ensure that its recommendations take account of current views. Even when using a wide range of national figures on Task Groups the Commission is taking a very small sample of world professional opinion. The cost in labour and time (and also in money) of issuing consultative drafts and reviewing the resulting comments is beyond the Commission's resources but some way is likely to have emerged by 1990 for tapping a wider range of expertise than at present.

A much more difficult area is that of reflecting current, and indeed future, opinions of the exposed public and workers. The Commission will have to maintain a careful balance between understanding and reflecting these views and retaining its own much valued standing as a body of independent individuals. The real places for consultation in standard-setting are at the national level, when the Commission recommendations are being transformed into national policies and at the workplace, when these policies are being implemented.

Conclusion

Radiation protection standards for the 1990s start from an enviable position of scientific groundwork, logical coherence, international agreement and reasonable practical acceptance. If they are to be improved then ICRP in particular should concentrate on maintaining its credibility by not becoming too detached from the real world. It may need to widen the involvement in its deliberations and to decrease the complexity of its pronouncements.

THE BASIC SAFETY STANDARDS FOR RADIATION PROTECTION PRESENT AND FUTURE OUTLOOK

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THE SYSTEM OF DOSE LIMITATION

The Basic Safety Standards (BSS), Safety Series No.9, 1982 edition, jointly sponsored by the International Atomic Energy Agency, the International Labour Organization, the Nuclear Energy Agency of the OECD and the World Health Organization, are based on the system of dose limitation recommended by the International Commission on Radiological Protection. This system incorporates the three basic requirements of justification of the practice; optimization of protection, which is synonymous with the principle of "doses to be kept as low as reasonably achievable, social and economic factors being taken into account" which is commonly known by its acronym ALARA; and individual dose limitation.

This paper is intended to contribute to the Congresses objectives by describing some of the main features of the BSS together with a discussion of some problems of implementation in practice and presenting the future outlook of optimization.

The implementation of the BSS policy requires that individual related requirements and source related requirements be carried out. The first requirement is a rather common feature to all norms for health protection, whereby the objective is to prevent relatively high exposures that could produce causal biological effects. However, the source related requirements are a device rarely used in other protection standards. Thus, although the individual related requirements provide safety to all individuals, the source related requirements require further decrease in the remaining collective detriment through the ALARA principle.

Individual related requirements: The dose limits in the BSS are not to be used for purposes of planning and design; thus it is not a priori permissible to approach the limits.

Secondary limits - related to the primary dose limits - have also been specified, e.g. in terms of limits of intake of radioactive material into the body. Annual limits on intake for members of the public have been recommended by the BSS; they are provisional values until appropriate ICRP recommendations become available.

As an individual may be exposed to several sources (present and foreseeable), the dose limits should not be used for limiting the maximum individual dose permitted to be received from a single source. The IAEA's Manual on Principles for Establishing Limits for the Release of Radioactive Materials into the Environment (IAEA Safety Series 45) recommends the use of "source upper-bounds" in order to ensure that the sum of all controllable exposures which may be received by a selected individual from various sources will not exceed the dose limits.

The fraction of the exposure due to natural sources not technologically enhanced is not considered in the summation of individual dose rates, which is to be compared to the limits. Individual exposures as patients, apart from research, are also excluded as individual benefit is assumed to override the risk. On the assumption of proportionality between radiation dose and probability of harm, the harm from natural background would be independent of any additional doses from artificial sources and also would not influence the harm from these sources and can be treated independently.

Source related requirements: These aim at keeping the stochastic health detriment ALARA; in other words optimizing protection.

OPTIMIZATION OF PROTECTION

Optimization applies to all situations where radiation exposures from a source can be controlled by protective measures, and could conceptually also be used for planning protective actions where a source may get out of control. The optimization requirement is intended for radiation protection only and can be applied independently of conventional protection requirements (e.g. a shield thickness is not related to any conventional protection parameter).

ACHIEVING OPTIMIZATION THROUGH COST BENEFIT ANALYSIS

Cost benefit analysis is shown in the BSS to be a simple quantification technique for the purpose of optimization of protection by minimizing the sum of actual protection cost and the cost assigned to the radiation detriment. However, it should be emphasized as ICRP did in its report No.37 on "cost benefit analysis in the optimization of radiation protection", that cost benefit analysis is only one method for optimization and other methods are not ruled out as long as the ALARA principle is respected.

Based on the ICRP recommendations, the BSS assumes that the stochastic component of radiation detriment (cancer and hereditary harm) is proportional to the collective effective dose equivalent (referred to in this document as collective dose). The concepts of risk, detriment, collective dose and its committment has been defined in the BSS. Only a general outline of some of the practical problems encountered will be briefly mentioned here.

SOME PRACTICAL ISSUES

Direct proportionality between dose and response. The validity of this assumption in the BSS has been challenged due to the absence of human data at very low doses. However, the fact is that doses to which people are currently being exposed to are not negligible. The United Nations Scientific Committee on the Effects of Atomic Radiation shows that such exposures are in the range of some milli sievert per annum. Therefore, the marginal relatively small increase in dose rate due to nuclear installation would fall within the range of the average dose rate incurred by the world population and as given in UNSCEAR 1982 and it is in this range of doses that linearity must be judged. Furthermore, as

reported during the IAEA Symposium in Venice, "... the amount of harm in any tissue should in large parts be simply proportional to the number of particle tracks within that tissue; and the dose effect relationship should then be linear at the small incremented dose rates with which we are mainly concerned in non-medical radiation exposures".

Truncation of dose rate integral: The question is, is it possible to disregard negligible rates and therefore arrive at the "de minimus" dose concept? The BSS do not provide any justification for neglecting individual dose rates - however small - in the collective dose rate assessment; but the standards do not preclude ignoring negligible individual doses provided that they result in a negligible collective dose.

Integration of collective dose rates over infinity: Some practices may involve long-lived radionuclides which can cause exposures over thousands or even millions of years. The marginal collective dose is the relevant quantity in the optimization process. If available, protection systems can prevent the release of radioactive material during different time periods; then, the marginal collective dose will become the finite time integrals of the collective dose rates over the difference in the lifetime of the systems provided that the relevant functions are not changed.

Assigning values to parameters involved in cost benefit analysis: Based on direct proportionality between dose and effect for the objective health detriment, the proportionality constant alpha is then the monetary value assigned to the unit collective dose. This valuation of life expectancy applies to unknown statistical individuals and not identified ones and its magnitude determines the attainable level of radiation protection. It has nothing to do with a valuation of human life but is a rational device for conserving lives.

Optimization of protection against releases of radioactive materials causing transboundary exposure: The IAEA in co-operation with WHO are in the process of recommending policies for such optimization. These recommendations include a minimum value of alpha to be applied to the part of the collective dose that causes exposures beyond the frontier of the country that controls the source. The recommended value is US \$3000 $(\text{man sievert})^{-1}$ adjusted to 1981 prices.

Distributional problems: In the BSS there are no discriminations in the assignment of value to different spacial and temperal components of a detriment. The IAEA Safety Series No.45 indicates that careful judgement by regulatory authorities must be exercised in deciding whether it is reasonable to attach less weight to doses far in the future. A group of experts convened by the Holy See's Pontifical Academy of Sciences recommend that future doses that can be avoided by protective measures should always be given the same weight as present doses.

Other components of the detriment: In the BSS the cost assigned to the non objective detriment components can be taken to be proportional, with a proportionality factor called "beta", to the number of individuals receiving high dose equivalents, and to a function of the individual dose equivalents which could depend on risk aversion attitudes and national regulations.

FUTURE OUTLOOK

Strictly speaking, the BSS system of dose limitation applies ony to controllable exposures. It could also be applied to exposures arising from incidents for which emergency arrangements for limiting exposures were planned in advance. However, human beings may also incur doses from sources having the potential, but low probability, to get out of control. Such potential, low probability exposures may occur from both unexpected situations as well as from foreseen accidents for which sufficient emergency arrangements were not planned. Examples of these are large accidents in nuclear installations or disruptive events in radioactive waste depositories. In many of these cases, the eventual exposure is of probabilistic nature in the sense that a probability can be defined to quantify the chance of the exposure occurrence. This probability need not necessarily be objectively determined from experimental data; conceptually, it can be subjectively assumed as a 'degree of belief' and substantiated by logical and technical arguments of deterministic nature.

It is tempting to associate the idea of limiting the probabilities of occurrence of accidents, and their subsequent exposures, with the two relevant and interrelated requirements of the BSS dose limitation system, i.e. limiting the probability of individual harm, and keeping the mathematical expectation of collective harm to a level as low as reasonably achievable. It can be realistically assumed that this idea will be worked out in the near future and that substantial conceptual developments on the basic criteria will be carried out.

CONCLUSIONS

The BSS system of dose limitation is expected to have a substantial impact on radiation protection regulations all over the world. Many of the practical problems are solved but new issues appeared. It is felt that the current status of both the system and the optimization of protection can be seen from a positive perspective.

Current issues such as nuclear safety goals or criteria for radioactive waste disposal could be more rationally treated. The IAEA is looking into this problem with particular attention and closely follows scientific developments on the subject. In other forums the matter is also under consideration. The future outlook for the principles of the BSS system of dose limitation is highly promising.

CANCER COMPENSATION CRITERIA FOR RADIATION WORKERS: USE OF THE PROBABILITY OF CAUSATION APPROACH

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A need exists for methods to assess the contributions of carcinogenic agents, especially ionizing radiation, to the production of various types of cancer in man, on both an individual and population basis. Such methods are essential to the evaluations of risk to health that underlie important activities such as the setting of health protection standards and safety goals, and decisions on important matters including health research, the disposal of toxic wastes, and potential liability for compensation claims.

The use of attributable risk and probability of causation concepts for assessing workmen's compensation for cancer are reviewed and applied to radiation workers in U.S. nuclear power plants. Estimates of incidence for certain primary cancer sites in these workers are used to derive possible compensation costs for the period 1980 to 2045: these costs are shown to range from about 30 to 400 million U.S. dollars, depending primarily on the level of probability of causation above which compensation is paid, and on the compensation schedules selected. The probability of causation approach is shown to be quite sensitive to input parameter selection. Further, if the worker population is expanded to include those workers receiving occupational radiation exposure other than at nuclear power plants, the estimate of potential compensation liability overall would be larger, perhaps by a factor of 2 to 10.

Ultimately, the possible future liability for the nuclear industry will be determined largely by the level of probability established by law as the floor for compensation of cancer cases. Results from this study suggest that for the major cancers of interest, this floor should not deviate too far below the 50% probability of causation level, and that levels as low as 10% would be unjustified technically and would lead to gross public misunderstanding of the actual radiation risks involved. Of equal importance is the selection of the compensation schedules associated with these liability levels, because the combination of these two parameters primarily determine future compensation impact overall.

PROPOSED REVISION OF 10 CFR PART 20-- USNRC STANDARDS FOR PROTECTION AGAINST RADIATION

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Following issuance of ICRP Publication 26 in 1977, there was increased awareness in the U.S. Nuclear Regulatory Commission (NRC) staff of the need to review and make recommendations on revising the NRC's present 10 CFR Part 20, "Standards for Protection Against Radiation." An interoffice drafting group was established in 1981 to develop a proposed revision.

The staff considered several alternative plans to revise Part 20. One such option was to update only those sections, such as values of concentrations of radionuclides in air and water, which would be most affected by the increases in scientific understanding of biological models, of biochemical behavior of specific radionuclides, and of assessment of risks. However, updating these features alone would incur a large portion of the impact of a complete revision of Part 20 without the benefits associated with a consistent radiological protection system which includes summation of risk from external and internal doses, and consideration of weighted doses to multiple organs from any number of radionuclides.

ICRP APPROACH

The revision would be consistent, in principle, with most of the recommendations of the ICRP set forth in ICRP Publications 26, 30, and 32. It would, however, retain the so-called "special units" (curie, rad, and rem), which are more familiar to NRC and State licensees than the International System of units.

The proposed revision would adopt the ICRP risk-based system which provides dose limits based on estimated risks, comparing health risks in the nuclear industry with health risks in other industries, and adding doses from dissimilar exposure modes.

OCCUPATIONAL DOSE LIMITS

A 5-rem dose limit for occupational exposures would apply to the sum of the whole-body dose (deep dose equivalent) from sources outside of the body and the committed effective dose equivalent (the sum of the 50-year committed dose equivalent to each organ or tissue multiplied by the weighting factor appropriate to that organ or tissue) contributed by radionuclides taken into the body.

The proposed revision would provide an exception to the basic limits, that are based on 50-year committed effective doses, for a few very long effective half-lived radionuclides, such as plutonium. For these long-lived radionuclides, it would provide retrospective control of the exposures of an individual worker on the basis of annual effective dose equivalent. A number of provisions would be added, including prospective use of estimated committed effective dose equivalent in the design and operation of facilities, an added constraint of 3 rems/year on the contribution from specified long-lived radionuclides, and provision of reporting both the annual and committed dose estimates to the worker, along with instruction concerning the significance of both. The staff believes that the proposed exception, in essence, would codify good contemporary practices and would permit implementation at a very nominal cost.

The present Part 20 permits whole-body external exposures up to 3 rems/quarter, but over many years of exposure would not allow an individual to exceed an average of 5 rems/year (the 5(N-18) dose-averaging formula). Internal exposures are controlled by limiting exposure conditions, but the doses are not added to external doses in meeting the present quarterly limits.

The proposed revision would remove the 5(N-18) "dose-averaging" provision in the present Part 20, but would contain a provision for "planned special exposures." This provision is designed to be used only in exceptional situations when alternatives which might avoid the higher exposure are unavailable or impractical. In the proposed planned special exposures, doses as high as one times the annual dose limits would be permitted for a single event, but an individual's dose due to all planned special exposures could not exceed one times the annual dose limits in a year, or 5 times the annual dose limits in a lifetime.

In developing the proposed revision, consideration was given to regulating the exposure of classes of workers who might be at relatively higher-than-average risk from radiation exposures, e.g., minors and pregnant women (embryo/fetus). For minors (persons less than 18 years of age), the proposed annual limits would be 1/10 of those for an adult worker. Consideration is being given to providing a dose limit for the embryo/fetus of a declared pregnant woman. It would place on the woman the responsibility for the decision, after instruction, whether or not to declare her pregnancy. Following such a declaration, the licensee would be required to limit to 0.5 rem the dose to the embryo/fetus from occupational exposure of the pregnant woman throughout the period of pregnancy.

ALARA

The proposed revision would require all licensees to incorporate ALARA considerations into their radiation protection programs. However, the Part 20 revision would not require quantified optimization studies because of the difficulties in performing the analyses and because it is recognized that the decisions must be largely judgmental in any event.

PUBLIC DOSE LIMITS

The proposed revision of Part 20 would explicitly set the annual dose limit as 0.5 rem to an individual member of the public, considering all sources of both external and internal dose, other than natural background and medical diagnosis and therapy.

REFERENCE LEVEL

A licensee would be assumed to meet the 500 mrem/year limit if the licensee controls exposure to members of the public to doses within a 100 mrem/year reference level from its own operations. A licensee could apply to the NRC for prior authorization of operations which might result in public exposures greater than the 100 mrem/year reference level. Such applications would include the licensee's program which assesses and controls dose within the 500 mrem/year limit; would demonstrate a clear need to operate in excess of the reference level; and would be supported by an explanation to do so in terms of ALARA requirements.

DE MINIMIS

Consideration is being given to a de minimis feature, a level of risk (or dose rate, as a surrogate measure) so low that it would be a trifle in comparison to the risks which the individual is subjected to daily as part of normal living habits and activities. It would constitute a level of risk so low that no resources could be justified to control it, or to be further concerned with it.

However, concern has been voiced over the possible abuse in applying the provision for a "de minimis" individual exposure (1 mrem/yr) to activities, such as waste disposal and the use of radioactive material in consumer products, and the feature is being reconsidered. Another proposed de minimis provision would truncate consideration of collective dose at the point where individual doses do not exceed 0.1 mrem/year.

HIGH RADIATION AREAS

The requirements in the existing Part 20 for controlling very high radiation areas (500 rads/hour at 1 meter), applicable only to irradiators, would be modified and applied to all licensees.

STATUS

Our approach to the development of the proposed revision of Part 20 has been different from the usual rulemaking procedure. Before, and in addition to receiving formal review inside NRC, the issues in the draft revision have been discussed informally with individual members of the NRC technical staff, and several hundred representatives of national and international radiation protection communities, Agreement States, licensees, labor unions, and others. These individuals provided useful views, particularly in regard to technical and administrative problems foreseen in implementing the revision and identifying impact potential. The drafting group has attempted to address each of the principal issues in the draft statement of considerations. The draft revision reflects and attempts to resolve many of these issues with minimal impact, while maintaining the central thrust of the revision to ensure that protection against radiation is adequate when judged by contemporary standards.

Drafts of the proposed revision were circulated for formal NRC Office level review in May 1983. The drafting group is now considering changes to the proposed rule, statement of considerations, and all of the many supporting documents to reflect changes in response to suggestions from these Offices and other interested parties. Formal reviews of the proposed revision also will be provided by the NRC's Advisory Committee on Reactor Safety and the Committee for the Review of Generic Requirements. It is expected that these changes will require several months. Subsequent to these reviews, the plan is to submit the proposal to the NRC Commissioners for their decision on publishing the rule as a proposal for public comment. This is expected to occur in 1984.

EPA

Members of an interagency working group, including NRC and other involved Federal agencies, Conference of State Radiation Control Program Directors, and NCRP have been involved in discussions with the U.S. Environmental Protection Agency (EPA) concerning proposed "Federal Radiation Protection Guidance for Occupational Exposures." It appears that the revised EPA guidance will reflect much of the system of dose limitation recommended by ICRP.

CONCLUSION

The expected result of promulgating and implementing the proposed revision is an improved rule that provides better assurance of protection, establishes a clear health protection basis for limits, applies to all licensees in a consistent manner, and reflects current information on health risk, dosimetry, and radiation protection practices and experiences.

RADIATION EXPOSURE STANDARDS FOR RADON DAUGHTER BASED ON LUNG CANCER RATES IN NONSMOKERS*

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Introduction:

Risk based standards for allowable emissions of radionuclides to the environment require risk estimates which are as precise and accurate as possible. For radon and its associated decay products or "daughters" two studies of uranium miners, one from the United States (1, 2) and one from Czechoslavakia (3,4), are of sufficient size and duration to yield reasonably precise estimates of health effects, but their accuracy when applied to the general nonmining population is questionable because the principal health effect associated with radon daughter exposure is lung cancer and the principal cause of lung cancer is cigarette smoking. This would not present a problem, if smoking and radon daughters act additively in the induction of lung cancer, but the results of two recent analyses of the American miners data (5,6) as well as our review of the Czech studies suggest multiplicative interaction. The following discussion shows that, if smoking and radon act multiplicatively: (1) the smoking status of the population at risk in large part determines the magnitude of one's risk estimate, and (2) risk based standards for radon based on the general population contain a "hidden subsidy" for smokers. For these reasons, we argue that radon health effects coefficients and resulting risk based standards should be based on relative risk models applied to nonsmoking populations.

The Evidence for Multiplicative Interaction

Until recently, work on the American miners data seemed to suggest that though radon induced lung cancer might occur somewhat sooner (i.e., shorter latency) in smokers than in nonsmokers, the ultimate risk of lung cancer per unit of radon daughter exposure (commonly expressed as working level months (WLM) (7)) was comparable (2). However, a later study (6) using a Cox proportional hazard model (8) showed the interaction between smoking and radon daughter exposure was best expressed by a multiplicative model of the form:

$$R_{se} = R_s \cdot R_e \qquad (1)$$

where R_{se} is the risk in individuals exposed to both radon and smoking, R_s is the risk in those exposed to only smoking, R_e is the risk in those exposed to only radon, all relative to risk in those exposed to neither. A still more recent evaluation of the same data (5) verified that equation (1) gives a good fit to the American miners data and used an extension of the Cox model (9) to formally reject an additive model.

The Czech miner data has not had the benefit of analysis by the newer procedures mentioned above, but BEIR III (10) notes that relative risks of 4.7 and 5 are seen in radon exposed smokers and nonsmokers respectively, assuming that background lung cancer rates in Czech smokers are 10 times those in nonsmokers. Thus, it appears that multiplicative interaction is the rule in these data also.

Calculating the Influence of Smoking

Actuarial populations of "pure" smokers and nonsmokers were developed using the 1969 United States abridged life tables for white males and females (11), lung

*The views expressed here are those of the authors and do not reflect an official position of the U.S. Nuclear Regulatory Commission.

cancer rates for nonsmokers taken from the American Cancer Society Study of nonsmokers (12), and general smoker mortality data from the 1979 Surgeon General's Report (13). This calcuation assumes known age specific lung cancer death probabilities for smokers, a constant all cause standardized mortality ratio (SMR) (14) for smokers versus nonsmokers for ages 40 and above (we assume 1.75 for males and 1.5 for females), and known age specific percentages of smokers in the base population (we assume 10-45% in males and 10-33% in females, depending on age). This information is used to separate the general population life table into two life tables, one for smokers, the other for nonsmokers. Details of the procedure are given in Ginevan (15).

Given these life tables we can calculate health risk for a particular level and duration of radon daughter exposure and either a relative risk (RR) model of the form:

$$Q_{ex} = B \cdot d_x \cdot Q_{cx} \qquad (2)$$

where Q_{ex} is the age specific excess probability of dying of lung cancer, B is a risk coefficient with units of percent increase per WLM, d_x is the effective exposure in WLM at age x, and Q_{cx} is the baseline probability of dying of lung cancer at age x, or an absolute risk model (AR) of the form:

$$Q_{ex} = C \cdot d_x \qquad (3)$$

where C is a risk coefficient with units of cancers per year per WLM. The modeling approach is a cohort life table (16) which calculates effects in terms of premature deaths per 100,000 persons and months of lost life expectancy. The model itself is described in detail in Ginevan (15).

Modeling Results and Discussion

Our radon risk evaluation assumed the parameters given below:

Period of exposure - lifetime; exposure per year-1 WLM; $^{+}$latency-10 years; *age at first risk-30; relative risk coefficient - 0.01/WLM; absolute risk coefficient - 1×10^{-5} lung cancers per year/(WLM.person).

The results of our risk modeling are shown in Table 1. (We note the differences between smokers and nonsmokers in terms of both life expectancy and risk of lung cancer are in accord with those given in the Surgeon General's Report (13)). Under the relative risk model there are dramatic differences in the impact of radon exposure, depending on the smoking status of the population at risk. For the excess death measure, the ratio of the highest risk (smoking males) to lowest risk (nonsmoking females) populations is about 13.1. For loss in life expectancy the same ratio is about 11.7. Absolute risk models appear to provide a way out in that the same ratios are 1.5 for excess deaths and 1.8 for loss in life expectancy. However, this ignores the fact that our risk coefficients were derived from miner populations, over half of whom are smokers, and as shown earlier absolute risk models do not fit these data.

+Assumes dose must be received at least 10 years in the past to affect present lung cancer risk.

*Assumes lung cancer risk is zero before age 30 regardless of dose.

The "hidden subsidy" mentioned earlier arises because a given amount of radon daughter exposure is about 4 times more hazardous if one takes the general population rather than nonsmokers as a baseline. This difference is entirely attributable to smoking. For the same degree of protection a risk based standard for the general population is therefore about 4 times lower than one for nonsmokers. The cost of this fourfold reduction (which might be small or large depending on the scenario) is our "hidden subsidy."

One might argue that this subsidy is justifiable, but this contention is vitiated by considering the health risk voluntarily assumed by smokers. Reducing risk in nonsmokers to a level of 10^{-5} to 10^{-6} per person per year would limit exposure to about 0.1 WLM per year (17). An additional reduction to about 0.025 WLM per year would be required if general population risks are considered. The health benefit to smokers of this 0.075 WLM difference is about 250 excess lung cancer deaths per 100,000 or about .4 months of life expectancy. Compared to nonsmokers, smokers' excess lung cancer risk is about 5280 cases per 100,000 and their loss in life expectancy is about 4.6 years. Therefore the health benefit is either 5 percent (excess deaths) or 0.7 percent (life expectancy) of the risk smokers have voluntarily assumed. Thus, if the cost of the fourfold reduction in radon daughters dictated by the presence of smokers in the population is even moderately large, it seems difficult to justify.

Conclusions

Radiation protection standards for radon daughters based on the general population contain a "hidden subsidy" because of enhanced risk in radon daughter exposed smokers. While the cost of this subsidy is variable, the health benefits to smokers are small relative to the risk they have voluntarily assumed. The principal conclusion of this exercise is therefore that radiation protection policy for radon daughters should be based nonsmokers, except where the "hidden subsidy" for smokers, inherent in standards based on the general population, is negligible. Validation of such a policy will require careful consideration of what "reference" nonsmoking populations should be assumed. Likewise, careful consideration should be given to both new analyses of existing data sets, and if possible studies of new populations, to rigorously test the assumptions put forward here. We feel that such efforts will allow development of a radiation protection policy for radon daughters which is both equitable and cost effective.

TABLE 1: LIFE EXPECTANCY (LE) LOSS IN LIFE EXPECTANCY, LIFETIME RISK OF LUNG CANCER AND EXCESS LUNG CANCER DEATHS (GIVEN AS DEATHS PER 100,000 AT RISK), FOR SIX POPULATIONS EXPOSED TO 1 WLM PER YEAR LIFETIME, UNDER THE RR AND AR MODELS.

	MALES			FEMALES		
	GENERAL POPULATION	NON-SMOKERS	SMOKERS	GENERAL POPULATION	NON-SMOKERS	SMOKERS
LE	67.82	70.29	64.81	75.23	76.48	72.69
LE Loss (Months)						
RR	3.94	0.56	6.55	1.20	0.56	2.29
AR	3.95	4.52	3.26	5.45	5.78	4.78
Expected Deaths	4662	683	9529	1157	623	2344
Excess Deaths						
RR	2641	426	5213	669	398	1295
AR	1714	1874	1519	2152	2238	1977

References:

(1) Lundin, F. E. Jr., J. K. Wagoner, and V. E. Archer. 1971. Radon Daughter Exposure and Respiratory Cancer, Quantitative and Temporal Aspects. NIOSH/ NIEHS, Joint Monograph No. 1, USHEW, PHS (NTIS No. PB 204 871), Washington, DC GPO.

(2) Lundin, F. E., Jr., V.E. Archer and J. K. Wagoner, 1979. Energy and Health, N.E. Breslow and A. S. Whitemore, Eds., Proceedings of a SIMS Conference, Alta, Utah, June 26-30. Pages 243-264.

(3) Kunz, E., J. Sevc, V. Placek, and J. Horacek. 1979. Health Physics 36:699-706.

(4) Sevc, J., E. Kunz, and V. Placek. 1976. Health Physics, 30:433-437.

(5) Whittemore, A.S. and A. McMillan. 1983. JNCI 71:489-499 In Press.

(6) R. W. Hornung, and S. Samuels. 1981. Radiation Hazards in Mining: Control, Measurement, and Medical Aspects, M. Gomez (ed.) Soc. Mining Engineers of Amer. Inst. Mining, Metallurgical and Petroleum Engineers, Inc. pp. 363-368.

(7) Evans, R. D. 1980. Health Physics 38:1173-1197.

(8) Cox, D. R. 1972. J. Roy, Statist. Soc. Ser. B 34:187-220.

(9) Thomas, D.C., 1981. Biometrics 37:673-686.

(10) NAS 1980. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980. National Academy Press, Washington. 524 + xv pp.

(11) NCHS 1974. Vital Statistics of the United States 1969, Volume II, Part A. US National Center for Health Statistics. HRA 74-1101.

(12) Garfinkel, M.A. 1981., JNCI 66:1061-1066.

(13) USPHS 1979. Smoking and Health, A Report of the Surgeon General. DHEW Publication (PHS) 79-50066. 1212 p.p.

(14) Fleiss, J.L. 1981. Statistical Methods for Rates and Proportions. John Wiley. New York 321 + xviii pp.

(15) Ginevan, M.E. 1983. A Computer Code for General Analysis of Radon Risk (GARR) Draft NUREG No. 1029

(16) Bunger, B.M., J.R. Cook, and M.K. Barrick. 1981. Health Physics 40:429-455.

(17) ICRP 1977. ICRP Publication 26. Permagon Press. New York.

A RADIOACTIVE MATERIALS TRANSPORTATION TRAINING PROGRAM

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INTRODUCTION

Sandia National Laboratories is a multidisciplined research and development facility operated for the Department of Energy by Western Electric. The laboratory employs approximately 7500 people and is principally located in Albuquerque, New Mexico, and Livermore, California. It has remote facilities and field test activities in many other areas of the United States and around the world. The major responsibility of the laboratory is national security. Other responsibilities involve aspects of nuclear reactor safety, nuclear waste disposal, nuclear materials transportation safety, and fusion, solar, wind, geothermal and fossil fuel energy research. These diversified activities generate the transportation in commerce of hundreds of hazardous materials shipments each year.

In 1982, the Albuquerque facility shipped 1076 hazardous materials shipments with 374 shipments containing radioactive materials (RAMs). Most of the radioactive shipments were Type A quantity RAMs grouped as Radioactive Materials, N.O.S. and Radioactive Device, N.O.S. Eighty-three percent of the shipments were transported by common or contract carriers; 17% were transported by Department of Energy Safe Secure Trailers. Thirty-five percent of the shipments required no radioactive labels, 19% required Radioactive White I labels, 42% required Radioactive Yellow II labels and 4% required Radioactive Yellow III labels. This degree of RAMs transportation activity requires that a training program be developed and implemented to help insure both employee and public safety.

BACKGROUND INFORMATION

In October, 1979, Sandia's Environmental Health Department was given the responsibility for developing and administering a continuing Lab-wide Hazardous Materials Transportation Training Program for all employees that have any responsibility in the transportation process. The training policy was to be consistent with the Hazardous Materials Transportation Safety Act which states that all officers, agents, and employees having any responsibility for the movement of hazardous materials in commerce receive training commensurate with their individual responsibilities. Initially the training program consisted of either a two-hour orientation or a twenty-hour workshop.

The orientation was given to employees who work with hazardous materials and occasionally offer them for shipment. The objective of the orientation was to provide the employee with an overview of the transportation process with emphasis on the recognition, identification, and classification of hazardous materials in accordance with applicable regulations. Instruction was given on how to use this information in filling out the

necessary internal paperwork required to initiate a shipment. Approximately 275 personnel completed the orientation.

The workshop was given to employees that have a day-to-day responsibility in one or more phases of the transportation process or total transportation responsibilities at a remote site. The objective of the workshop was to provide the employee with an indepth coverage of the transportation processes and a working knowledge of the applicable regulations. The major emphasis was on packaging, marking, labeling, placarding, and preparation of shipping papers. Approximately 60 personnel completed the workshop. Both the orientation and the workshop informed the employee of their personal responsibility and liability under the law and that transporting hazardous materials is a team effort.

Subsequent and continued training involves the distribution of pertinent information through employee or management news briefs, orientations for new employees and specialized workshops on specific classes of hazardous materials. The Radioactive Materials Transportation Course is a specialized workshop developed as a continuing effort of the Hazardous Materials Transportation Training Program.

COURSE OBJECTIVES

The Radioactive Materials Transportation Training Course was developed for two equally important reasons. The Department of Transportation (DOT), the International Civil Aviation Organization (ICAO), the International Air Transport Association (IATA), and the Nuclear Regulatory Commission (NRC) recently adopted totally or in part the 1973 IAEA revised "Regulations for the Safe Transport of Radioactive Materials." These regulatory changes have an effect on the domestic and international shipments of RAMs. In order to comply with the regulations shipping and receiving personnel need to be made aware of and understand the changes. The second reason for developing the course is that it was apparent from the previous transportation workshops that the participants and even the packaging engineers with many years of experience had little understanding of radiation--what it is, where it comes from, how it is used or what are its hazards and risks. Their view of radiation was based largely on what they had heard or read from various news media sources or from various movies and television shows. The course would thus provide a better understanding of and compliance with the regulations by providing a better understanding of the fundamentals of radiation and a better perspective of the hazards in transporting RAMs.

COURSE DESCRIPTION

The course is approximately 20 hours in length and is conducted 4 hours per day for 5 consecutive days. It is directed at employees in our shipping and receiving organizations and in our traffic management organization. Many of these employees have limited educational backgrounds and therefore the course is constructed to be as simple and understandable as possible. The course is centered around the DOT/NRC regulations and uses lecture

materials, handouts, slide/tape programs and videotapes as methods of transferring training information. Most of the training materials are available commercially or through governmental agencies. Work exercises and quizzes are used as tools to indicate training effectiveness. Class sizes are kept at 12-15 students to allow greater individual participation and a more relaxed, informal atmosphere. Successful completion of the course requires only attendance and participation. Participants' names are kept on file for record and audit purposes and to verify that training requirements have been met. Approximately 25 employees have completed the course.

COURSE CONTENT

The basic outline of the training course is represented in Figure 1. The Introduction, Part I, provides a historical overview of the regulations as to purpose and application. It summarizes the magnitude of RAMs transportation and the necessity of the regulations by providing statistics of shipments by number, type, mode, accidents and incidents. General Hazardous Materials Requirements, Part II, reviews the fundamentals of identification and classification of hazardous materials, packaging, marking, labeling, placarding and preparing shipping papers. This review is based on a 60-minutes slide/tape program produced by Western Electric. The Radiation Fundamentals portion, Part III, is centered around two videotapes entitled "Radiation Naturally" and "Working with Radiation and Protecting the Unborn." These videotapes, reinforced with lecture material and handouts, explain what radiation is, where it comes from, how it is used and measured and answers questions about biological hazards and risks. This section introduces the definitions and terminology as used in the regulations.

In Part IV, RAMs Transport Hazards, we discuss the four basic types of radiation and their characteristics, radioactive contamination, criticality, and heating. Criticality is explained by use of one of two videotapes, "Criticality Safety Orientation" or "The Safe Handling of Enriched Uranium." In Part V, Theory and Principal of Controlling RAMs Transport Hazards, we discuss radiation dose guidelines and the methods of controlling internal and external radiation doses to these guidelines by packaging design, testing, monitoring and by quantity limitation. Packaging design and testing is explained by a Sandia-produced videotape entitled "Accident Testing." In this section we discuss basic health physics practices to reduce radiation exposure in both normal and emergency situations. In Part VI, Identification and Classification of RAMs, we discuss regulatory requirements pertaining to special and normal form materials, single, mixed, and fissile radionuclides. We discuss activity determinations, the new A_1 and A_2 values, and classification as to Type A_1, Type A_2, Type B, Limited Quantity or Low Specific Activity Material. We demonstrate the use of the DOT Hazardous Materials Table to deterine the proper shipping name, hazard class, UN number, labels, and where to find packaging information. In Part VII, RAMs Packaging Requirements, we discuss the standard and general packaging requirements, Type A requirements for normal transportation conditions, Type B requirements for accident

conditions and DOT/NRC authorized packages. In Parts VIII-X we focus primarily on the general and specific requirements of the regulations as pertaining to marking, labeling, placarding and shipping papers. In Part XI we discuss responsibilities and liabilities for noncompliance to the regulations, types of violations, how discovered, penalties and public image consequences.

SUMMARY

The DOT and the NRC are the lead agencies responsible for formulating, issuing and enforcing regulations pertaining to transportation of RAMs in the commerce of the United States. State and local governments also adopt or formulate similar regulations for the transportation of RAMs into and through their states and cities. Other political subgroups such as ICAO, IATA, and IAEA formulate regulations that affect RAMs shipments by air and shipments into and through foreign countries. The regulations are detailed and cover all modes of transportation and all transportation processes. The regulations are complex and difficult to understand, interpret and implement.

Violations of the regulations can result in civil and/or criminal penalties in the form of fines, imprisonment or both and may be assessed to a company and/or an individual. Violations are discovered through routine audits of shipper and carrier facilities, vehicles in transit or through transportation accidents. Transportation accidents involving violations can result in additional legal, financial, and public image consequences. Thus the transportation of RAMs is a serious business requiring trained and retrained individuals.

This course provides the employee with a broad overview of the transportation chain and a working knowledge of the applicable transportation regulations as they primarily relate to the shipper. The course informs the employee of Sandia's mode of operations that help insure compliance with the regulations and emphasizes the employees responsibilities and liabilities under the Hazardous Materials Transportation Act. The course provides training in the fundamentals of radiation, the principal hazards involved in transporting RAMs and basic health and safety practices to reduce hazards and risks to the transport worker in both normal and emergency operations.

Figure 1. RADIOACTIVE MATERIALS TRANSPORTATION COURSE OUTLINE

I. Introduction to RAMs Transportation
II. General Hazardous Materials Requirements
III. Radiation Fundamentals
IV. RAMs Transport Hazards
V. Theory and Principals of Controlling RAMs Transport Hazards
VI. Identification and Classification of RAMs
VII. RAMs Packaging Requirements
VIII. Marking and Labeling Requirements for RAMs
IX. Placarding Requirements for RAMs
X. Shipping Paper Requirements for RAMs
XI. Violations/Penalties

POSSIBLE APPLICATION OF FUZZY SET THEORY IN RISK ASSESSMENT, SUBJECTIVE PERCEPTION AND PUBLIC ATTITUDE STUDY ON NUCLEAR ENERGY

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I. INTRODUCTION

There are many types of uncertainties involved in risk analysis or risk assessment where biological or medical/societal or humanistic systems are concerned. However, they may be divided into two major categories: the uncertainty due to randomness and that due to fuzziness. Conventional methods of treating the uncertainty are to apply statistical methods of estimation which are, in turn, based upon the concept of probability. Even in the case where the source of uncertainty is of non-statistical nature, formal application of statistical methods of analysis is often done to deal quantitatively with the uncertainty by tacitly accepting the premise that uncertainty - whatever its nature - can be equated with randomness. Most of the works on risk analysis or risk assessment have been done using such methods. The term "probabilistic" is sometimes applied to data even in cases when it is not applicable. The results are often expressed in such terms as "very low probability", while it is not sure whether they should be called "probability" or "possibility". For the analysis of such events, the concept of Fuzzy Set may be applicable, because some uncertainties are accepted as uncertain with introduction of a membership function in the theory of Fuzzy Set. Instead of "0 or 1" or "true or false" of the non-fuzzy binary logic, any intermediate values could be assumed for membership function in Fuzzy Set. The uncertainties are not necessarily probabilistic.

It has been known that much of our real world is more or less fuzzy and a variety of ambiguities are found in our daily conversation. The concept of fuzzy sets seem to be included implicitly or explicitly even in some scientific papers in the past. In the target theory presented by Nishiwaki at the Induced Mutation Session of the International Symposium on Genetics held in Tokyo, 1956, the biological system of a living cell is compared to a complex parallel and series circuit of switches which consist of gene-enzyme system. When such a switching circuit system is irradiated, the switches consisting of gene and enzyme are considered to be disturbed or cut off randomly. The action field of radiation is assumed to vary depending on the quality and type of radiation and the intra-cellular biochemical milieu. When the action field is large with the high LET radiation, multiple switches may be simultaneously destroyed by a single hit and a single hit curve with an apparently large target size may be obtained. When the action field is small with the low LET radiation under anoxic condition, multiple number of independent hits on a multiple number of target switches may be required to produce an

observable effect, and a sigmoid dose-survival curve with a multiple number of small targets or a complex dose-response curve may be observed. In other words, the size of the action field is considered to vary depending on the radiation and the environment and therefore the apparent size of the target may be considered flexible and the boundary more or less fuzzy. The term "fuzzy set" was first proposed by Zadeh in 1965, to describe various types of ambiguity or fuzziness. Some examples of uncertainties associated with frequency perception and judgement on the use of nuclear energy are given in Fig. 1 & 2. The term risk often refers to different types of undesirable effects associated with various activities and its meaning is generally fuzzy. Since the notion of risk in subjective perception or public attitude study must be conceived in terms of the interaction between the object (environment, activity, thechnology,etc.) and the subject (individual, group, society, etc.), a concrete definition of risk must be provided with the specifications of the type "of what", "to whom" and "for what". Presence of the subject in the risk concept is an essential source of fuzziness, because the same ojbect may be differently risky to different subjects who are in different positions and who have different amounts of information, different degrees and types of perception and preference with respect to the object. In this sense, risk is a notion which is not of absolute or objective nature but rather of relative or subjective nature, and therefore, the concepts of fuzzy sets or fuzzy logics seem to be adequate in order to treat the problem associated with risk properly.

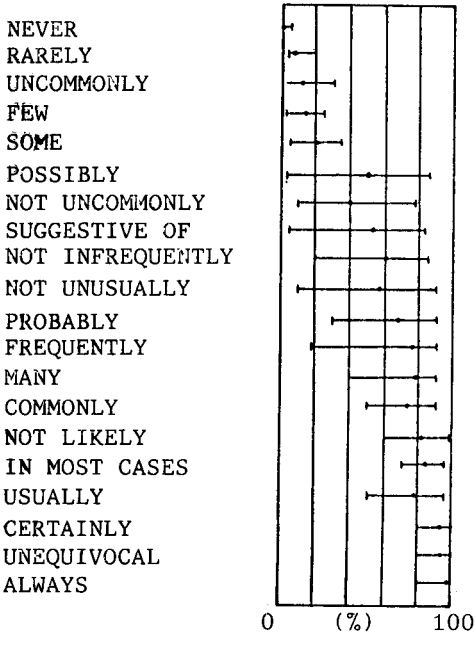

Fig. 1 FREQUENCY PERCEPTION
National Institute of Radiological Sciences (25 scientists)

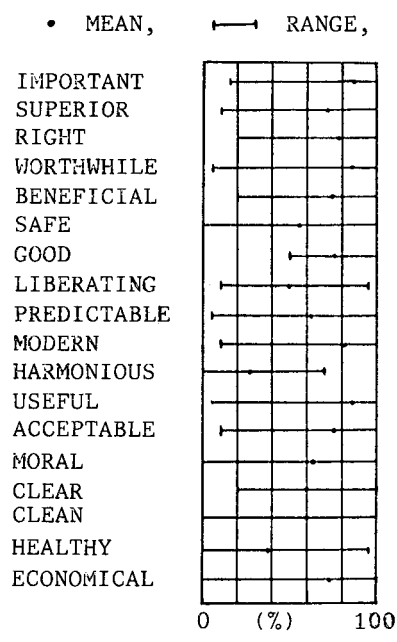

Fig. 2 JUDGEMENT ON THE USE OF NUCLEAR ENERGY
Kinki University (40 students)

II. FUZZY SET AND FUZZY MEASURE

According to Kruse (1983), mathematical concept about fuzziness which is a modality of uncertainty related to the subjective fuzziness of human being may be classified into two categories, one is the fuzzy set proposed by Zadeh (1965) and the other the fuzzy measure by Sugeno (1972). The fuzzy set deals with the relation between non-fuzzy element x and fuzzy subset $\tilde{A}$. On the other hand, the fuzzy measure deals with the relation between fuzzy element $\tilde{x}$ and the non-fuzzy subset B. The fuzzy measure approach generalizes probability measures by dropping the addity property and replacing monotonicity. The possibility measure (Zadeh 1978), belief measure, plausibility measure (Shafer 1976), etc. may be considered a kind of fuzzy measure. As a special form, g_λ measure has the following characteristics:

$$A \cap B = \phi \Rightarrow g_\lambda(A \cup B) = g_\lambda(A) + g_\lambda(B) + \lambda g_\lambda(A) g_\lambda(B), \quad -1 < \lambda < \infty$$

When λ=0, it coincides with the probability. The fuzzy integral is a concept of integral using fuzzy measure and may be considered an extension of Lebesque integral. It has a meaning of evaluation of objects by human sujectivity measure. The non-fuzzy linear model can be used only when the attributes are independent. The weighting factor in the linear model does not have such meaning as the grade of importance in the fuzzy model. Regarding the independence, it must be noted that even if physically independent,it may not be considered independent subjectively by the evaluator. In case of fuzzy measure, the assumption of independence is not required.

III. SAFETY ASSESSMENT AND FUZZY DIAGNOSIS

Failure diagnosis is essential for safety assessment of the plants. In case the plant falls in an abnormal condition, it may be desirable to have a device which could diagnose troubles automatically and identify causes of abnormal occurrences in order to prevent serious accidents or danger. However, with the automatic diagnosis devices which are currently employed, there is a possibility of serious misdiagnosis, partly because the logic of diagnosis is itself limited only to the troubles foreseen by man; and partly because a complete detection of all possible abnormalities may not always be possible, whether it be human being or automatic device, as long as a rigid non-fuzzy logic is used. In order to avoid these difficulties, an attempt is made by Terano, et al (1977) to introduce fuzziness in the logic of diagnosis of engine troubles to stimulate human judgement of operators. A more elastic model may be made by modelling the trouble causes and the expression of symptoms with fuzzy sets as a basis of diagnosis and by expressing these relations with fuzzy logics taking into consideration human knowledge and experiences of experts.

IV. PUBLIC ATTITUDE STUDY

For public attitude study, the following model is used. According to Fishbein the attitude A_o toward an object or event may be expressed by the summation of $(b_i \times e_i)$ where b_i = the strength of his belief about object "0", i.e. the subjective probability that "0" is related to some attribute i, e_i = the subject's evaluation of attribute i. However the data obtained in this type of study may be

considered more or less fuzzy. In the fuzzy set model, it may be expressed by the conditional proposition and its truth value: "if A, then B" is R, where $\tilde{A}$, $\tilde{B}$, $\tilde{R}$ are fuzzy sets: R is called linguistic truth value (Fig.3). If we assume A is the cause and B the result, R may be interpreted as a strength of belief about the implication. In the above relation, A_o may correspond to B, e_i to A and b_i to R. As a fuzzy model for the conditional proposition, the following possibilities may be considered: (1) Conditional Proposition and Truth Value Model: $A \rightarrow B$, R; (2) Fuzzy Relation Model: $B = A \circ R$; (3) Fuzzy Integral Model: $B = \int \sigma \cdot a$ where σ is conditional fuzzy measure, a is belief measure; (4) Fuzzy Number Equation: $\tilde{B} = \tilde{A} \times \tilde{R}$. In (4), normal mathematical equation may be used with fuzzy numbers for variables. Another effective communication tool for understanding multi-attribute problems would be structural model such as ISM or DEMATEL.

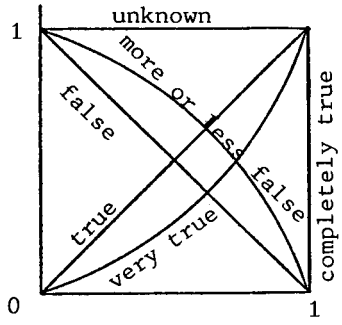

Fig. 3 - Some representative examples of membership functions for liguistic truth values. The truth values expressed with natural languages such as very true, more or less false, etc., are called Linguistic Truth Values (LTV), which may be expressed as fuzzy subset in Truth Value Space [0,1]. In this case, "true" in classical logic corresponds to "completely true" in the diagram. With the fuzzy set, even the qualitative rules expressed in natural languages could be modelled. When the data are more or less fuzzy, the quantitative model based on non-fuzzy logic may not always be the best.

V. CONCLUSION

In risk assessment, subjective perception or public attitude studies, we encounter a variety of sources of uncertainties which are due to ambiguity in our cognition or perception of objects. For systematic treatment of this type of uncertainty, the concepts of fuzzy sets could be applied to construct system models, which may take into consideration such an ambiguity.

References: L.A. Zadeh: Inform. Contr. 8, 338 (1965), Fuzzy Sets Syst., 1-1,3/28 (1978) North Holland; G. Shafer: A Mathematical Theory of Evidence, Princeton Univ.Press (1976); M. Sugeno: Keisoku to Seigyo (Measurement and Control) 8, 218 (1972), 9, 361 (1974), 22, 171, 380, 454, 554 (1983) Keisoku Jido Seigyo Gakkai (in Japanese); T. Terano: Summary of Papers on General Fuzzy Problems, Nos. 1-7 (1975-1981), Working Group on Fuzzy Sets, Tokyo Inst. Technology. Aimai Kogaku (Fuzzy Engineering) ed. T. Terano, 1982, Kodansha Tokyo (in Japanese); R. Kruse; Fuzzy Sets Systems 10, 309 (1983), North Holland; Y. Nishiwaki: Proc. Intl. Genetics Sym. 271 (1956) Tokyo, Biophysical Interpretation of Biol. Actions of Radiations, Monogr. Musashi Inst. Tech., (1960) Tokyo, J. Rad. Research 21, 22 (1961) Chiba, Xth Reg. Congr. IRPA, Avignon (1982), XIth Reg. Congr. IRPA, Vienna (1983); M. Fishbein: Belief, Attitude, Intention and Behavior, Addison-Wesley (1975); Y. Tsukamoto: Fuzzy Logic, Design and Control, Doct. Dessert., Tokyo Inst. Technology (1979)

IRENE H. ALLEN et al VS UNITED STATES OF AMERICA
A LEGAL CHALLENGE TO EARLY RADIATION PROTECTION PRACTICES

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INTRODUCTION

Since the conclusion of the United States (U.S.) continental nuclear weapons test series in early June, 1953, there have been many investigations of alleged illnesses, diseases and other consequences from the radioactive fallout entering the atmosphere from this series of 11 nuclear weapon detonations. These investigations have been conducted by agencies of the federal government, the U.S. Congress, state governments, universities and other institutions. The allegations claimed damage to animals, specifically horses and sheep, as well as members of the general public who lived in the vicinity of the test site.

The purpose of this paper is to discuss briefly the circumstances surrounding the one group of claimants which has reached the trial stage in the U.S. District Court in Salt Lake City, Utah as the case of Irene H. Allen et al. as the plaintiff versus the United States of America as defendant (1). Initially the legal action was brought by 1,192 people. Twenty-four individuals were selected to represent the larger group and to bring the number to a manageable level. It is estimated that the total amount of damages being sought by all 1,192 plaintiffs will exceed U.S. $2,500,000,000.

DISCUSSION

After nearly three years of legal preparation and maneuvering, the actual trial began September 13, 1982. It continued for eight weeks, then recessed for one month followed by the presentation of final arguments by the lawyers for each side which concluded on December 19, 1982. During this period the judge heard more than 100 witnesses, either in person or by deposition, and received more than 7,000 pages of testimony and 2,000 exhibits. Though the judge has not reached a decision at this time (September 30, 1983), it is expected that his verdict and its major conclusions, recommendations, and implications will be included in the oral presentation at the 6th International Congress of IRPA.

It has been estimated that the continental nuclear weapons tests in 1953 contributed approximately 80% of the regional radioactive fallout from all such tests which began in 1951 and which continue today. Thus, in terms of radiation exposure, the other several hundred atmospheric detonations are, or should be, of little consequence in this discussion. As a point of further simplification, the plaintiffs in this case are or were residents of southwestern Utah, an area 200-250 kilometers east of the weapons test site. The major population center is the city of St. George

with approximately 5,000 inhabitants and the home of a majority of the plaintiffs. The records indicate the total lifetime dose in St. George from all fallout is 3.7R (2) which is in terms of "effective biological dose" as given by Dunning (3). Recent measurements of residual Cs-137 and Pu-239,-240 have been used to reconstruct the external gamma exposure which corroborates this original dose estimate (4).

The radiation protection standards established by the U.S. Atomic Energy Commission for the 1953 series were the same as had been adopted unanimously by a special committee convened in 1951. The committee's views, reflected in Dunning (3), were that:

1. The external dose to non-participating inhabitants of radiation from gamma rays shall not exceed the accepted international permissible dose level of 300mR/week, which may be integrated over a maximum of 10 weeks.*

2. At a point of human habitation, the activity of radioactive particles in the atmosphere, averaged over a period of 24 hours, shall be limited to 100 microcuries per cubic meter of air (corresponding approximately to a ground level gamma intensity of 30mR per hour).

3. The 24-hour average radioactivity per cubic meter of air, due to suspended particles having diameters in the range of 0 microns to 5.0 microns, shall not exceed 1/100th of the above; nor is it desirable that any individual particle in this size range have an activity greater than 10^{-2} microcuries calculated 4 hours after the blast.

   *Subsequently modified to allow integration over the duration of the test series which in 1953 was 13 weeks or 3.9R.

Field measurements and collections of specific environmental samples for subsequent laboratory analysis were used to evaluate pertinent exposures to people in the off-site area. Calibrated portable survey instruments were used to monitor gamma exposures three feet above the terrain. Corrected instrument readings and knowledge of time of fallout were used to estimate whole body exposures to penetrating radiation (gamma) at places people lived or frequented (5).

Particulates were collected from the air by passing a known volume of air through a dust filter. These filter samples were then analyzed in the laboratory by determination of gross beta activity which was expressed as microcuries per unit volume of air, corrected for decay, and averaged over the specified sampling period. In addition, cascade impactors (four-stage) with an added membrane filter as a fifth-stage were used to collect air particulates in size ranges which could be determined by laboratory counting and calculational procedures (6).

Other types of measurements were made in the field and other environmental samples were collected for laboratory analysis but the external penetrating radiation and the air sample measurements were the primary basis for assessing the radiation health and safety program and the determination of compliance with applicable

radiation protection standards. The equipment and procedures used in the off-site health and safety program were state-of-the-art (5,6).

The questions raised in the Allen case, some three decades after the experience, are exemplified by these selected examples: What was the commitment to health and safety? Was the health and safety program adequate and effective? Were the personnel properly trained and competent? What equipment was used? What procedures were followed? When, where, and how were measurements made? How were the data used? Who interpreted the results? What were the standards? How were the standards used? What was the understanding of the effects and especially the health effects of fallout? What and how were warnings given to the offsite population? Why weren't special standards used to protect children? Where was the ultimate responsibility for health and safety? (7).

These and other technical questions were responded to during the course of the trial. Many of them speak to the core of a health and safety program established for the protection of the public from fallout in the vicinity of the test site. Some are philosophic and thus open ended. Nonetheless, each is important in the context of being asked and answered and thus contributing information to the trial record which is what is being evaluated and judged in the Allen case.

The lawyers for the plaintiff presented several witnesses who claimed ailments ascribed to radiation but without any relationship to exposure or cause and effect. Among the claims were diabetes, deterioration of eyesight, miscarriage, allergy, and a list of cancers - bladder, brain, prostrate, uterine, rectal, breast, abdominal, etc. Added to these were undiagnosed, non-specific complaints described as "foot problems", "nerves", "sleeplessness", "multiple-hemorrhages", or "constant pain" (7). Apparently it was the intent of plaintiff's counsel to produce an effect and if not consistent with the radiation exposure data of the defendant, then the logical conclusion must be that defendant's data are in error! One cannot minimize the impact of this technique on the thought processes of an individual, such as a judge, who is not familiar with radioactivity, radiation, fallout mechanisms, theories of biological effects, and the principles of radiation protection.

This lack of familiarity and understanding applies equally to the legal profession, at least as demonstrated by the lawyers involved on both sides of this case. Gross misrepresentation of fact was left unanswered simply because counsel did not have the knowledge necessary to carry out a successful challenge.

The health physicist needs to become a more active participant with his legal counsel to avoid these situations or to utilize their opportunities. Similarly, they should assist counsel in developing questions for the witnesses. The authors, as witnesses, found themselves unable to pursue important points to reasonable conclusions because the "right" questions were not asked. A witness does not have the luxury of using the witness box for free expression of professional opinions unless the dialog between witness and lawyer is well coordinated.

The Allen case also suffered from "loss of memory", i.e., events

which preceded the trial by 30 years were expected to be recalled as though they occurred only yesterday - and under oath. The actual sequence of events is impossible to reconstruct in intricate detail. One person's memory can easily contradict another and each individual is equally certain of his position. Obviously, the importance of thorough record keeping over a prolonged period of time cannot be emphasized too strongly. But even in this case, one witness for the plaintiff had his facts completely confused and denounced his written record of 30 years previous as being the work of others seeking to discredit him or to hide the "real story"(7).

There are always those who are willing and anxious to make any event involving radiation spectacular, mysterious, and alarming. It is quite satisfying to their egos to find fault with professional colleagues and to make government programs appear negligent. They offer the perfect program, given perfect hindsight. Several examples having these chracteristics appear in the testimony of the Allen case (7).

In legal language, the Allen case is a civil case, as opposed to a criminal case. Accordingly, the burden of proof is on the plaintiff and by a preponderance of evidence. It is the authors' opinion that counsel for the plaintiff failed to meet this requirement. However, the authors are also of the opinion that the verdict will turn on "effect" not "cause".

Do the authors and the judge agree? The oral presentation will answer that question by analyzing the judge's decision as to its "cause" and potential "effect" on radiation protection practices.

REFERENCES

1. In the United States District Court for the District of Utah, Central Division, Irene H. Allen, et al., Plaintiff vs. United States of America, Defendant; Case No. C79-515.

2. CIC No. 671, U.S. Department of Energy Coordination and Information Center, Post Office Box 14400, Las Vegas, NV.

3. Dunning, G.M., "Discussion of Radiological Safety Criteria and Procedures for Public Protection at the Nevada Test Site and Radiations From Fallout and Their Effects", in The Nature of Radioactive Fallout and Its Effects on Man, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the U.S., Part 1, GPO, Washington, D.C., 1957.

4. Beck, Harold L., Krey, Philip W., and Miller, Kevin M., "External Radiation Exposure of the Population of Utah from Nevada Weapons Tests", Abstract of paper presented at the 27th Annual Meeting of the Health Physics Society, Las Vegas, NV, June 27-July 1, 1982.

5. Report of Public Health Service Activities In The Off-Site Monitoring Program, Nevada Proving Ground - Spring 1953, U.S. Department of Health, Education and Welfare, Washington, D.C.

6. Skillern, C.P., Johnson, W.S., and Schulte, H.F., "Some Observations on Air Sampling Techniques Used at the Nevada Proving Ground", LA-1685, Los Alamos Scientific Laboratory, Los Alamos, NM, 1954.

7. See Reference 1, supra, Transcripts of Trial on file.

NUCLEAR REACTOR EMERGENCY EXERCISES AND DRILLS

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The nuclear facility operator in the event of an accident has the unequivocal responsibility for the initial assessment of the accident situation at the facility. He must predict the off-site conditions which can become significant within minutes or hours of the incident and whilst he has prime concern for the plant which he must place in a safe configuration, he must also implement personnel safety action and take action for local emergency response to be initiated. Those involved in the management of the emergency must make decisions as to whether or not countermeasures should be advised off-site.

Site personnel will have to re-enter previously evacuated places for saving lives, finding missing persons or for manipulating, repairing or recovering critical equipment or other articles. Contingency planning is, therefore, essential to reduce the consequences of an unlikely but major accident.

TRAINING

To ensure that emergency plans can be effective, provision must be made for periodic training of the emergency response staffs, including management, monitoring and support teams. Whilst most of the personnel will have the knowledge needed for their specialised emergency assignments, they may not be familiar with the different conditions imposed by the radiological characteristics of the accident, under which they must carry out their assigned duties. Initial instruction should clarify these differences and should also include appropriate information on the emergency organisation, plans, procedures, standards, standing orders etc.

Exercises and drills can be used as training for maintaining the proficiency level of the teams and they can also be used to test the adequacy of the plans, procedures, equipment, communications etc. Plans which were adequate when drawn up may have become ineffective due to changes in:-

1) The nature of the hazard
2) Key personnel
3) Communications
4) The population at risk

The following categories of personnel should participate in the training programme:-

1) Senior Management, including Public Relations Staff.
2) Supervisory and staff personnel of the emergency organisations.
3) Personnel responsible for accident assessment.
4) Radiological monitoring teams.
5) Advisory groups of experts.
6) Fire control teams and fire brigades.
7) Civil Defence or Emergency Services Organisations.
8) First aid and rescue teams.
9) Off-site emergency services.
10) Medical and health personnel.
11) Headquarters personnel.
12) Police units.

It is possible to involve members of the public in exercises but it is not recommended. Local government organisations and officials of emergency services should be notified in advance when exercises are planned. The public should be made aware that exercises may be conducted within a certain time period. All exercise communications should be clearly identified as being an exercise or drill since it is not unusual for emergency services messages to be overheard and reported to the media.

EXERCISES AND DRILLS

Training exercises usually relate to the functioning of the Emergency Response organisation as a whole, while training drills usually develop skills such as the operation of equipment.

It is not always desirable to exercise the whole of an emergency organisation. Top management of the emergency team will be primarily concerned with defining the strategy and can be kept fully occupied in any exercise. Staff at the tactical level may have little to do during long spells of an exercise and will learn best from drills. Lower level tasks tend to be 'discrete' well defined and are easy to drill in isolation compared with higher level tasks. The various emergency response teams should be trained in their assignments the operation of equipment and operations as a group.

At the strategic level a wide range of scenarios using "paper" or "table top" exercises can be used effectively and critically. For example, the scene is set and the exercise consists of questions for group discussion as in a seminar. This gives the participants an overview of the situation each being able to present his own particular problems. A brief summary of an exercise of this type is provided at the end of this paper.

Exercises and drills can be conducted on several levels:-

1) drills involving single teams in specific procedures and uses of equipment,
2) communications drills involving simulated responses by on-site and off-site personnel,
3) exercising of the entire on-site and off-site emergency response organisations,
4) off-site organisation exercises with or without the involvement of personnel of the nuclear facility,
5) exercises involving simulated activity on the part of the public.

It is important to decide on the aim of each exercise at the very onset of its planning and once decided upon, must be kept firmly in view. The choice lies between a test of the plan itself or the training of personnel. If the aim is to test the plan itself then it must be made clear that the exercise will <u>not</u> test the performance of individuals nor award praise or blame accordingly. Any exercise inevitably involves a degree of training which can conflict with the exercise aim, especially if this is too complex; a limited objective could be to test communications only or to evaluate personnel rescue.

The aim of the exercise will demand an hypothetical situation around which the exercise will be played. For example this may involve the release of radioactive material such as fission products and the mechanism postulated for the release will require considerable imagination, knowledge and judgement if it is to carry any degree of realism. Participants must believe that they are exercising a possible real situation. The magnitude of the release, weather,

time of day having been specified, the radiological consequences (radiation levels, contamination levels) have to be specified in considerable detail at various locations.

The exercise scenario can specify a number of situations which follow logically one from another, irrespective of players decisions and actions. Alternatively players may be free to make their own decisions and left to follow through the consequences. The limitations of time allocated for exercise play will detract from realism, a real emergency may last for several days and so many events have to be exercised in a few hours (three hours is a useful limit). The main impact of time limits is that decisions have to be taken quickly and based on incomplete information so are unrealistic.

A high degree of realism is desirable at all stages but some aspects cannot be easily simulated. Media and public response is difficult to simulate, attempts can be made using experienced exericse teams supported by Press Officers and Public Relations experts. The actual uncertainty which surrounds any real emergency in its early stages, combined with the inevitable conflicting reports and failures in communication are almost impossible to simulate.

DIRECTING STAFF

An emergency exercise should be controlled by Directing Staff who must take a broad view of the whole situation and must have the authority to ensure that unnecessary hazards are avoided in the exercise play.

The Senior Director must be given the authority to:-

1) initiate the exercise,
2) modify the exercise play,
3) terminate the exercise,
4) and to initiate correct action if a real emergency develops.

The Directing Staff should:-

1) observe all aspects of the exercise play, provide simulation of plant operation where relevant to exercise play.
2) Control radioactive contamination simulation and radiation fields simulation. Radioactive sources must be carefully recorded and recovered.
3) Restrict outside authorities action unless essential to the play. Authorities can report time, message received and proposed action. Special code words and phrases should be reserved for use in the exercise only.
4) Simulate damage and casualties realistically.
5) Collect the exercise timed sequence of events accurately.

A post exercise discussion between players and Directing Staff must be convened within a few hours of the end of the exercise to identify lessons learned and to highlight inadequaces in the plan.

The Directing Staff must be clear and concise when presenting their critique and exhibit a high degree of tact. This should be followed by a written report from the Directing Staff, detailing the lessons learned. Management must ensure that lessons are learned by the players and any shortcomings in emergency plans are made good. If these steps are not taken all the planning effort and time spent will have been wasted.

EXAMPLE OF A TABLE TOP EXERCISE

Aim

The aims of this table top exercise is

1. To evaluate the organisation for dealing with an accident to a pressurised water reactor.

2. To examine the problems which might arise following a reactor accident from the point of view of the Emergency Co-ordinator.

Note: Nuclear Regulatory Commission - Emergency Planning Final Regulations Federal Register Part VIII Tuesday 19 August 1980

Section 50.47 Emergency Plans a.(i) states

"No operating licence for a nuclear power reactor will be issued unless a finding is made by NRC at the state of on-site and off-site emergency preparedness provides reasonable assurances that adequate protective measures can and will be taken in the event of a radiological emergency".

Notes for Directing Staff

This exercise involves an accident to a pressurised water reactor at an imaginary location. For the purpose of this exercise the PWR develops a major primary coolant leak due to a failure in steam generator tubes and subsequent actions leads to a major accident and a release.

The players are required to deal with the situation which develops and play the role of Emergency Co-ordinator, working as a syndicate, preferably with 5 persons per syndicate. Each syndicate should choose a Chairman for each stage of the exercise. The Chairman should record the decisions reached and should present them, and if necessary defend them, at the final wash up.

A description of the site will be distributed before the exercise commences and a status report for 0600 (the start of the incident) will be distributed at the start of the exercise. The remainder of the exercise will be conducted in five sessions of 30 minutes. A sequence of events, a set of questions and a staff solution will be provided for each session. NUREG 0654 (FEMA) "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants", will be issued at least 1 for each syndicate. The exercise will combine with a plenary session to share lessons learned.

PREPARING A BOOK ON RADIATION FOR THE PEOPLE OF THE MARSHALL ISLANDS

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INTRODUCTION

Between 1946 and 1958 the United States conducted 23 atmospheric nuclear weapons tests at Bikini Atoll and 43 tests at Enewetak Atoll in the Northern Marshall Islands. The inhabitants of these atolls were relocated prior to the tests. In the 1960's, the people of Bikini and Enewetak began to explore the possibility of returning to their respective atolls. In 1968, following radiological surveys, the Bikini people were allowed to return to part of their atoll; however, in 1978 these people were again relocated because additional radiological surveys and monitoring of the people living at Bikini indicated higher than expected levels of radioactive cesium.

In the mid-1970's, the United States undertook a major effort to clean up debris and transuranic contaminated soil from Enewetak Atoll in preparation for possible resettlement. Additional radiological surveys were completed by scientists from Lawrence Livermore National Laboratory on these atolls and other islands and atolls in the Northern Marshalls. Information obtained by these surveys of air, water, and land included measurements of gamma radiation emanating from radionuclides in the soil of the atolls, and measurements of ^{90}Sr, ^{137}Cs, ^{60}Co, ^{239}Pu, and ^{241}Am in soil, water, and in plants and animals used for food. Based upon several possible defined living patterns for relevant islands in each atoll, the Livermore scientists calculated radiation doses that people might receive. The Department of Energy subsequently used these dose values to estimate risks of fatal cancer and genetic effects for populations living on these islands during the next 35 years.

Interest by the Marshallese in the results of these surveys was heightened by many events: the second relocation of the Bikini people; renewed interest by the Bikini and Enewetak people to resettle their atolls; the evolution of self-government in the Marshall Islands; claims initiated on behalf of certain populations in the Marshall Islands against the United States to compensate for loss of their land; and concern by other northern Marshall Island residents about current and future possible radiation-caused health effects.

In response to this interest, the United States offered to describe and explain to the Marshallese the results of the radiological surveys and the estimates of health risks. This was to be done at public meetings in the Marshall Islands. To aid in communicating this information, the Department of Energy prepared several bilingual books in Marshallese and English: one for Enewetak Atoll in 1979, one for Bikini Atoll in 1980, and one for other atolls in the Northern Marshall Islands in 1982.

The following describes the preparation of these three books and offers examples of the subject matter developed.

THE APPROACH

Recognizing that we knew little about the Marshallese language and culture, we sought guidance from a cultural anthropologist at the University of Hawaii, Pacific Studies Center, who had lived with and written about the Bikini people. He identified various problems we would face in trying to communicate technical information to the Marshallese. He also directed us to one of the few people qualified to assist in translating the information we wanted to communicate. This person, Mrs. Alice Buck, was an American who had spent many years living with and working among the people of the Pacific Islands, first as a daughter of a missionary and later as the wife of a minister working in the Marshall Islands. She was in the final phases of translating the Bible into Marshallese, was familiar with the language and culture of the Marshall Islands, and was known and respected by many Marshallese.

Mrs. Buck's first recommendation to us was that it would be not only helpful but necessary to include in this effort Marshallese informants who were fluent in both English and Marshallese. Since Mrs. Buck wanted both sexes and a spectrum of ages represented, she selected a young woman and a more mature gentleman to assist us. The woman was a teacher who had received university training in the Phillippines and who, among other things, taught English; the gentleman held a responsible position with a large American contractor at the Kwajalein Missile Range base.

Our initial discussions soon dramatized the breadth and depth of difficulties we found. It became readily apparent that it was first necessary for us to familiarize Mrs. Buck and the two Marshallese with the basic concepts of radiation physics and biology and with information on radiological assessments. We also learned in our preliminary discussions that we would need to provide basic information on the metabolism, deposition, and excretion of radionuclides following inhalation and ingestion.

ORGANIZATION, TRANSLATION, AND PUBLICATION PROBLEMS

The usual difficulties with ambiguity and loss of tone or style that one encounters in translating from one language to another were considerably heightened by the vast differences between English and Marshallese. Moreover, since Marshallese has few technical or scientific terms, it was not possible to translate directly from a scientific English text. Conceptual comprehension was particularly difficult--the fact that gamma radiation can go "through" a person or material and not leave a hole was astounding. Probability and numbers smaller than one or greater than a few hundred are not only foreign to Marshallese culture but are difficult to accommodate linguistically. Therefore, risk and probability could only be approximated, and we attempted to avoid the use of numbers whenever possible, except in explaining dose and risk where there was no alternative. Only through extensive and repeated discussions and the use of examples, such as playing cards and dice, was a base of mutual understanding and an acceptable, if somewhat nontechnical, text established.

Certain peculiarities of Marshallese grammar and syntax only added to the difficulties. Marshallese lacks, for example, a conditional mood or tense and has no passive voice. Its spelling is also considerably fluid. Because of these and other language limitations and the general levels of educational achievement among the Marshallese, descriptions of technical topics had to be limited to an elementary level.

Word meanings also required extensive discussion. Many technical terms (e.g., atom, radioactive atom, the several types of radiation, cells, various elements) had no Marshallese counterparts; in these instances we were compelled to retain English words. Following publication and distribution of the first book in 1979 (i.e., The Enewetak Atoll Today), we were asked to provide a glossary of such English words. This worthwhile suggestion was incorporated in the second and third books published in 1980 and 1982. The discussions of metabolism, deposition, and excretion of radionuclides were also expanded in the later books at the request of a Marshallese teacher who had read the first book.

Each word, phrase, sentence, paragraph, and concept had to be discussed and reviewed numerous times among the authors and translators to ensure that the intended meaning was clearly understood and translatable without connotations that might be offensive or humorous or that might be negatively associated with beliefs or cultural characteristics.

We recognized immediately that illustrations, photographs, and color would need to be used to convey the information we hoped to include. Radioactive elements were color-coded throughout, and color gradations and intensities were employed to denote relative amounts of radioactive materials. (Other graphic techniques for conveying this information included dot density and direct numerical displays on maps.) These techniques helped to depict (1) the concept of radioactive decay; (2) absorption of types of radiation; (3) the way in which nuclear detonations contaminated the atolls; (4) the presence of radioactive materials in soil and how they enter food and reach man; (5) the movement of radionuclides in the body. Other illustrations showed division of normal and cancerous cells, health effects of radiation, and examples of how radiation is measured in the body and on the atolls.

This information preceded a discussion of the levels of contamination remaining on the several atolls. In particular, because of the intense desire of the Bikini and Enewetak people to return to their islands, dose and risk estimates were calculated for various combinations of time spent on residence islands, time spent on other islands, and sources of food supply (food grown on contaminated islands and food grown elsewhere). These various combinations in part were illustrated graphically and were accompanied by calculated estimates of the numbers of radiation-induced cancers and genetic defects among each population of each atoll.

Finally, the actual publication of the book required considerably more than the usual effort. The book's design had to ensure that the connection between text and related illustrations was apparent and that the text of each of the two languages was in juxtaposition. Waterproof paper and stainless steel

staples had to be located. In order to distinguish color shades and differences adequately, four-color printing was required. To ensure that illustrations and colors were properly registered and that last-minute corrections were executed, each word of text and each color was reviewed on proofs prior to printing. A press inspection of the final product was also conducted.

COMMENTS AND SUMMARY

These explanatory books, including the estimates of risk, have, we hope, enabled the people of the Marshall Islands, together with their government, to understand more about the radioactive contamination that exists on some of the atolls and what this means in terms of potential hazards. Unfortunately there has not been an opportunity to evaluate the effectiveness of these books in that regard. The books have also been of considerable interest to other parties including the U.S. Congress, other U.S. agencies, as well as organizations and institutions in the private sector. While they might not be directly applicable to the needs of other countries and while it might be necessary to improve certain portions of the books (each successive book was extensively revised), they might serve as models upon which other books could be written to serve comparable needs of people elsewhere.

The lessons learned in attempting to communicate this type of information to people whose language is technically limited or lacking linguistically or semantically might be summarized as follows:

1. People indigenous to the culture must be directly involved in the translation.

2. The authors and translators must work together and interact extensively; communication cannot be left to the written word.

3. Illustrations carefully designed to amplify concepts and information presented in the text require the skills of a competent graphics illustrator.

4. It is important that a graphics illustrator and technical editor be present in many if not all of the discussions.

5. Because literally hundreds of sentences and paragraphs will be written, rewritten, translated, and retranslated, it is imperative that a system for control of paper flow be established before translation is begun.

6. The authors, illustrator, translators, and editor must work closely with the printer to ensure accuracy of text and use of color.

7. The process of translating and rewriting is extremely time-consuming and should be carried out in several phases of one to three weeks each over a period of months.

4. CONCLUSION

These results illustrate the interest of a new approach of public risk perception ; it shows the multidimensional aspects of risk perception taken at its psycho-social level. If a synthetic indicator of fear and concern is conceivable, for many activities it has been demonstrated that the social signification of declaring they are risky is by far ambiguous. For example : if the denunciation of the dangers of narcotics is certainly a mark of anxiety, it also expresses a feature of conservatism.

To conclude, it is useful to follow simultaneously in the wider public the nuclear risk perception and the national debate on nuclear energy. For some people, risk is used as an argument, favourable or not ; but for many others, it works as a substitute to explain a more global anxiety. So for us, the perceived nuclear risk has no clear link with detriment ; but as social phenomenon, it takes place obviously in communication flows.

5. REFERENCES

/1/ J.P. PAGES, G. MORLAT, E. STEMMELEN
Structures de l'opinion publique et débat nucléaire dans la Société française contemporaine.
Revue Générale Nucléaire, n° 2, p. 140-149, 1982.

/2/ J.P. PAGES, G. MORLAT, E. STEMMELEN
Variété des points de vue et perception des risques.
Société Française de Radioprotection, Congrès annuel 1982, Avignon 1982.

CONCEPTS, QUANTITIES AND UNITS FOR NON-IONIZING RADIATION PROTECTION

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Protection against non-ionizing radiation (NIR) is the subject of an increasing interest, but the use of very different concepts, depending on the type of radiation or application, makes it rather difficult to compile studies and the data obtained in an uniform way. Also, the legal application and public appreciation of the concepts of radiation protection are hampered by the lack of uniformity in this field.
The International Non-Ionizing Radiation Committee (INIRC) of the IRPA set up a working group** with the task to prepare a review of concepts, quantities, units and terminology for NIR protection as a service to workers in this discipline. The present paper summarises the results obtained by this working group which are to be published fully in a forthcoming report. The main aim of the report is to provide an inventory of concepts, quantities, units and terminology currently used for purposes of NIR protection. Furthermore a systematic classification and comparison of these quantities is given, and in particular the concepts used to quantify exposure limitation and radiation protection standards are summarised and discussed.

The inventory of quantities and units is following as closely as possible the recommendations of the ISO Standards Handbook. The material is subdivided in four sections:
- periodic and radiation phenomena;
- electromagnetic radiation and fields;
- optical radiation;
- ultrasound.

The classification and comparison of quantities is done according to three criteria:
1. The physical characteristics of the radiation field, taking sources and receptors of radiation into account (radiometric quantities).
2. The interaction of NIR with matter.
3. Quantities adequate for the specification of exposure of biological objects to NIR (dosimetric quantities).

Radiometric quantities can be grouped into seven generic terms: energy; energy per time; energy per area; energy per volume; energy per time and solid angle; and energy per time, area and solid angle. This grouping can be compared for various NIR radiation and the ionizing radiation.
The quantities energy and its time derivative are most frequently applied to characterise a source.
The radiation field at any given point can be characterised by the spatial energy density, and by defining the energy transport through space (energy per area and energy per time and area) at that point.
To indicate the existing diversity, consider the term energy per time and area. It can be used in three ways:

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a. radiant power per unit cross section of a small sphere (energy fluence rate, energy flux density, radiant flux density, power surface density, acoustic intensity);
b. radiant power per unit area of the source surface ("radiant exitance" for optical radiation); and
c. radiant power per unit area of a receiving surface (optical "irradiance").

Sources of radiation with an angular dependence can be characterized by the power per solid angle ("radiant intensity") and power per area and solid angle ("radiance").
Great care has been taken to specify the orientation of the surface of a source or receptor in relation to the radiation field. Three approaches are being used and must be carefully distinguished in practical applications:

- Area element perpendicular to the direction of radiation: this approach is used e.g for the definition of optical radiance and for energy radiance in ionizing radiation.
- Cross-sectional area of a sphere: this is the reference surface in the definition of e.g. energy fluence (for ionizing radiation), of power surface density (for the electromagnetic radiation), and of acoustic intensity (for ultrasound).
- Area element not perpendicular to radiation: this reference surface is used e.g. for the definition of irradiation and radiant exposure (da = area element of the receptor surface) since for the optical radiations, the penetration depth in matter is generally very small. Therefore, the degree of biological effect is often determined by the amount of radiant energy incident on a given surface area.

For increasing wavelenghts, energy, power and derived radiometric quantities are concepts which are becoming less satisfactory. In the near field, which becomes more important under such conditions, the inverse square law does not apply, and the ratio of the electric to the magnetic field is not fixed. This problem becomes important for radiofrequencies and extremely low frequencies. Under such conditions, a sufficiently general specification of the electric and magnetic fields, $\underline{E}$, $\underline{H}$, of the electromagnetic wave at each location of interest is required.

Concepts for the characterisation of interaction processes, such as scattering, attenuation, transmission, reflection, refraction and diffraction are well known from the physical theories describing radiation phenomena in media and at boundaries. Analogies between ionizing and non-ionizing radiation are limited, due to the rather different nature of underlying physical interaction processes.

In the broad sense, the term "dosimetry" is used to quantify the parameters necessary for the characterisation of an exposure.
In the field of NIR, different characteristics of physical interaction mechanisms, measurement conditions and techniques, as well as differences in and poorer knowledge of biological response mechanisms have led to a diversity of quantities used for the specification of exposure.

It can be stated in general that, across the NIR spectrum, the temporal characteristics of exposure are of critical importance, and the contribution of ambient factors such as temperature must be taken into account. For example, if the emphasis is on the limitation of thermal effects, in the RF region many data at present seem to support introduction of exposure limits which are based on the instantaneous rate of energy depostion, as opposed to the case of ionizing radiation (see NCRP report no.67). The acoustic and ultrasonic fields represent another example where quantities other than cumulative energy deposition may be useful to specify exposure limits.
A compilation of the quantities used for expressing limits of exposure is given in Table 1. For a detailed discussion of the table, the reader is referred to the forthcoming report of the working group.

In conclusion, a well-developed and internationally accepted terminology exists for the classical fields of electromagnetism, optics and acoustics. It can be utilized for the purposes of NIR protection, under consideration of the practical circumstances and biological constraints.
There exists already a considerable degree of harmonization of radiometric physical concepts, with some remaining differences in the names of quantities.
Interaction coefficients reflect a higher degree of non-uniformity and more work should be done to clarify the picture.
Exposure quantities (Table 1) show a great diversity and non-uniformity. It seems at the moment that the introduction of unified dosimetric concepts is not practicable and desirable for all types of NIR. Coexistence of different quantities for different frequency ranges and purposes appears as a necessity, both for intrinsic and practical reasons, at least at the present time. However existing possibilities of introducing a greater uniformity require further serious examination.

References:

ISO Standard Handbook 2, 1979.
Units and measurements.

ICRU Report 33: Radiation Quantities and Units.
Issued 15 April 1980.
International Commission on Radiation Units and Measurements,
7910 Woodmont Avenue, Washington D.C. 20014, U.S.A.

Radiofrequency electromagnetic fields. Properties, quantities and units, biophysical interaction, and measurements.
Recommendations of the National Council on Radiation Protection and Measurements.
NCRP Report no. 67: Washington D.C., 1981.

Table 1.
Selected quantities for the specification of limits of exposure to non-ionizing radiation (dosimetric quantities).

	GENERIC TERMS					
Radiation type	Field parameters	Energy/area	Energy / time/area	Energy / time-area-solid angle	Energy / mass	Energy / mass-time
Ionizing radiation					Absorbed dose (Gy)	Absorbed dose rate (Gy/s)
					Dose equivalent (Sv)	Dose equivalent rate (Sv/s)
Radiofrequencies	Effective electric field strength (V/m) Effective magnetic field strength (A/m)		Power surface density (W/m^2)		Specific absorption (J/kg)	Specific absorption rate (W/kg)
ELF	Electric field strength (V/m) Magnetic field strength (A/m)					
Optical		Radiant exposure (J/m^2)	Irradiance (W/m^2)	Radiance ($W.m^{-2}.sr^{-1}$)		
			Effective irradiance (W/m^2)			
Ultrasound			Acoustic intensity (W/m^2)			
Ultrasound (airborne)			Acoustic pressure level (dB)			

PROBLEMS WITH REGULATING RADIOFREQUENCY (RF) RADIATION EXPOSURE.

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Many concerns have been raised, both by the scientific community and in the press, that quite divergent opinions exist in industrialized countries on the nature and degree of hazard from exposure to RF radiation. This divergence of opinion is reflected in widely differing national exposure standards. In response to this, there has been intense activity, both at the international and national level, on the evaluation of biological effects literature and assessment of health hazards of human exposure to RF radiation. It has been only recently that our understanding of some of the factors influencing RF absorption in biological systems has reached a stage that it can be usefully employed in the development of human exposure limits.

IRPA's International Non-Ionizing Radiation Committee (INIRC) recognised the problems associated with RF exposure and, in collaboration with the United Nations Environment Programme and the World Health Organization, developed a health criteria document on radiofrequency radiation (1). This document incorporates a review of RF sources, characteristics of RF fields, measurement instruments, RF energy absorption in biological systems, and a thorough review of reports on biological effects in animals and health effects in man. An outline of existing national and international standards and their rationale is also included. The criteria document provided a scientific basis for the development of an exposure standard for the IRPA/INIRC.

The INIRC has composed interim guidelines on limits of exposure to electromagnetic fields in the frequency range from 100 kHz to 300 GHz (2). Following approval by the IRPA Executive Council, a draft was distributed to Member Societies of IRPA, and to various Institutions and individual scientists for comments. Many helpful comments and criticisms were taken into account to form the present guidelines. The Committee recognized that when exposure limits are drafted, various value judgements are made. The validity of scientific reports has to be considered, and extrapolations from animal experiments to effects in humans have to be made. Cost versus benefit analyses are necessary, including the economic impact of such controls. However, the limits in these guidelines were based on scientific data and no consideration was given to economic impact or other non-scientific priorities.

In summerized form the guidelines state that occupational exposure to RF radiation at frequencies at or above 10 MHz should not exceed a specific absorption rate (SAR) of 0.4 W/kg when averaged over the whole body in any 6 minute period, or 4 W/kg when averaged over any one gram of tissue in any 6 minute period. For RF radiation exposure at frequencies below 10 MHz, the levels of unperturbed root mean square (RMS) electric or magnetic field strength should not exceed the values given in Table 1.

The limits of occupational exposure given in Table 1 for the frequencies between 10-300,000 MHz are the working limits derived from the SAR value of 0.4 W/kg. They represent a practical approximation of the incident plane wave power density needed to produce the whole body average specific absorption rate of 0.4 W/kg. These limits apply to exposure from either continuous or modulated electromagnetic fields from one or more sources, averaged over any 6 minute period during the working day (8h per 24h).

Although little information is presently available on the relationship between biological effects and peak values of pulsed RF fields, it is suggested that the instantaneous peak values for all frequencies not exceed 100 times the limits in Table 1 for the frequency concerned.

Table 1: Limits for whole and partial body occupational RF exposure

Frequency Range	Unperturbed RMS Electric Field Strength	Unperturbed RMS Magnetic Field Strength	Equivalent Plane Wave Power Density	
MHz	V/m	A/m	W/m^2	mW/cm^2
0.1-1	194	0.51	*100	*10
1-10	$194/f^{1/2}$	$0.51/f^{1/2}$	*100/f	*10/f
10-400	61	0.16	10	1
400-2000	$3f^{1/2}$	$0.008f^{1/2}$	f/40	f/400
2000-300,000	137	0.36	50	5

* These values are not for determining compliance.

Note: (i) f = frequency in MHz

(ii) The limits in the frequency ranges above 10 MHz may be exceeded for specific applications provided the SAR remains below 0.4W/kg when averaged over the whole body and below 4W/kg when averaged over any one gram of tissue. The limits for frequencies at or below 10 MHz may be exceeded (up to 615 V/m or 1.6 A/m) provided workers take the necessary precautions to prevent potentially severe RF burns.

When simultaneous exposure occurs from radiations emitted from sources operating at different frequencies, the exposure should be measured at each frequency and expressed as a fraction of the power density limit or the square of the electric or magnetic field limit for each frequency range (in Table 1). Then the sum of these fractions should not exceed unity. Exposure to radiofrequency radiation emitted from low power devices, such as citizen's band radios, land mobile and marine transmitters, and walkie-talkies can be excluded from consideration in assessing compliance with the prescribed limits provided the radiofrequency output power of the device is seven watts or less. Such devices only generate very localized fields.

Limits of RF exposure for the general public were set at one-fifth of the occupational exposure limit in the appropriate frequency range. Lower exposure limits for the general public were recommended for a number of reasons:

(i) exposure could occur for 24h/day

(ii) broader spectrum of health sensitivities in the general public than the working population (sick, disabled, infirmed, babies, children); and

(iii) our lack of knowledge of possible health effects from long term chronic exposure suggested that an additional safety factor was necessary.

When developing the IRPA guidelines, a number of questions had to be addressed:

(i) What are the populations being protected and are there any differences that should be considered (occupational verses general public exposure)?

(ii) In view of our incomplete understanding of the interaction mechanisms underlying biological effects of RF exposure, and there presently being no predictive theory possible for non thermal effects, what allowances must be given to these effects?

(iii) From the information available on the absorption of RF energy in humans, and the fact that most experimental data were accummulated at frequencies above 1 GHz, how should the frequency ranges be divided to provided the same degree of protection from 100 kHz to 300 GHz?

(iv) How valid is it to extrapolate results of animal experiment to possible effects in humans?

(v) Should the exposure standard take account of indirect or secondary effects such as RF shocks and burns?

(vi) How should one overcome the lack of knowledge of relating peak SAR to observed biological effects?

(vii) Environmental conditions - should the exposure limits be protective under the most adverse conditions of temperature, humidity and air movement?

(viii) Is there an altered response of humans taking medicines while being exposed to electromagnetic fields?

(ix) Are there possible combined effects of RF electromagnetic energy with drugs, chemicals or physical agents?

(x) What are the implications of effects reported with modulated microwave fields on the central nervous system and the possible existance of "power" and "frequency" windows for effects?

(xi) Is there sufficient data on effects from long term, low level RF exposure? If not, is it reasonable to require an increased safety factor?

Effects on behaviour (convulsion, work stoppage, work decrement, decreased endurance, perception and aversion of the exposing field) seem to be the most sensitive indicator of human health hazard from exposure to RF fields giving SAR's above 4 W/kg (an implied "dose threshold") (3). Using this and the data on human absorption of RF energy, a set of exposure limits can be developed which incorporates an "apparent" constant safety factor. The frequency ranges in the exposure limits were determined taking into account RF energy absorption for all possible human sizes (including babies), partial body resonances (e.g. head) and the "hot spots" that can occur up to 2 GHz.

The IRPA/INIRC felt that a clear distinction was necessary between occupational and general public exposure. Safety factors should incorporate: some allowance for "non-thermal" effects (especially for the general public); the fact that most bioeffects data were obtain in animals at frequencies above 1 GHz and results are extrapolated to possible effects in human exposured at lower frequencies; and our poor knowledge of potential effects from long term, low level exposure. It was also felt that exposure standards should be "safe" in the most adverse environment (temperature, humidity etc) and working conditions (RF shocks and burns).

References

1. "Radiofrequency and microwaves", Environmental Health Criteria 16, United Nations Environment Programme, World Health Organization, International Radiation Protection Association, WHO, Geneva, 1981.

2. "Interim guidelines on limits of exposure to radiofrequency electromagnetic fields in the frequency range from 100 kHz to 300 GHz", International Radiation Protection Association, Health Physics (in press).

3. "Safety levels with respect to human exposure to radiofrequency electromagnetic fields, 300 kHz to 100 GHz", American National Standards Insitute, ANSI Committee C95.1, IEEE, New York, NY, 1982.

EXTREMELY LOW FREQUENCY (ELF, 0 TO 300 Hz) ELECTRIC AND MAGNETIC FIELDS, AND HEALTH PROTECTION

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The principal and universal source of human exposure at ELF frequencies is the widespread use of electrical energy distributed at frequencies of 50 Hz and 60 Hz. Research on biological effects concentrated at 50/60 Hz and thresholds for several effects (paralysis, heart fibrillation, field perception, etc) have been estabilshed. Levels, at which no immediate health effects and no discernible pathology over a period of years appear, can be indicated. However, recent research raised the question of long-term effects, such as leukemogenic and genetic effects. This is far from proven, and epidemiological and animal studies are needed to verify these hypotheses.

At frequencies other than 50/60 Hz, frequency-dependent phenomena appear over a certain field strength range, below and above which the effect cannot be demonstrated (amplitude window). Such an effect is the calcium efflux from brain tissue at 8 to 16 Hz. The threshold for induction of magnetophosphenes varies with frequency. Specific time-amplitude relationships are needed to induce electrical bone growth stimulation, electrical cell fusion and rotation.

The empirical observations will be discussed in the light of possible mechanisms of interaction, the understanding of which is still incomplete, and which are still controversial. Most data indicate that the site of primary interaction for several phenomena is the cell membrane, and that interference with electrochemical information transfer plays an important role.

Practical implications for general public, occupational and patient (bone growth stimulation, electroanaesthesia, nuclear resonance imaging) health safety will be discussed.

LIMITATION OF EXPOSURE TO AIRBORNE ULTRASOUND

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Ultrasonic equipment used e.g. for cleaning, drilling and emulsification has caused workers to complain from symptoms like nausea, headache and fatigue, and high noise levels of audible subharmonic components may result from ultrasonic cavitation. Consumer devices such as ultrasound intrusion controls, door openers and guidance aids for blind people can cause nervous reactions.

Based on a thorough review of the physics, the bioeffects and the possible hazards from ultrasound, the International Non-Ionizing Radiation Committee of IRPA has formulated guidelines for the setting of exposure limits to airborne ultrasound. For the ultrasonic frequencies, third-octave band acoustic pressure levels of 110 dB for workers and 100 dB for the general public are the proposed limits. The rationale for setting these limits and practical methods of protection including pressure level measurement will be discussed.

ESTABLISHING EXPOSURE LIMITS FOR ULTRAVIOLET RADIATION

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INTRODUCTION

The International Non-Ionizing Radiation Committee (INIRC) of the International Radiation Protection Association (IRPA) has recently proposed a set of guidelines for maximum personnel exposure to ultraviolet radiation (UVR). This paper will discuss the basis for these guidelines. The guidelines cover occupational and general population exposure to optical radiation within the wavelength range of 180 nm (the edge of the vacuum ultraviolet) to 400 nm (the edge of the visible spectrum). UV laser radiation is not covered by this proposal. Some of the underlying assumptions of the UVR guidelines may be invalid for highly monochromatic laser radiation.

BIOLOGIC EFFECTS OF UVR

The UVR spectrum is frequently divided into three spectral bands for ease in discussing biologic effects and health protection standards. As with any such spectral band scheme, the dividing lines are not truly fine lines. These bands (from the CIE) are: UV-C from 100 nm to 280 nm, UV-B from 280 nm to 315-320 nm, and UV-A from 315-320 nm to 380-400 nm. Relatively low irradiances of UV-C and UV-B radiation can cause photokeratitis ("welder's flash") and erythema ("sunburn") if delivered over a period of hours. Far greater irradiances of UV-A are required to cause these effects--often leading to the mistaken impression that UV-A radiation is harmless. Chronic exposure to UVR, especially UV-B, is known to cause accelerated skin aging, skin cancer and lenticular opacities (cataract), as well as other ocular effects. The widespread use of UVR in industry includes many new applications in photoresist processes, photocuring and welding; and UVR is used in cosmetic tanning, dermatology, and dentistry. This increased use necessitates the development of exposure limit guidelines for UVR.

GUIDELINES ON LIMITS OF EXPOSURE TO UVR

In recommending exposure limit (EL) guidelines, the INIRC was well aware of the great difficulties of deriving a generally applicable set of limits for this part of the optical spectrum. Firstly, it is generally recognized that UVR has beneficial health effects as well as adverse effects. The limits are not meant to preclude the beneficial use of UVR in medicine (nor elective UVR exposures for cosmetic purposes which normally exceed the EL guidelines). Secondly, skin sensitivities to UVR exposure vary enormously with racial factors and skin pigmentation for both acute and chronic effects. UVR irradiances which may not affect some individuals may be a hundredfold above levels which may affect sensitive skin.

The opinions or assertions herein are those of the author and do not necessarily reflect the official position of the US Department of the Army or the US Department of Defense.

Photosensitization resulting from pharmaceuticals, chemicals and systemic disease states make some individuals extremely sensitive to UVR. The geometry of exposure and the wearing of hats and other apparel also greatly affect the likelihood of adverse effects resulting from a given exposure dose. While chronic exposure to UVR is known to cause skin cancer, it is extremely difficult to quantify a threshold exposure below which there is no risk of carcinogenesis. When considering all of these factors, it is quite necessary that the guidelines be applied intelligently by professionals with a knowledge of these controlling factors. The EL's certainly cannot be considered as fine lines between safe and hazardous exposure conditions.

For acute affects it is well known that reciprocity exists between irradiance (exposure dose rate) and exposure duration; i.e., the exposure dose required to ellicit a specific biologic effect is constant over a wide range of exposure durations--from microseconds to several hours. Natural biologic repair of injured tissue causes this reciprocity relation to break down for exposures greater than 8-24 hours.

It is generally accepted that the primary UVR interaction mechanism with biological tissue is photochemical. For this reason there can be very significant variations in tissue sensitivity with a change in wavelength. The term "action spectrum" is used to describe the variation in radiant exposure necessary to ellicit a given tissue response as a function of wavelength. When the action spectra for threshold photokerititis and skin erythema are plotted together, it is possible to draw an envelope curve to include both, This approach has been followed in deriving an exposure limit (EL) action spectrum. Figure 1 shows some of the biologic threshold data along with the envelope curve of the guideline EL. A tabulated list of EL values at representative wavelengths in the UV-C and UV-B is provided in Table 1. Of course intermediate values of the EL exist at intermediate wavelengths and may be determined by interpolation. For most broad-band light source spectra (e.g., from lamps and arcs) spectroradiometric data taken at every five to ten nanometers is quite sufficient to calculate permissible exposure durations.

To calculate permissible exposure durations for broad-band sources it is necessary to have a spectral irradiance distribution at the location of the potential exposure. The radiation spectrum is then weighted against the EL envelope action spectral values of S_λ to obtain an effective irradiance, E-eff in W/cm^2:

$$\text{E-eff} = \sum E_\lambda \cdot S_\lambda \, \Delta\lambda \qquad (1)$$

where E_λ is the spectral irradiance in $W/(cm^2 \cdot nm)$ and S_λ is the UVR EL spectral sensitivity function as a function of wavelength λ in nm. The maximum permissible exposure duration in any 24-hour period is then t-max (in seconds) and is the maximum daily exposure at the normalization wavelength of 270 nm (i.e., 0.003 J/cm^2) divided by E-eff in W/cm^2:

$$\text{t-max} = 0.003 \text{ J/cm}^2 / \text{ E-eff} \qquad (2)$$

Some judgement of occupancy times and exposure conditions must be made in the proper measurement of irradiation levels and calculations of E-eff and t-max.

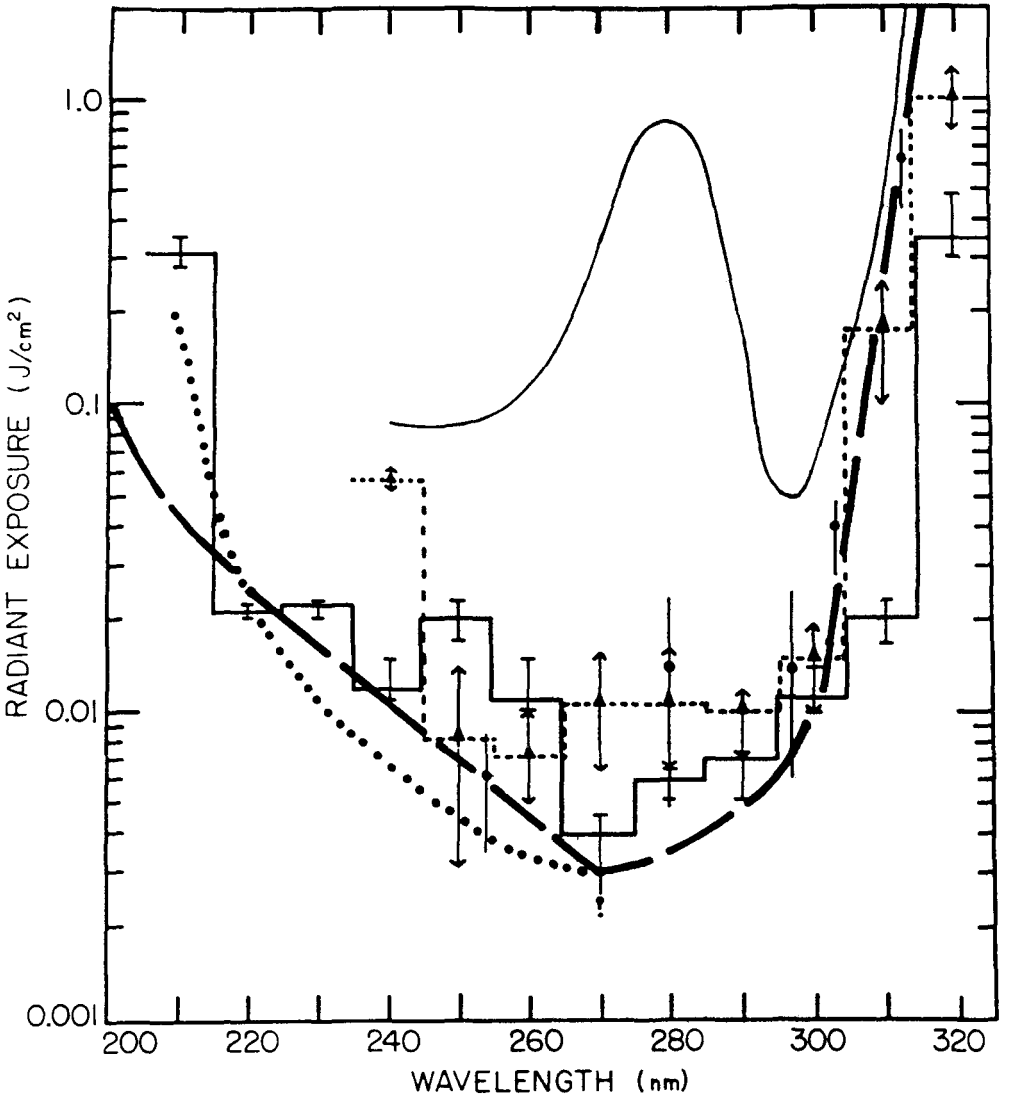

Figure 1. EL Envelope Curve for UV-B and UV-C Wavelengths. The bold dashed line is the EL. The solid curve is the classical erythema curve developed in the 1930's. The dotted-line histogram is threshold erythema; the solid-line histogram is threshold photokeritits. See references for detailed sources of this data.

For UV-A exposures, no spectral weighting function is applied. For wavelengths between 318 nm and 400 nm the EL is 1.0 J/cm^2 for 0.1 μs to 1000 s. The total UV-A irradiance is divided into that EL to determine the t-max for that spectral band, unless the irradiance is less than 1 mW/cm^2, in which case it does not exceed the UV-A EL irradiance for more lengthy periods, i.e., greater than 1000 s.

TABLE 1. Representative EL Values.

Wavelength (nm)	EL (J/m^2)	EL (mJ/cm^2)	Relative Spectral Effectiveness S_λ
180	2,500	250	0.012
190	1,600	160	0.019
200	1,000	100	0.03
205	590	59	0.051
210	400	40	0.075
215	320	32	0.095
220	250	25	0.12
225	200	20	0.15
230	160	16	0.19
235	130	13	0.24
240	100	10	0.30
245	83	8.3	0.36
250	70	7.0	0.43
254	60	6.0	0.50
255	58	5.8	0.52
260	46	4.6	0.65
265	37	3.7	0.81
270	30	3.0	1.0
275	31	3.1	0.96
280	34	3.4	0.88
285	39	3.9	0.77
290	47	4.7	0.64
295	56	4.6	0.54
297	65	6.5	0.46
300	100	10	0.30
303	250	25	0.19
305	500	50	0.060
308	1,200	120	0.026
310	2,000	200	0.015
313	5,000	500	0.006
315	10,000	1,000	0.003
316	15,000	1,500	0.002

CONCLUSIONS

While some difficulties remain to be resolved to achieve UVR exposure guidelines, this IRPA/INIRC effort has generally been met with a favorable response. Hopefully an approved guideline EL will be approved during 1984 and you may expect to see it published in the Health Physics Journal within 12 months.

REFERENCES

1. IRPA/INIRC, Draft Guidelines on Limits of Exposure to Ultraviolet Radiation of Wavelengths Between 180 nm and 400 nm (Incoherent Optical Radiation), INIRC Secretariat, Fontenay-aux-Roses, March 1983.

2. Sliney, D. H., and M. L. Wolbarsht, Safety with Lasers and Other Optical Radiation Sources, Plenum Publ. Corp., New York, 1980.

THE DEVELOPMENT OF EXPOSURE LIMITS FOR LASER RADIATION

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Lasers have been a cause of public concern only for the past 20 years. During this period extensive research has resulted in a reasonably good understanding of the biological effects induced by optical radiation emitted from such devices. There is now good agreement between national and international bodies concerning exposure limits and exposure standards. The exposure limits are reasonably complex being related to both variations in physical parameters of the source and fundamental differences in the biological targets, namely the skin and the eye. Between 400 and 1400 nm the eye is a particular problem in that it is specifically adapted to increase the effective irradience between the cornea and the retina. A further complication for the eye is that three different damage mechanisms occur whose thresholds are roughly related to time duration of the exposure.

This paper will discuss the above problems highlighting those areas of limited data and relating the scientific arguments used in establishing the present standards. Particular reference will be made to the IRPA guidelines on the limits of exposure to laser radiation.

MEASUREMENT OF THE RADIATION OF MICROWAVE OVENS

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Due to the high increase of the use of microwave ovens in households and restaurants, it was necesarry to control the radiation of each installed oven. Till now there are no law or regulations by the Austrian Government. Two years ago a working-group was established to work out limits for occupational exposures and exposures for members of the public.

While the first part of the work shows the fundamental characteristics of microwave radiation, the exposure effects in man and the standards, the second part gives the measurement techniques for measuring the power density of the microwave ovens.

Over 50 microwave ovens were measured and also the electrical circuit was controlled. Nevertheless there was no value over the allowed limit of 5 mW/cm^2 in 5 cm distance, we could find out some problems, for instance the outlets of the ventilation system and the doors of the cookingroom.

As a result it was tried to group the different ovens and to show their advantages and disadvantages.

ULTRA VIOLET HAZARDS FROM THE USE OF ARTIFICIAL TANNING EQUIPMENT

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Introduction

In recent years one of the growth industries in the U.K. has been the manufacture and use of artificial tanning equipment. The increase in the use of Ultra Violet Radiation (UVR) for cosmetic purposes has been accentuated by the availability of fluorescent tubes which principally emit the UV_A range of the spectrum (400-315nm) and are said to be free from short term effects. This paper provides quantitative data on effective irradiance levels in the UV_A and UV_B (315-280nm) ranges of the Ultra Violet Spectrum from artificial tanning equipment available for use in commercial premises (solaria) and in the home. These data are placed in perspective by comparing them with the levels of UVR from natural sunlight and with the levels known to cause acute UVR injury.

Casualty Department records from three major hospitals in the North West of England have been analysed to ascertain the frequency, type and causes of short term UVR induced injuries.

Although at present there is no specific legislation in the United Kingdom covering solaria, operators have responsibilities particularly under section 3 of the Health & Safety at Work etc. Act 1974, i.e. to ensure that their activities do not endanger anyone and to provide information to their clients regarding the potential hazards to their health and safety. In December 1982 the Health & Safety Executive issued a Guidance Note (GS18) entitled "Commercial Ultra Violet Tanning Equipment" which provided general recommendations in relation to all types of UV tanning equipment. Some six months after the publication of GS18 an assessment has been made of the extent to which operators were complying with the document's recommendations. Particular attention has been paid to those recommendations, which if not followed would be liable to lead to an increase in the number of acute UVR injuries.

Methods

Visits were made to fourteen commercial solaria. Measurements of the effective irradiance levels were carried out using an International Light IL730A Actinic Radiometer coupled to a SEE240 detector and a BLK270 filter for the UV_B region and to a SEE015 detector for the UV_A region. Measurements were made at the central position on the sunbed and the data is presented in Table 1. Allowance was made for the average thickness of a body in positioning the detector. Data for solaria using UVR sources below the table, were obtained with the detector head in contact with the table. A questionnaire covering the recommendations of Guidance Note GS18 was completed at each solarium visit.

Effective irradiance measurements in both the UV_B and UV_A regions using the IL730A were made on sun/health lamps used in the home and are shown in Table 2.

Data on patients attending Casualty Departments for sunlamp/sunburn injuries during the first seven months of 1983 have been obtained from three hospitals in the North West of England. This information, presented in Table 3, was obtained from both the casualty department records and from a questionnaire completed by hospital staff.

TABLE 1

Measurements of the Effective Irradiance (Wm^{-2}) - Solaria

Solarium No.		UV_A Region		UV_B Region		Ratio UV_A : UV_B	
		Over Head	Under Table	Over Head	Under Table	Over Head	Under Table
1	Unit A	58·9	73·0	0·05	0·08	1202	890
	Unit B	50·8	67·8	0·06	0·06	891	1210
2		104·6	91·1	0·13	0·12	811	792
3		56·3	55·4	0·09	0·05	654	1131
4		65·6	69·8	0·07	0·08	911	931
5		73·7	80·3	0·07	0·07	1052	1085
6		249·0		0·08		3112	
7	Unit A	32·0		0·04		821	
	Unit B	30·1		0·03		1003	
8		43·3		0·05		941	
9		76·2	67·9	0·07	0·06	1030	1095
10	Unit A	32·3	64·9	0·07	0·07	482	915
	Unit B	38·3	80·7	0·05	0·10	798	791
11		2·4		1·94		1·23	
12	Unit A	6·3		0·60		10·5	
	Unit B	3·5		0·63		5·6	
	Unit C	2·1		3·37		6·2	
13		1·6		0·28		5·7	
14		12·61		0·19		650	

TABLE 2

Effective Irradiance Measurements (Wm^{-2}) - Home Use Equipment

Unit No.	UV_A	UV_B	Ratio UV_A:UV_B
1	10·1	13·1	0·8
2	13·2	1·4	9·4
3	6·5	1·3	5·0
4	9·8	12·3	0·8
5	51·3	0·07	738

Measurements made at lamp - detector distance of 0·3m

TABLE 3

Hospital Questionnaire - Summary of Format

Date	Sex/Age	Sunlamp or Sunburn	If Sunlamp Home or Commercial	Injury/ Treatment

Summary of Hospital Data

Hospital	Total No. of patients seen by casualty department	Total No. of male patients with sunlamp injuries	Total No. of female patients with sunlamp injuries	Total No. of patients, male plus female with sunburn injuries
1	27900	6	6	39
2	14242	2	2	26
*3	37352	7	28	48

* Jan - June inclusive only.

Discussion

There are both chronic and acute effects of UVR but as the penetration of the radiation is small these effects are limited to the surface of the eyes and the unprotected skin. Erythema (sunburn) is a visible sign of UV injury which can appear after latency periods of a few minutes to a few hours. Erythema caused by UVR is confined to the exposed areas and in its more severe forms can result in inflammation, blistering and peeling of the skin. It is usually followed by tanning of the skin which becomes noticeable within two and three days after irradiation. Induction of erythema by UVR is wavelength dependent with the maximum effect being produced in the range 290-320nm. UVR in the range 320-400nm can produce erythema but at much higher doses than UV_B.

Exposure to UVR can result in ocular damage before the recipient is aware of the potential danger. The symptoms of the principal effect, keratoconjunctivitis, are discomfort or pain similar to that resulting from a foreign body in th eye, an aversion to bright light and inflammation of the cornea and conjunctivia. The individual is usually incapacitated for 6-24hours; the discomfort disappears within 48hours and exposure rarely results in permanent damage. UV_B and UV_C (280-100nm) wavelengths are primarily responsible with the maximum effect for a given exposure produced at a wavelength of 270nm. UV_A wavelength light is not known to cause keratoconjunctivitis.

There are four types of ultra violet tanning equipment in use, i.e. UV_A low

pressure fluorescent tube systems, UV_B low pressure fluorescent tube systems, UV_A medium and high pressure high intensity lamp systems with filters for UV_B and UV_C provided, and UV_B medium pressure lamp systems. The ultra violet spectral emission characteristics of these are all quite different and this is reflected in the data presented in Tables 1 and 2. UV_B radiation was found to be present in all lamps and tubes surveyed.

Natural UVR exposure (i.e.from the sun) received by different individuals will depend not only upon the quality and quantity of the UV environment but also on the behaviour of individuals concerned. The effective irradiance in the UV_B region at noon on a summers day in the U.K. is typically in the region $0{\cdot}1\text{-}0{\cdot}5 Wm^{-2}$. The reported minimal erythema dose (MED) for normal untanned human skin due to wavelengths 297-300nm is in the order of $200 Jm^{-2}$(1). The worst case from the Solaria survey was a unit with an effective irradiance output of $3{\cdot}37 Wm^{-2}$ thus providing a dose to the client of $6066 Jm^{-2}$ for a 30minute session. The majority of the units surveyed were of the low pressure UV_A tube systems which provide typical doses of $125 Jm^{-2}$ for 30minute exposure. In addition to UV_A systems units for use in the home typically are combined with an infra red facility and are known as Health Lamps. All the units of this type provided 30minute doses in excess of the MED with the worst case providing over $23{,}000 Jm^{-2}$. Although the minimum threshold radiant exposure for corneal effects is $40 Jm^{-2}$ at 270nm the value remains below $100 Jm^{-2}$ from 270-310nm(2). The data in Tables 1 and 2 demonstrate that goggles are clearly essential for all types of UV tanning equipment.

The three recommendations of GS18 with the most likely bearing on reducing the number of short term effect injuries are that operators should (a)ensure that properly trained staff are available, (b)provide an adequate exposure regime for each individual customer and (c)provide adequate customer information. Of the fourteen solaria surveyed none provided adequate staff training, only two provided personalised exposure regimes and only three provided adequate customer information. The information provided for customers purchasing health lamps for home use is unsatisfactory. It is not surprising, therefore, that the casualty department data revealed a significant number of patients with UVR injuries. The age group most at risk was found to be the 16-25s with the majority of sunlamp injuries being sustained as a result of using equipment in the home.

Current evidence indicates that the incidence of chronic skin injuries (premature skin ageing, induction of skin cancers) may increase as a consequence of the use of sunbeds, with the size of the increase dependent on accumulated exposure (1).

Conclusions and Recommendations

1. The risk of acute skin injury other than minimum erythema due to exposure of UV_A fluorescent tube systems for a normal exposure time of 30minutes is low. This is not the case for other forms of commercial UV tanning equipment.
2. The wearing of protective eyewear is essential for all UV tanning equipment.
3. Equipment for use in the home provides the greatest risk of acute UVR injury. Because of the high UV_B output sun/health lamps are the highest risk but abuse of UV_A fluorescent systems can also lead to excessive exposures.
4. In the U.K. strict compliance to GS18, by solarium operators, will minimise the number of acute UVR injuries.
5. Provision of much more detailed information to would-be purchasers of home use equipment is essential.
6. There is a need to promote an increase in public awareness regarding the risks involved in prolonged exposure to UVR.

References: (1) Environmental Health Criteria 14:Ultraviolet Radiation World Health Organisation 1979. (2) Pitts,D.G.(1973), Health Physics 25, 559

INTERNATIONAL PROGRAMMES AND MANAGEMENT OF NON-IONIZING RADIATION PROTECTION

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Progress in science results in ever faster developing technological applications which besides their beneficial contribution to human life conditions may also carry along some adverse health effects. Thus the rapidly expanding use of ultrasound and radiofrequency radiation emitting devices as well as of lasers in industry, research, medicine and even consumer products, during the last twenty years has increased the possibility of human exposure to such radiations and at the same time concern about their possible health effects.

Therefore, the health authorities of an increasing number of countries become involved in the development of recommendations, regulations or technical advice to limit the exposure of the workers and the general public and they are looking to the international organizations for some guidance in this respect. As a matter of fact, the methodology of protection against ionizing radiation which enabled the safe development of its industrial and medical applications all over the world is often quoted as an example, and every one has in mind the invaluable help provided by the international organizations to resolve the complex scientific and administrative problems encountered in the early management of radiation protection.

During the last decade, several international organizations, through their own programmes or in cooperation, have attempted to answer this request by reviewing the available data on biological effects, providing guidance on exposure limits and product performance standards, and developing recommendations for safe practice. Let us briefly review the role played by these different organizations which contribute to progress in protection against NIR.

The INTERNATIONAL RADIATION PROTECTION ASSOCIATION (IRPA) started its activities in NIR in 1973 with a session devoted to this topic at its Third International Congress in Washington. The year after the Executive Council instituted a Working Group on NIR which in some way was the first stone of the International NIR-Committee (INIRC) that IRPA created in 1977. The chief objectives assigned to the Committee were to lay down the general principles of non-ionizing radiation protection, to determine the most appropriate exposure limits for the different NIR and to explore with other international organizations ways of furthering protection in this field.

The development of protection standards requires a correct assessment of the possible biological effects and of the exposure-response relationship as well as appropriate information on the various sources in use, the resulting levels of exposure and the people at risk. The collection of the necessary background data requires the contribution of a number of experts, institutions and countries. Therefore, it was agreed that the WHO/Environmental Health Division and the IRPA/INIRC would cooperate in the preparation of environmental health criteria documents relating to NIR with the financial support of the UNEP. On the basis of the scientific data thus collected the IRPA/INIRC has developed guidelines on exposure limits for the different NIR (1), which have been proposed to the World Health Organization and the International Labour Organization for consideration.

The interim guidelines for electromagnetic radiation in the radiofrequency range (100 kHz to 300 GHz) and for ultrasound have been recently issued in Health Physics. Those for lasers and for ultraviolet radiation are expected to be published before long. They will be subjected to periodic revision to be kept in line with advances in knowledge of the relevant biological effects. The purpose of the guidelines is to deal with the basic principles of protection against the different NIR, including appropriate exposure limits, so that they may serve as guidance to international and national bodies or individual experts who are responsible for the drafting of regulations, recommendations or technical advice to protect the workers and the general public from the potentially adverse effects of NIR. Besides, the Committee reviewed the various physical and radiometric quantities recommended by several organizations such as ISO, IEC, CIE, ICRU and others, and is preparing a report on this topic.

The IRPA/INIRC also collaborates with the International Labour Office by drafting codes of safe practice which are intended to inform workers of the hazards and the appropriate precautions that need to be taken to minimize NIR exposure.

The UNITED NATIONS ENVIRONMENT PROGRAMME (UNEP), a governmental organization created in 1972, devotes all its efforts to the protection of the human environment against nuisances of any kind. UNEP financially supports a programme on the assessment of health effects of environmental pollution which is carried out by the World Health Organization and explicitly includes non-ionizing radiation.

The WORLD HEALTH ORGANIZATION (WHO), whose very broad objective includes any topic related to health and environmental hygiene, has manifested since 1971 its interest in the health problems raised by the increasing use of NIR emitting devices through various activities conducted either by the Main Office in Geneva or the Regional European Office in Copenhagen. Some of its chief actions in this field are: the convening of international working groups; co-sponsoring of international symposia dealing with biological effects of NIR; establishment of two WHO Collaborating Centres, respectively in Warshaw (Poland) and in Rockville (USA); publication of a Manual on non-ionizing radiation protection by the Regional Office for Europe (2). But more particularly, the WHO is responsible for the Environmental Health Criteria Programme supported by UNEP that resulted in the publication of several health criteria documents relating to NIR, for which the WHO/Environmental Health Division cooperated with the IRPA/International NIR-Committee. Before being published, the draft documents are submitted to a number of institutions and individual experts, mostly through the national focal points of WHO, and all comments are reviewed and taken into account.

As a general rule, the purpose of a health criteria document is to summarize, review and evaluate available information on the effects of the specific factor considered that may influence human health and the environnement. Four such documents (3) have already been published under the joint sponsorship of UNEP, WHO and IRPA, dealing respectively with Ultraviolet Radiation, Radiofrequency and Microwaves, Ultrasound, Lasers and Optical Radiation. A further document on Extremely Low Frequency Radiation is in preparation. Each document includes a description of the physical characteristics of the radiation concerned, an overview of the sources, applications and exposure levels, measurement methods and instrumentation, a review of the data on biological effects gained from animal experimentation and observations in humans, an evaluation of the health risk of human exposure and a survey of the existing protection standards. These criteria then become the scientific data base for the development of exposure limits and codes of practice.

Future possible WHO/IRPA joint activities might include environmental health criteria for other NIR's, the development of principles for health surveillance, the organization of workshops on the objectives of NIR protection...

The INTERNATIONAL LABOUR ORGANIZATION (ILO) cares for the protection of workers against sickness, disease and injury arising out of their employment. One of the most important tools of ILO for the achievement of safe working conditions is the " Model Code of Safety Regulations for Industrial Establishment.", which has no binding force but is intended as a guide to government and industry. This code contains a section on infrared and ultraviolet radiations. Information on occupational risks due to some specific NIR may also be found in ILO's Encyclopedia on Occupational Health or obtained through the International Occupational Safety and Health Information Centre.

In 1976, The International Labour Conference adopted a resolution which amongst other matters called for appropriate studies of occupational hazards arising from non-ionizing radiation to be included in ILO's International Programme for the Improvement of the Working Conditions and Environnement (PIACT). This programme seeks to give governments, employers' and workers' organizations, the necessary help in drawing up and implementing programmes for the improvement of the working environment. The ILO applied to the International NIR-Committee of IRPA as an expert body to carry out a study on the occupational hazards arising from the various types of non-ionizing electromagnetic radiation and the measures to be taken to protect workers against such hazards (4). More recently, the ILO also decided to include a Code of practice on the protection of workers against radiofrequency radiation in the series of such codes that it issues to provide guidance for government agencies, employers ans workers. As this same topic was already included in IRPA/INIRC's programme, the ILO applied again to the INIRC for the preparation of a draft. Among others, this Code will include a description of the responsibilities of the employers and workers; an overview of the sources; guidelines on exposure limits, operational protection and health surveillance.

THE INTERNATIONAL ELECTROTECHNICAL COMMISSION (IEC), provides an important technical contribution by the development of product performance standards that represent large international consensus. Several IEC standards (5) have already been issued for NIR-emitting devices. Although they bear mainly on electrical or other associated hazards, some of them include requirements for protection against excessive exposure to radiation.

At a more regional level, the COMMISSION OF THE EUROPEAN COMMUNITIES (CEC) became involved in NIR in the early seventies when concern developed in some european countries about the rapidly increasing importation and use of electromagnetic devices. Following the request of several member countries, a draft Directive on protection standards for microwaves, that was proposed in 1980, is presently being revised and extended to cover a largerradiofrequency range. The Commission also requested the Directorate for Research to examine the possibility of launching a specific multi-annual Community research programme in this sector, similar to that undertaken for ionizing radiation.

CONCLUSION.

The protection of the health of the workers and the general public requires that activities which might involve hazards from exposure to non-ionizing radiation be made subject to regulation. At an international level, however, the management of NIR protection may be considered as having a delay of almost a generation when compared to that existing in the field of ionizing radiation. Indeed, except for some pathological effects resulting from higher exposures, our knowledge on the interaction of most non-ionizing radiations with living matter and the resulting biological effects is still poor. This sometimes leads to very different assessments of the health risks associated with exposure to NIR and as a consequence, in some cases such as that of radiofrequency radiation for instance, to a great diversity in the protection policies and exposure limits already used in some countries. The lack of international agreement on the protection standards to be used for NIR

constitutes a major drawback for the development of safety regulations in those countries where they do not yet exist. Therefore, it must be hoped that the efforts outlined above to achieve international cooperation in the field of NIR together with progress in our knowledge on the biological effects will allow protection against NIR to develop in the same climate of international agreement than that which has been reached for ionizing radiation.

REFERENCES.

(1) IRPA/INIRC. Interim guidelines on limits of exposure to
-electromagnetic fields in the frequency range from 100 kHz to 300 GHz.Health Physics (in press), 1984
-airborne ultrasound. Health Physics(in press), 1984
-laser radiation of wavelengths between 180 nm and 1 mm (in preparation)
-ultraviolet radiation of wavelengths between 180 nm and 400 nm (Incoherent optical radiation) (in preparation).

(2) WHO/Regional Office for Europe. Non ionizing radiation protection (WHO Regional Pub., European Series n° 10).
WHO, Copenhagen, 1982.

(3) UNEP/WHO/IRPA. Environmental Health Criteria
-EHC 14. Ultraviolet radiation. WHO, Geneva, 1979
-EHC 16. Radiofrequency and microwaves. WHO, Geneva, 1981.
-EHC 22. Ultrasound. WHO, Geneva, 1982
-EHC 23. Lasers and Optical Radiation. WHO, Geneva, 1982.

(4) ILO. Overview of occupational hazards from non-ionizing electromagnetic radiation. ILO, Geneva,(in press).

(5) International Electrotechnical Commission.
-Safety of household and similar electrical appliances, Part2: Particular requirements for microwave cooking appliances, IEC Pub. 335-25, Geneva, 1976.
-Safety in electroheat installations, Part 6: Specifications for safety in industrial microwave heating equipment . IEC Pub. 519-6, Geneva, 1982
-Radiation safety of laser products,equipment classification, requirements and user's guide. IEC Publication (in peparation)

TEN YEARS EXPERIENCE WITH A LASER SAFETY CODE

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During the period 1967-1972, we made several studies of the use and need for the control of industrial laser exposures in New York State. These studies indicated rapidly expanding laser usage by generally untrained individuals particularly in the construction industry. At the time both high and low intensity lasers were in field use. Based in part on experience in controlling ionizing radiation exposures a performance type Industrial Standard consistent with national guides such as ACGIH (1) was formulated and distributed for comment. Public Hearings were held throughout the State and a generally satisfactory standard Rule 50, Lasers (2) was promulgated.

Rule 50 separates lasers into three classes - exempt, low and high intensity laser. The division between exempt and low intensity being set at a value which was considered reasonably safe to view without permanent damage - 10 microwatts per cm^2 for CW lasers. The definition of low intensity laser and the demarcation value between low and high intensity laser was more difficult to establish. This was set at 3 watts per cm^2 for CW lasers. Lasers above this value are classified high intensity lasers. The requirement for some controls on high intensity lasers has never been in doubt. There is controversy on the need and extent of controls for low intensity lasers. The performance-type Rule required the registration of all high intensity lasers, certification of all operators of mobile lasers and approval of low intensity lasers. Approval of low intensity lasers was made considerably easier by the promulgation by U. S. Food and Drug Administration of a federal standard covering lasers (3). The intent of approving lasers was to make it easier to control laser exposure by built-in safeguards and also to simplify inspection procedures. A general safety inspector could check the laser for the approval tag and assume that it met the standard if tag was present.

Our experience with regulating ionizing radiation sources led us to believe that mobile lasers - those used outside a permanent installation would require additional controls. The standard therefore required the certification by examination of mobile laser operators. Since control requirements varied greatly between low and high intensity laser, two classes of certification were specified Class A for use of low intensity mobile lasers and Class B for use of high and low intensity lasers.

All firms possessing high intensity lasers must register them and appoint a laser safety officer with appropriate experience who is required to assure that the laser safety standards are adhered to by the firm. One area of controversy arose involving the extent of controls needed for low intensity lasers especially the helium neon lasers in use in the construction industry. The literature on the subject is somewhat ambiguous and it is uncertain that a 5 milliwatt helium neon laser could cause permanent eye damage. On the other hand the potential for projecting the laser beam, even a low intensity beam, is too great and the possibility for causing disturbances in the highway or airways justify the requirement of certification of the laser user.

Lasers in use by New York State Industry

The distribution of high intensity and unapproved low intensity lasers by type of industry is given in Table 1a. The only suprise and change from a

similar table, prepared in 1975, is in the lasers used for light shows and advertising displays. This distribution does not indicate, because they are unregistered, the large number of firms in the construction field and industry in general using low intensity, primarily helium-neon, lasers for precision measurements. Most high intensity lasers are used for welding, cutting, scribbing and trimming. While only in its infancy lasers are adaptable to automated processes. Problems have arisen in this application such as the automated sealing of tubes with tritium (4). This has arisen because insufficient development work has been done on the application of very high power densities on the materials involved. Precision work in the electronics industry has been met by using lasers. Table 1b summarizes the type of laser in use. Even for registered firms helium neon lasers are the predominant laser. Part of this arises from the fact that most high intensity lasers must be precision aligned which is done with the HeNe laser. Another type of laser in increasing use is the solid state laser-mainly by the communication industry. To date field measurements of these lasers have indicated that they can be used safely with no hazard to worker. This is due to the relatively large divergence of the laser light both from solid state laser and out the end of the fiber optic cable into which the solid state feeds.

Laser Light Shows

The entertainment and advertising industries have embraced lasers as dramatic adjuncts to their visual effect repetoire. Lasers are used not only for entertainment but their claimed artistic effect. Over the past seven years when this usage has increased, experiments of all types were tried and the Laser Code has permitted a relatively controlled increase in usage. While there are countervailing claims that the use of lasers in this area is too dangerous and inappropriate, even frivolous, use of high technology, Rule 50 does not prohibit this usage if the laser is used within the constrains of the rule. The predominant lasers in use are the argon, krypton and argon-krypton gas lasers. An occasional attempt is made to use the helium-neon laser but without a great deal of success. The helium-cadmium laser is also used on occasion for its deep blue line. We have required the laser light shows to comply with the following main requirements: 1) registration of firm 2) certification of laser operator and 3) maximum power and energy densities restricted in occupied areas to less than 1 microwatt/cm^2 and 1 microjoule/cm^2 in spectral region from 400 to 700 nanometers and no spectral radiation shall be emitted from laser system outside this band.

A recent inspection in a facility using a krypton laser tuned to red line (647.1 nm) had an output of 3.7 watts and with a scanning beam off a stationary mirror ball read 2 microjoules/cm^2. The operator increased the beam scanning frequency to reduce the energy density to 0.7-1 microjoules/cm^2. In practice the mirror ball is turning which reduces the effective power density. Our inspections of laser light show operators has generally found willingness on the part of the operators to comply with Rule 50 requirements.

Incidents

In general the history of the expanding laser use has been accomplished safely but it has not been incident free. Four incidents were reported and all occurred with laser radiation in the infra red band. Three were in a research environment and one was on the production line.

Incident #1 involved a technician with about 2 years laser experience working with the following laser:

Mfg: Coherent Radiation, Model No. 41
Type: CO_2 gas
Power: 325 watts
Use: R & D experiment
Date of Incident: 8-4-75

The technician was involved in sliding test material into beam irradiation area and irradiating material to study laser beam effects. In sliding test item into irradiation position and firing laser he irradiated his left index finger causing a hole in fingernail approximately 1 mm diameter and 2 mm deep. The employee was seen in plant clinic and no lost time was involved. Corrective measures involved additional laser safety training of laser workers.

Incident #2 involved the following laser:

Mfg: GTE-Sylvania Model 971
Type: CO_2 gas
Power: 1200 watts
Use: Automated Welding
Date of Incident: 6-8-81

A maintenance engineer with 1 year laser experience was attempting to remove the laser beam protector tube to monitor output power with station control and shutter control in auto position. The proper procedure for this authorized maintenance function was to place the shutter in "off" position and take the station out of the automatic run position. As the employee removed the tube the laser fired striking the employee in the right thumb. A 0.75 inch diameter burn pattern resulted from the unfocused laser beam. The burn healed uneventfully. Corrective measures included additional training for the maintenance engineers.

Incident #3 involved the following laser:

Mfg: Molectron Model MY34
Type: Neodymium Laser with Methane Cell
Power/Energy: 15 watts/1.5 joules
Wave Length: 1064, 532, 355, 266nm
Date: 10-15-82

Two researchers with several years experience working with lasers were performing stimulated Raman emission experiment. The 532 nm laser beam was focused into a methane cell causing emission at 630 and 460 mm. One researcher was looking into the cell with laser safety goggles to check for gaseous breakdown when he experienced a bright flash. Subsequent reconstruction of incident indicated that cell also radiated at 770 nm and it was this radiation that caused the damage. Measurements indicated the individual received approximately 0.8 millijoules in 10 nsec pulse. Literature (5) indicates that it would require 30 J/cm^2, 1064 nm photons in 15 nsec pulse to cause a 10 μm retinal lesion. Assuming a similar size lesion one can calculate that this individual received an exposure of about 260 J/cm^2 on the retina. This is sufficient to have caused the 3 mm blind spot that researcher described in clinic following incident.

Incident #4 involved the following laser:

Manufacturer: In-house
Type: Neodymium
Wavelength: 1064 mm
Date: 10-26-82

The person was working in lab with 4 other persons. He reported to company dispensary with complaint of irritation of eyes. Examination by opthamologist indicated minor damage to corneas of both eyes attributed by opthamologist to ultraviolet light. The person while working in lab on specified day did not wear laser safety goggles and viewed diffuse laser light. Subsequent investigation showed no exposure in spectral region from 200 to 400 mm. The Nd laser has a shielded Xenon pump lamp. The person indicated he knew of no other possible UV source he might have been exposed to.

In summary these laser incidents involved in 2 cases, Nd laser and in the other 2 cases, CO_2 laser. The recent incident literature also reveals the potency for damage of the infra red radiation, partially I am certain because it is unseen until damage occurs. The second point also common with other recent incidents is the susceptibility for laser injuries of an r and d environment even for experienced workers. Precautions in such situations must include a vigorous in-house program of laser safety instruction of all personnel which should be periodically repeated and independent audits of laboratory precautions to ensure compliance with good laser safety practices.

References:

1. ACGIH Publication "A Guide for Uniform Industrial Hygiene Codes or Regulations for Laser Installations" (1969); Cincinnati, Ohio 45201

2. Official Compilation of Codes, Rules and Regulations of State of New York, Title 12, Part 50, Lasers (1972)

3. Code of Federal Regulations, Title 21, Part 1040 (1976)

4. Bradley, F. J., Schuster, L., Cabasino, L., Tedford, C., Warren, D. and Fitzrandolph, L., Proceedings of 5th Congress of IRPA (1980) Pergamon Press, Oxford

5. Sliney, D. and Wolbarsht, M. "Safety with Lasers and Other Optical Sources" (1980) Plenum Press, New York.

Table 1a. Laser Use by Industry

General Manufacturing:	31%
Laser Light Shows :	29%
Electrical Machinery and Electronics:	21%
Research and Development:	8%
Aerospace:	6%
Electro Optical & Telecommunications:	5%

Table 1b. Type of Laser in Use

HeNe:	48.0%
Solid State (GaAs):	20.0%
Nd :	10.3%
CO_2 :	7.4%
Ar :	5.5%
Kr :	2.7%
Ruby :	1.2%
HeCd :	1.1%
Ar/Kr :	0.5%
Miscellaneous:	3.3%
(Dye, Xe, CO, etc.)	

DOSIMETRIC CONCEPTS AND HEALTH RISK EVALUATION IN NONIONIZING ELECTROMAGNTIC RADIATION PROTECTION

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GENERAL CONSIDERATIONS

IRPA recently issued guidelines (1) on limits of exposure to electromagnetic fields (EMF) in the frequency range from 100 kHz to 300 GHz. A concept of basic and derived working limits was introduced for the range from 10 MHz to 300 GHz. Basic limits are considered in terms of the specific absorption rate (SAR) expressed in W/kg. Working limits are derived using the frequency dependence of EMF energy absorption in the human body, and are specified in terms of incident unperturbed electric (E) and magnetic (H) field strengths or equivalent plane wave energy flux density (S) expressed respectively in V/m, A/m or W/m^2. Below 10 MHz no distinction between basic and derived limits was made, and the limits were specified only in terms of incident E and H field strengths. The need for refinement of guidelines was stressed. This requires further development of dosimetric concepts useful in health risk evaluation. The aim of this paper is to discuss possible approaches modelled after those used in ionizing radiation protection (2, 3).

The first step consists in identification of those parameters and quantities, which characterize EMF and their interaction with living systems, and which can be meaningfully correlated with bioeffects. The quantities mentioned above (E and H field strengths, S, SAR) do not account fully for the complexity of exposure conditions and mechanisms of interaction. EMF can be generated as continous (CW) or pulsed (PW) waves. A carrier wave may be amplitude (AM) or frequency (FM) modulated. EMF pulses may be of various shape, amplitude and duration (width). The pulse repetition rate and on-off periods (duty factor) may vary. Simultaneous exposure to EMF generated by various sources and intermittent, repeated exposures (fractionated dose) variously spaced in time introduce additional complexity. Analysis of biological responses should include the considerations of : 1. the relationship between exposure conditions and the propagation, deposition and conversion of EMF energy within biological objects or what may be termed the primary interaction, 2. the relationship between the primary interaction and bioeffects defined as changes in biological function and/or structure, 3. the distinction between direct bioeffects, i.e. direct interference of the primary interaction with biophysical and electrochemical phenomena in the living system, and indirect effects, i.e. changes mediated through chain reactions, feed-back loops and responses to direct effects. An indirect bioeffect may be many steps removed from the initial event, and may be detectable at sites different from that of the primary interaction (4).

SPECIFIC ABSORPTION RATE (SAR), SPECIFIC ABSORPTION (SA) AND THERMAL DOSAGE

Conversion of EMF energy into heat is the best understood mechanism of primary interaction. The SAR, defined as the time derivative

of the incremental energy absorbed by or dissipated in a mass contained in a volumed element of a given density (5), is convenient for considerations of bioeffects in terms of heat generation and dissipation rates. However, SAR is a "bulk" quantity dependent on dielectric propertie of tissues. Therefore, SAR does not lend itself to considerations of mechanisms of interaction at the cellular or macromolecular level in biological systems, where many species of molecules with different properties coexist. SAR can be spatially and termporally averaged. EMF energy absorption in biological objects is typically nonuniform (5). A whole body average (WBA-SAR) or an average over i g of tissue is used (1). Spatial peak SAR may be an order of magnitude higher than the WBA-SAR. Peak instantaneous SARs associated with modern PW sources may be many orders of magnitude higher than a corresponding time-averaged SAR. Once the SAR exceeds the capability of the system for heat dissipation, a temperature increase will occur. A predetermined temperature will be reached after a certain period of time, i.e. after a certain amount of energy will be absorbed. The total amount of energy absorbed over a period of time is termed the specific absorption (SA) and is expressed in J/kg. SA can be, similarly as SAR, averaged over a mass (volume) of tissue. An inherent danger in using the SAR and SA concepts lies in that that the use of any of those quantities may obscure the temporal and spatial variations of EMF energy absorption characteristics.

No systematic investigations of the relative significance of SAR vs SA in quantitating the dose rate or dose - bioeffect relationships were found in available literature. However, it should be noted that for rats and mice exposed under comparable environmental conditions to 2450 MHz CW radiation the lethal dose (WBA-SA) varied between 21 and 25 J/g (6, 7, 8). The SARs, exposure durations and body masses of exposed animals varied over one order of magnitude, from 10 to more than 100 W/kg, from 2 to 30 minutes, and from 30 to 450 g respectively. In other experiments (9), convulsions, which precede death, occurred in rats(400 g, 65 W/kg) and mice(30 g, 80 W/kg) at 24 or 25 J/g. However, an attempt to correlate literature data on effects on behavioral responses (9) with SAR or SA, did not demonstrate any clear advantage of using either the dose rate or dose to establish a quantitative relationship with effects. Assuming a thermal basis for disruption of operant behavior in rats, squirrel monkeys and rhesus monkeys, it was suggested (10) that this this phenomenon occurs only if body temperature is raised by at least $1^{o}C$.

For a proper analysis a definition of thermal bioeffects is needed. Such bioeffects may be directly dependent on temperature increase, as eg the passive increase in metabolic rate by 10% per ^{o}C. Indirect bioeffects consist in responses needed to maintain thermal homeostasis or in responses evoked by local thermally induced alteration in function of organs, which control functions of other parts of the body, as eg the thyroid gland. Such an example is also the decrease in core body temperature following EMF energy absorption in the hypothalamus (11). In this case, the thermal EMF bioeffect, evoked by dielectric heating, leads to a net decrease in body temperature and may seem paradoxical, as long the nonuniformity of EMF energy absorption is not taken into account. It should be added that SAR and SA do not account also for the work needed for activation and maintenance of thermoregulatory responses, i.e. the physiologic cost of thermoregulation.

Nonuniformity of EMF energy deposition leads to the induction of temperature gradients, which may reach values of 4 to 5°c and persist for long periods of exposure (12). This possibility is being used in clinical practice for hyperthermic EMF therapy of cancer. Experiences with clinical EMF hyperthermia demonstrated the usefullness of local thermal dosage. It was shown that cell killing depends on the minimum temperature and the duration (time) of heating at a particular location, i.e. within the tumor (13, 14). A concept of thermal dosage was developed (14). This is a function of time combined with a mathematical description of the time-temperature relationship. It was proposed to calculate an equivalent exposure time, which would correspond to "equivelent - minutes" at some reference temperature, chosen from an Arrhenius plot of cell killing rate at various temperatures. The reference temperature is chosen at the point of a break in the plot. For clinical hyperthermia in cancer therapy 43°C may be a convenient reference temperature, and equivalent "degree - minutes can be calculated (14).

This approach could be used to analyse effects of nonuniform EMF energy deposition in various organs, and to establish temperature thresholds for changes in function. Equivalent degree-minutes can be used to correlate the effects with the time during which a particular organ is subjected to an elevated temperature. Such an approach would allow to establish firmly a thermal (defined as temperature dependent) nature of an bioeffect. At present many authors consider an effect as thermal if any increase in temperature has occurred, or even if such an increase may heve had occurred. Such a speculative approach to thermal EMF bioeffects (eg in 15) serves only to obscure the physiologic basis of EMF bioeffects and deos not allow a scientific analysis of thermal vs nonthermal mechanisms of interaction.

In conclusion further progress in analysis of EMF bioeffects and health risk evaluation may be achieved if a concept of local organ dose (SA) and dose rate (SAR) together with a thermal dose equivalent will be introduced. Experiments should be designed to test the value of this approach. The design should allow to differentiate between the relative importance of SAR, SA and equivalent thermal dosage for the threshold for a given biological response during exposure, and the time - course, as well as the severity of response, i.e. the tolerance limit. Both theoretical considerations and the limited data available (eg 16) indicate that at phase shift temperatures the effects will not depend on SAR or SA, and proper correction factors will have to be introduced.

In the millimeter wave range resonant frequency dependent effects were demonstrated (17). It was also shown that biologically important molecules (eg DNA, see 18) may show resonant frequency absorption of EMF energy. Further refinement of analysis of such bioeffects requires the introduction of appropriate correction factors to account for frequency dependent phenomena, which necessitate interpretation in quantum mechanical terms. These do not lend themselves at present for dosimetric considerations, however, an attempt at developing correction factors for frequency dependent effects may provide an approach for accounting for such mechanisms indosimetric considerations in future.

DOSIMETRIC CONCEPTS AT LOWER EMF FREQUENCIES

The incident E and H field strenghts, proposed as limits for exposure at frequencies below 10 MHz (1) do not account for mechanisms of interaction, and can be considered as a preliminary approach. Available data (19) indicate that the cell membrane may be the site of the primary interaction at lower, particularly extremely low frequencies (19). Frequency and field strength windows for effects complicate the analysis of the dose effect relationship.

An analysis of such phenomena as calcium efflux, effects on cell membrane receptors and permeability (18) seem to indicate that induced current density and/or possibly peak incuded E field strength may be the parameters, which may serve best to characterize the interactions, and should be taken into account as quantities for dosimetric considerations, proper allowance being made for exposure durations, and pulse characteristics.

REFENCES

1. International Non-Ionizing Radiation Committee of the International Radiation Protection Association (INIRC/IRPA): Interim Guidelines on Limits of Exposure to Radiofrequency Electromagnetic Fields in the Frequency Range from 100 kHz to 300 GHz. Health Physics vol.46, 1984 (in press). 2. ICRP publication No 26, 1977. 3. ICRU report No 33, 1980 4. Czerski, P., and Szmigielski, S.: Proc. 5 European Microwave Conference, Hamburg 1975, p.348. 5. NCRP report No 67, Washington,D.C., 1981. 6. Chernovetz, M.E., et al. : J.Microwave Power 10, 391, 1975. 7. Rugh, R. et al. : in "Biologic Effects and Health Hazards of Microwave Radiation. (P.Czerski et al. eds), Warsaw, Polish Med.Publ. 1974 p.98. 8. Phillips, R.D. et al. Ann. N.Y.Acad.Sci 247, 499, 1975. 9. Justesen, D.R. : Ann.N.Y. Acad. Med. 55, 1058, 1979. 10. de Lorge, J.O. : NAMRL Report No 1286, Naval Medical Research Laboratory, Pensacola, 1982. 11. Way, W.,I. et al.: Bioelectrmagnetics 2, 241, 1981. 12. Spiegel, R.J. et al. : Bioelectromagnetics 1, 253, 1980. 13. Dewhirst M.,W. et al.: Cancer Res. 1984 (in press). 14. Sapareto, S.A., and Dewey, W.C : Int.J.Radiat.Biol.,1984 (in press). 15. Michaelson S.M.: in "Nonionizing Radiation Protection" (M.J.Suess,ed). WHO Regional Office for Europe, Copenhagen 1982,p.97. 16. Olcerst, R.B., et al. : Radiat.Res. 82, 82, 244, 1980. 17. Frohlich, H., and Kremer, F. (eds) : Coherent Excitations in Biological Systems. Springer, Berlin,1983. 18. Swicord, M.L., and Davis, C.C.: Bioelectromagnetics 4,21,1983. 19. Pilla, A.A., et al. J.Biol.Physics, 11, 51, 1983.

ULTRAVIOLET RADIATION SOURCES FOR COSMETIC USE : MEASUREMENT OF INTENSITY AND SPECTRAL DISTRIBUTION

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Introduction

Ultraviolet radiation has been known for many years to be both a benefit and a risk to mankind. During the last few decades a rapid increase in malignant melanoma has been observed probably owing to rising exposure to UV-radiation. This increase in UV-exposure may be attributed to the low protection from sunlight by modern clothing or to the new fashion of tanning either by solar UV-radiation or by artificial UV-sources.

A noxious development concerning UV-induced skin cancer can possibly only be stopped by comprehensively informing the public about the risk of tanning and by supervising the artificial UV-sources for cosmetic purposes. This control of industrially produced UV-radiation emitting devices, may be based on the published standards(1-3) from different national and international organisations dealing with UV-safety and measurement, even though most of those standards are only proposals up to now.

Based on the standard proposed for the Federal Republic of Germany (3) a system for the measurement of UV-radiation was built up and several types of UV-sources were measured and their biological effectiveness calculated.

Materials and Methods

The main part of the measurement device is a spectro-radiometer model 6051 from Roffin Sinar, consisting of a scanning monochromator, a photomultiplier and a wavelength marker unit. The monochromator uses a ruled grating as dispersion element to ensure a constant bandwidth over the whole wavelengths (2.5nm from 200 to 400nm). The radiation is coupled to a fused silica optical fibre by a wide-angle diffusor with nearly cosine response. The other end of the fibre has a rectangular shape, to couple light directly to the entrance slit of the monochromator. The radiometer is linked via an a/d-converter to a small computer (CBM 8032) for data acquisition, manipulation and storage and for the representation of results.

The spectral responsivity of the radiometer, - about which knowledge is necessary to calculate the absolute irradiance values -, is determined by calibration against a UV-standard source described by Rössler (4). For the calibration only the wavelength with high values of radiation (mercury lines) were used. The values of the responsivity at other wavelengths were calculated by an interpolation procedure. The calculations of the biologically effective irradiation and exposure times for threshold doses were based on the data published in "DIN 5031 Teil 10" (3).

The tanning devices which were measured up to now were grouped into different classes representing the different kinds of UV-sources used. Type 1 devices are those using high pressure mercury arcs without a special filter, type 2 devices are equipped with so-called sunlamps which are lamps combining incandescent lamps and high pressure mercury arcs within the same bulb. Sometimes lamps of type 2 are combined with infrared lamps whereas type 1 is always in combination with an IR-emitter. The type 3 solaria use fluorescent lamps for UV generation and type 4 is the sun at a standard sunny day (3).

Results

Examples for the spectral irradiance of different UV-sources are shown in Figure 1 a-c. The great differences of the tanning sources shown in this Figure are also reflected in the biological effectiveness shown in Table 1. In this Table the Irradiance effective in producing erythema I(ery) is compared with that effective for the immediate pigment darkening I (ipd). Obviously I(ery) is in part also effective in forming new melanin and therefore in pigmentation (late pigmentation). I(ipd), however, is meant to be the part of the biologically effective irradiance that is responsible for the "safe" pigmentation without a risk for skin cancer. Therefore and because it is tabulated in the standard (3) I(ipd) should only be considered here. Table 1 also gives an idea of the threshold times for one "minimal erythema dose" T(med) and for one "minimal tanning dose" T(mtd).

Type	I(ery)	T(med)	I(ipd)	T(mtd)
1	188 $\mu W/cm^2$	3 min	610 $\mu W/cm^2$	280 min
2	60 $\mu W/cm^2$	10 min	655 mW/cm^2	250 min
3	6.4 $\mu W/cm^2$	100 min	19 mW/cm^2	9 min
4	18 $\mu W/cm^2$	30 min	4.2 mW/cm^2	40 min

Table 1 Four different types of tanning sources showing great differences in erythemal - I(ery) - , immediate pigment darkening - I(ipd) - effectiveness, times required for a minimal erythema T (med) and for minimal tanning T(mtd).

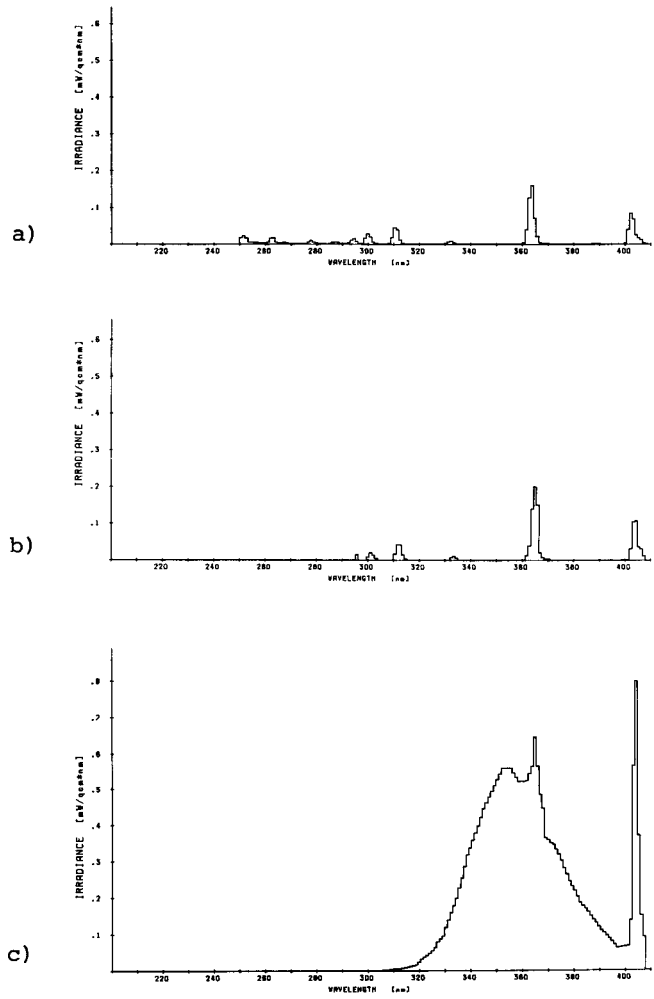

Figure 1

a) Spectral irradiance of an UV-source using high pressure mercury arc, measured at a distance of .7 m from the lamp

b) Spectral irradiance of an UV-solaria using sunlamps at the ceiling of the room at a distance of 1.3 m.

c) Spectral irradiance of a tanning bed using fluorescent tubes, measured directly on the acrylic plate

Conclusions

The type 1 devices which emit a high amount of UV-B (280nm to 320nm) should only be used very cautiously for cosmetic purposes and a therapeutical use should be controled by a physician. Exposure to that kind of radiation is coupled to a very high risk for erythema and long-time exposure may result to a considerable risk for skin cancer.

The type 2 solaria consist of sunlamps mounted together with IR-lamps on the ceiling of the exposure room. This devices want to emulate natural sunlight and, therefore, a considerable amount of UV-B is also emitted. Like the sunlight, this part of the emitted spectrum is also suspected to increase the risk for skin cancer especially in long-time application.

The type 3 devices show little UV-B and therefore they seem to be safe because of nearly no erythemal effectiveness. Furthermore experiments have proven that UV-A alone is not able to produce skin cancer (5). However recent experiments show that large amounts of UV-A in combination with a small amount of UV-B can induce skin tumors in animals (5). Although it is not known wether these results can be applied to man or not, a cancer risk for people using this type of tanning equipment after exposure to sunlight cannot be excluded.

References

1) International Radiation Protection Association. International Non-Ionizing Radiation Committee. "Guidelines on limits of exposure to ultraviolet radiation of wavelength between 180nm and 400nm (incoherent optical radiation)". 1983
2) Department of health and human services. Food and drug administration. "Performance standard for sunlamp product; proposed rule". Federal register 48/99, 22886-91. 1983
3) Deutsches Institut für Normung e.V. "DIN 5031 Teil 10 Strahlungsphysik im optischen Bereich und Lichttechnik"
"DIN 5050 Nichtterapeutische UV-Bestrahlungsgeräte für den menschlichen Körper". Entwurf 1983
4) F. Rössler, "Strahlungsmessungen an einer Quecksilberhochdrucklampe". Ann Phys 34, 1-20, 1939
5) B. Staberg, H.C. Wulf, T. Poulsen, P. Klemp and H. Brodthagen "Carcinogenic effect of sequential artificial sunlight and UV-A irradiation in hairless mice". Arch Dermatol 119 641-3 1983

STRAYFIELD MEASUREMENTS AROUND MICRO-WAVE-AND SHORTWAVE-DIATHERMY DEVICES IN THE FEDERAL REPUBLIC OF GERMANY

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Radiofrequency radiation(RF) with frequencies between 13.56 MHz and 2.45 GHz has been used for many years for therapeutic heating in various deseases including cancer. Besides radio- and TV-stations, radar, industrial dielectric or inductive heaters and microwave ovens these diathermy devices are one potential source of exposure of persons to non-ionizing radiation. The reason is that a considerable part of the therapeutic radiation is not absorbed by the target body-region, but is emitted as "stray-radiation" in the surrounding and thus may expose unintendedly the patient himself and the operator of the device.

Methods

To obtain detailed information about the amount of exposure, we measured the stray-fields around RF-therapy -units both with test persons and with phantoms.In order to have reproducible conditions the systematic measurements were carried out on plexiglass phantoms exposed to the therapy-units most commonly used in Germany, i.e. the Siemens ULTRATHERM 708 S (shortwave, 27.12 MHz), SIRETHERM 609 S (UHF, 433.92 MHz) and RADIOTHERM 306 (microwave, 2450 MHz).The phantoms were filled with saline, with approximately the same conductivity as human muscle-tissue at the different frequencies(27.12 MHz: 5.9mS/cm; 433.92 MHz: 11.7mS/cm;2.45GHz: 18.2mS/cm). Control-measurements with different "patients" indicated differences between phantom and patient ranging from 10 to 50%.Since the variance of the stray-radiation of different patients was of the same magnitude,the results of the phantom measurements are thought to be valid for any patient with a maximum uncertainty of a factor 2.

In the shortwave region(27.12 MHz) the electric and magnetic fieldstrength of the stray-radiation was measured with the AERITALIA field strength measuring equipment TE307 in combination with 2 electric sensors, 14RV and 15RV, and 2 magnetic sensors,16RV and 17RV.

In the microwave region(433.92 and 2450 MHz) the power density of the stray-radiation was measured with the NARDA radiation monitor 8316 and 2 sensors 8321 and 8323.

Measurements

The most important parameters,determining the stray -radiation are the power setting of the device,the type and size of the electrode or applicator, the applicator-phantom distance (d_o) and, of course, the distance (d) between the measuring point and the applicator.

Instead of simulating all possible treatments under the various conditions,we used at every frequency that applicator where we found the maximum stray-field,and studied the influence of the above mentioned parameters on the stray-radiation,to find out "worst case"-conditions, according to which "security-areas" could be proposed.The following measurements were carried out:

- stray-radiation as a function of the power setting for a given distance (d) and applicator-phantom distance (d_o).
- stray-radiation as a function of the applicator-phantom distance for a given distance (d) and power setting.
- stray-radiation as a function of the distance (d) for a given power setting and applicator-phantom distance(d_o).
- determination of places of equal fieldstrength or power density around the applicator in 22.5° steps for different conditions.The exposure limits valid in the Federal Republic of Germany(1) were, of course, of particular interest and used for comparison.In the frequency-range from 30 MHz to 3 GHz these limits are: 100 V/m , 0.25 A/m and 2.5 mW/cm^2 respectively.

Results

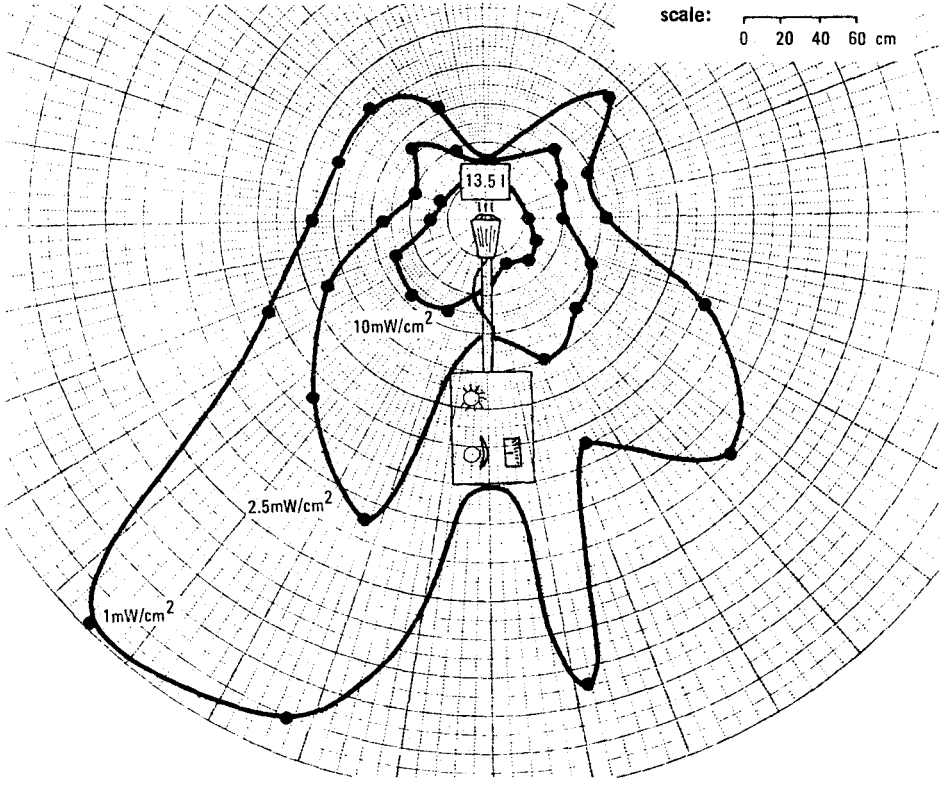
The following results have been obtained:

1. Shortwave (27.12 MHz)
   The maximum stray-fieldstrength results from the capacitor -field electrodes after Schliephake, the E-field being the dominating part of the electromagnetic field in this case. The steps of the power setting 1-8 correspond to a factor of 3-5 in the stray-fieldstrength. Increasing the applicator-phantom distance (d_o) from 0 to 4 cm results again in a four-fold increase of the electric stray-fieldstrength.The exposure-limits are not exceeded in "worst case" (power setting 8, d_o=4 cm) in a distance of 1.5-2m. A significant source of stray-radiation are the unshielded cables,which lead to the electrodes. At the highest power setting we have measured in a distance of 60 cm still 3 A/m and 500 V/m.

2. UHF (433.92 MHz)
   The maximum stray-fieldstrength was measured with the circulatory field director.The steps 1-7 of the power setting make a factor of 6-8 in the power density of

the stray-radiation. Increasing the applicator-phantom distance (d_o) from 1 to 5 cm results in a factor of about 3 in the power density of the stray-radiation, which is highly angle-dependent. In the "worst case" (power setting 7, d_o =3-5 cm), the exposure-limits are not exceeded in a distance of 0.3 - 1 m from the applicator.

3. Microwave (2450 MHz)
   There is usually only one applicator: the circulatory field director. At this frequency interference phenomena due to reflections in the room could be recognized. The power density of the stray-radiation is proportional to the power setting. A variation of the applicator-phantom distance (d_o) between 4 and 20 cm results in a factor of 3-4 in the power density of the stray-radiation, which is again highly angle-dependent. (See figure)



Power density polar diagramme in the applicator-plain

Siemens Radiotherm 306, Cirulatory field director

Power: 100 W

distance applicator-phantom : 7 cm

In the "worst case" (power= 200 W, d_o =7 cm) the exposure limits are not exceeded in a distance of 0.5 - 2.4 m, but there is surely still a dependence on the circumstances in the room due to reflections and standing waves.

Conclusions

The exposure limits are valid only for the technical personal (operator) for exposure times longer than 6 minutes, but not for the patient. For shorter exposure times, as they may occur when the operator controls or changes the parameters during treatment, higher values of exposure are tolerated by the guidelines. However both the operator and the patient should, of course, be protected from unnecessary exposure. Therefore the operator should at least stay at those distances, where the exposure limits are not exceeded or, even better, leave the room, if possible. The patients' non-target tissues, expecially the eyes, gonads and brain should, depending on the kind of treatment, be protected from unnecessary exposure by correct placement of the applicator and, possibly, by using safety goggles (in microwave treatments) or absorbing material. In shortwave treatments non-target body regions of the patient should not come too close to the cables. On the other hand, the manufacturers should attempt to reduce the large stray-fields around the cables. Because of electromagnetic interference problems, it should be clear that two therapies never be made in the same room at the same time.

(1) DIN 57848 Teil 2
VDE 0848 Teil2/...83
Gefaehrdung durch elektromagnetische Felder, Schutz von Personen im Frequenzbereich 10 kHz bis 3000 GHz

OCCUPATIONAL EXPOSURE TO POWER FREQUENCY ELECTRIC AND MAGNETIC FIELDS: RESULTS OF A SURVEY

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INTRODUCTION

The transmission of electrical power by high voltage a.c. overhead lines is well established and some of the associated problems such as noise, interference with radio and television transmission and the danger of flashover are well controlled. Reports[1] of the occurrence of subjective complaints such as increased headache, lassitude,nausea and loss of libido amongst Russian substations workers were published in 1966 and these were followed by other reports[2,3] from the USSR between 1967 and 1972. It was suggested that these symptoms were due to occupational exposure to high electric field strengths in substations.

In spite of the reassuring conclusions of numerous studies, the suspicion remains in some quarters that currents or charges induced by high electric fields, even though they are imperceptible, can be damaging to the health of exposed persons[4]. Four major epidemiological studies of occupationally exposed persons have been undertaken in the past seven years[5-8]; it should be noted however that in some studies the medical examinations did not coincide with the period of exposure, the control groups were not always closely matched and the field strength and exposure, expressed as kilovolt per meter hours, were estimated and not measured.

The aim of this paper is to report the preliminary results of measurements of power frequency electric and magnetic field strengths obtained in a survey of the power substations of the "Ferrovie dello Stato" (the Italian Railways, FS) in order to assess exposure levels and times of service of personnel during the working day. The present survey can be seen in the frame of a more general investigation[9] based on the coupling of environmental and medical data in order to verify the existence of possible relationships between the exposure conditions and the health state of people working in the substations.

TEST SITE AND INSTRUMENTATION

The measurement site was located in the outskirts of Rome, about 12 km northeast of center city. Measurements were made under the operating 220 kV, 50 Hz three-phase line entering the substation and in the area of the substation itself. The area was flat and nearly free from vegetative or geographic anomalies. On the days the measurements were taken, the weather was fair with temperature ranging between 20 and 25 °C.

For the measurements presented here two types of commercially available instruments have been used, the TE 307 model manufactured by Aeritalia, which has probes sensitive to the electric field in the frequency range 20 Hz-100 Hz, and the Hewlett-Packard 428 B model analog voltmeter equipped with a 3529 A model magnetometer probe able to measure magnetic flux density strengths up to 1 mT (10 G) in the frequency range 0-80 Hz with an accuracy of ± 3%. The Aeritalia sensor is an active isotropic and balanced antenna with full scale sensitivity selectable up to 10 kV/m with a calibration accuracy at calibration frequency of ± 0.5 dB and frequency sensitivity and isotropic response of ± 1 dB. The measurement procedure was the following.A 30 m profile line, perpendicular to the center phase of the transmission line, was established using a non-metallic 50 m tape measure. The observer-meteraxis was perpendicular to the transmission line, with

the observer's back toward the line. Tests were made positioning the instruments at 0.7 and 1.5 m above ground level and at least 3 m from the observer in the case of the electric field measurements.

EXPERIMENTAL RESULTS

Figure 1 illustrates the electric field strength profiles obtained at 0.7 and 1.5 m above ground level. Measurements of electric field strength were made with available instrumentation with a reproducibility of ± 10%.

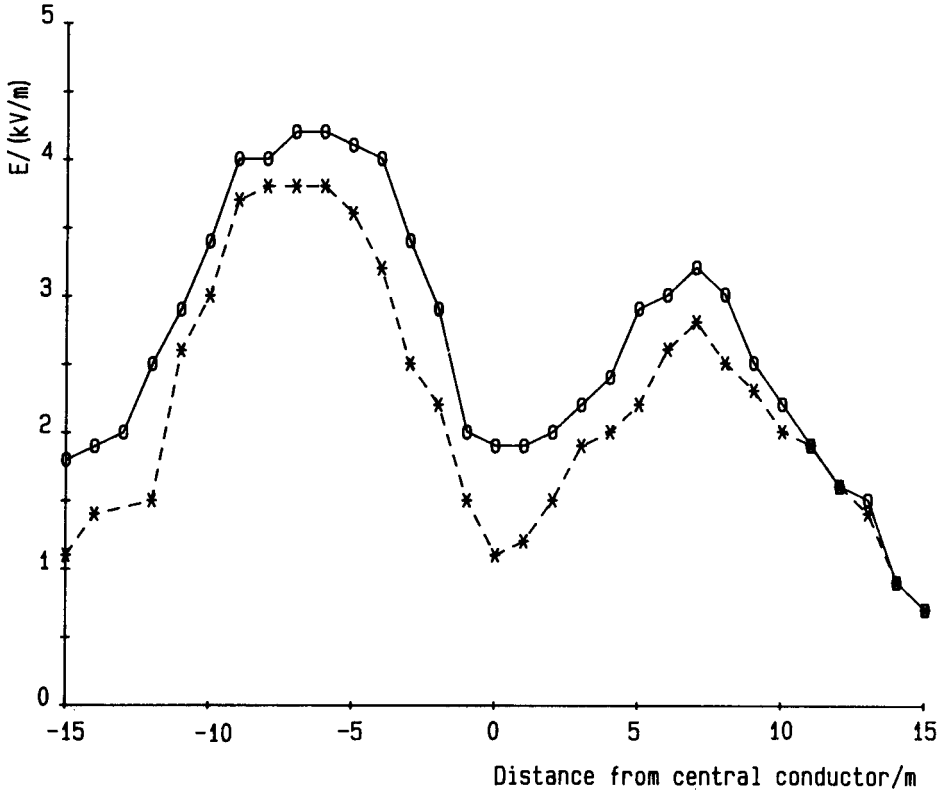

Fig. 1. Electric field measured under a 220 kV transmission line at 0.7 m (--*--) and 1.5 m (-0-) above ground level. Line height at the measurement point about 6 m.

Figure 2 shows a plot of each of the peak magnitudes of the three spatial magnetic flux density vectors. The different phase relations between the currents in the phase wires cause the magnetic field vectors to rotate in space to give rise to the three spatial components. Table 1 shows the typical electric field and magnetic flux density strengths measured at 1.5 m above ground level near different construction parts in a 220 kV substation.

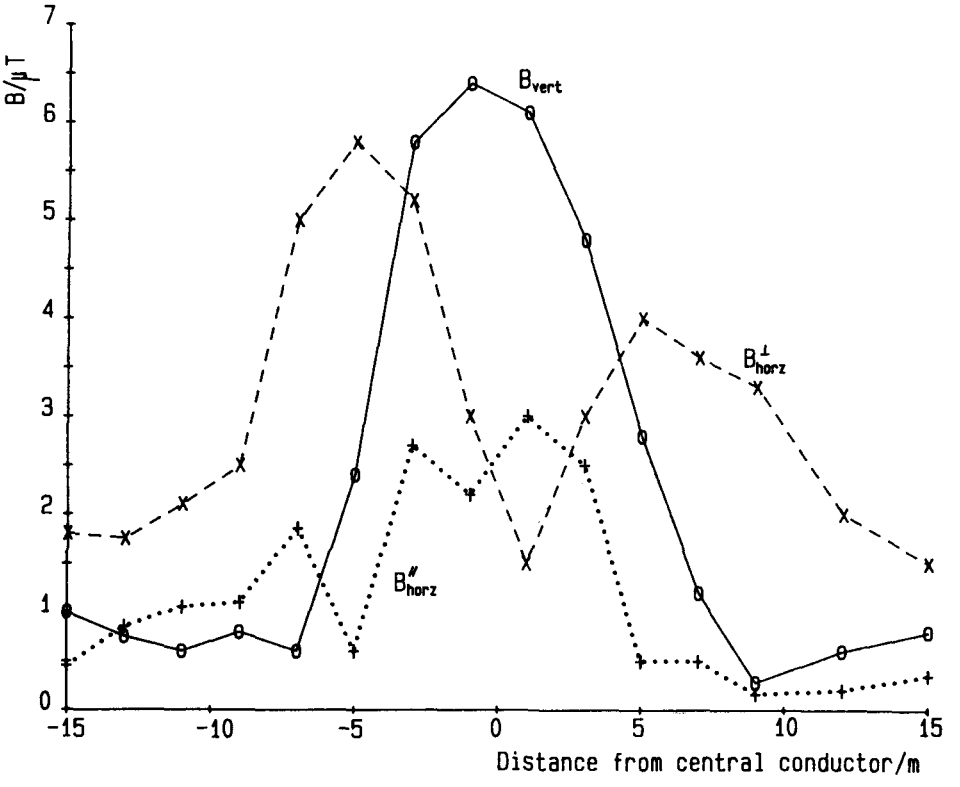

Fig. 2. Profile of three magnetic flux density components under a 220 kV line. Line height at the measurement point about 6 m.

Table 1 - Typical electric field and magnetic flux density strengths in a 220 kV substation measured at 1.5 m above ground level

Construction part	E/(kV/m)	B/μT
Disconnector busbars	1.2 - 4.6	4 - 12.5
Circuit breakers	3 - 4.6	7 - 10.8
220 - 132 kV transformers	1 - 4.2	6 - 15

CONCLUSIONS

Electric field and magnetic flux density strengths have been measured in areas in which access was limited to employees or was normally forbidden for everybody, except in emergency situations. In these areas maximum field strengths ranged between 1 and 5 kV/m for the electric field and between 4 and 15 μT for the magnetic flux density. Workers can be categorized as attending personnel and are exposed

for an average of 3 hours per working day.

On the basis of available experience and these preliminary results it can be concluded that electric and magnetic fields caused by electric power FS substations do not seem to constitute a danger to human health. Due to the chronic,long-term exposure, however, an epidemiological investigation to obtain a final assessment of possible hazards is now under way[9].

REFERENCES

1) T.P.Asanova, and A.I.Rakov, The state of health of persons working in the electric field of outdoor 400 and 500 kV switchyards, Gig.Trud.Prof.Zabol. 10:50 (1966).
2) T.E.Sazonova, Physiological assessment of the work conditions in 400-500 kV open switchyards, Scient.Publ. of Institute of Labor Protection 46:34 (1967).
3) V.P.Korobkova, Yu.A.Morozov, M.S.Stolarov, and Yu.A.Yakub, Influence of the electric field in 500 and 750 kV switchyards on maintenance staff and means for its protection, in "Conference Internationale des Grands Reseaux Electriques à Haute Tension (CIGRE)", Paris (1972).
4) World Health Organization Environmental Criteria Document, "Extremely low frequency (ELF) electric and magnetic fields (up to 300 Hz) with particular reference to power frequencies (50/60 Hz)", WHO Publisher, Geneva (Draft April 1983).
5) E.Malboysson, Medical control of men working within electromagnetic fields, Rev.Gen.Elec. 7:75 (1976).
6) P.F.Roberge, "Study on the state of health of electrical maintenance workers on Hydro-Quebec's 735 kV power transmission systems", Hydro-Quebec (1976).
7) G.J.Stopps, W.Janischewsky, "Epidemiological study of workers maintaining HV equipment and transmission lines in Ontario", Canadian Electrical Association Research Report (1979).
8) B.Knave, F.Gamberale, S.Bergstrom, E.Birke, A.Iregren, B.Kolmodin-Hedman, and A.Wennberg, Long-term exposure to electric fields. A cross-sectional epidemiologic investigation of occupationally exposed workers in high-voltage substations, Scand.J. work environ. and health 5:115 (1979).
9) A.Checcucci, An epidemiological investigation of HV substations workers: study design and preliminary results, in: "Biological Effects and Dosimetry of Static and ELF electromagnetic fields", M.Grandolfo, S.M.Michaelson, and A.Rindi,eds., Plenum Press, New York-London (in press).

RADIATION PROTECTION OBJECTIVES AND PRINCIPLES FOR RADIOACTIVE WASTE

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ABSTRACT

The approach to the special problems of the long term radiation protection principles is described. This includes the concept of probability of an event, the significance of uncertainties in future doses, optimization procedures, etc. All radioactive waste problems should be considered thoroughly with regard to radiological consequences and social and economic factors. The significance of individual and collective doses is discussed. The concepts "cut off" and "de minimis" levels as general terms are not easily applied in practice following the principles of ICRP. However, exemption rules may be defined for various types of low level waste and different routes for disposal.

INTRODUCTION

By 1985 the Swedish nuclear power program which started in 1964 will consist of 12 reactors with a total effect of 9.5 GW_{el}. According to a parliamentary decision the last reactor shall be closed down by the year 2010. The total amount of waste generated during this time will be of the order of 7.000 tonnes spent fuel and 93.000 m^3 low and intermediate level waste. In addition to this, core components and decommissioning waste will be generated. An application for a low and intermediate level waste repository has been approved. Smaller quantities of radioactive waste are also generated in research, medicine and non-nuclear industry.

OBJECTIVES AND PRINCIPLES

General

It is not possible completely to isolate all radioactive waste from man for all time. The necessary degree of containment and isolation of the waste from man depend on a total judgement of the economical, social and political factors. The overall objective for radioactive waste disposal is the protection of man and his environment from the ionizing radiation. The general principles of ICRP, publ. 26, are applicable to radioactive waste. The justification of the waste management should, however, already be included in the justification of the practice causing the waste.

The Solution of Problems; Time Aspects

It is sometimes argued that the solution to a problem would not necessarily need to be good in the long-time perspective but it is left for future generations to solve the long-term problems. That would be their share of the bill for the inherited welfare. There are at least two objections to that. Firstly, there is no guarantee for continued prosperity. Secondly, the application of such a principle to all long term problems would make the future burden of unsolved problems unacceptable, reducing the future margins for expenses and efforts of the future generations to solve their own problems and increase their own prosperity. However, in a short-time perspective it might sometimes be preferable to wait for improved technical

solutions, i.e. extend the storage time before the ultimate disposal.

Individual Doses

The limitation of the individual doses caused by waste disposal requires an identification or definition of a critical group and an assessment of the resulting doses to that group. Sensitivity analysis identifies the significance of various parameters and assumptions and the critical pathways. Future doses and effects are estimated as if future man, habits and society are the same as to-day. Due consideration is given to additive contributions from several practices and sources. The design of nuclear power reactors is based on expected doses below 0.1 mSv a^{-1} to the critical group. For waste disposal it has been found reasonable to judge the acceptability of a planned waste disposal system on the basis of the same principles. The individual doses to the critical group caused by the radionuclides from waste disposal would normally be expected to be below 0.1 mSv a^{-1} and often below 0.01 mSv a^{-1}.

Resulting doses are assessed with some conservatism to ensure that limits will be observed with a reasonable margin. The conservatism follows from the limited knowledge of pathways, living habits etc especially as concerns the future. However, a waste disposal system is established on the basis of best knowledge and most probable events or scenarios. Alternative scenarios should be much less probable or, if not, be taken into account such that the sum of probability-weighted dose consequences is less than the set limit. It is therefore necessary to estimate the probability of each alternative event occurring first.

The application of the annual dose limit to the dose commitment by one-year releases of reactor operation prevents an unacceptable build-up of doses in the future. However, for longlived waste these methods of limiting future individual doses from waste disposal sites are not applicable. The repository should be considered as one source which leaks radionuclides in amounts and rates only depending on the quality of the repository and the waste and on the activity and there is no correlation to the rate of accumalation of the waste in the repository.

Collective Dose Commitments

The collective dose commitment (here called collective dose) is estimated for the purpose of judging the detriment and of optimizing the protection (ALARA). It is assessed irrespective of where and when the doses occur and of the size of the individual doses. For optimization, the collective doses are assessed as realistically as possible and conservatism in assumptions is not intentional. Differential cost benefit analysis is usually applied and an α-value of about 10^4 US$ per manSv is used without discounting.

The interpretation and significance of the assessed collective doses vary depending on the purpose and the uncertainties. It is up to the appropriate authorities to give the collective dose due consideration in the final decision-making. A comparison of the collective doses caused by the various parts of the uranium fuel cycle is not decisive but is of general interest and should be known. The distribution in time of the collective dose is also relevant information.

Increasing uncertainties in the collective dose over a very long time decrease its significance in the judgement of future detriment. The major concern in this case is the individual doses. Options for disposal of long-lived low level waste such as tailings may be acceptable even if there are great uncertainties concerning stability and protective barriers after say 10 000 years and the collective dose after that time might be large. This is because the resulting doses to the critical group are moderate and because it will still be possible to make improvements in the protection at that time.

In the case of high level waste the individual doses require long isolation from the biosphere and at the time the radionuclides appear in the biosphere to any significant extent the resulting individual doses must be low because remedial action is practically impossible. The collective dose may nevertheless be high.

For long integration times the increasing uncertainties in the collective dose make its use for justification purposes less meaningful. In the optimization procedure these uncertainties will cancel out if the various options coincide at a future time as concerns the environmental consequences. This is, for instance, the case with very long-lived radionuclides such as ^{129}I which eventually disperse to the ocean irrespective of earlier containments.

Other Bases for Judgement of Methods for Waste Disposal

If the optimization procedure does not give clear answers, preference is given from the point of view of radiation portection to the option that:

- causes the lowest collective doses
- causes the lowest individual doses
- involves the smallest risk for intrusions and accidents
- involves least institutional control
- involves least uncertainties.

Other bases for judgement and factors to consider for the decisions are

- engineering and quality aspects
- confidence in presented solutions
- public relations considerations.

Exemption Rules

It is not practical nor appropriate to define general exemptions and exemptions must be made for individual cases. Exemptions would be justified in the following circumstances:

- the cost of containment is unreasonable as compared with the cost that is normally involved in avoiding a collective dose equal to that caused by exemption of the waste;
- the annual dose equivalent to individuals in the critical group will be below 10^{-2} mSv with a good margin even considering possible summation of doses caused by various wastes;
- satisfactory assurances have been given that the nuclides and activities are those given in the presumptions and conditions.

Special requirements may be needed for the first phase of the distribution of the waste following the exemption (e.g. that only a certain type of disposal or use is permitted).

Institutional Control

Institutional control is necessary for a limited time after the closure of the waste repository. But in the time perspective of several hundred years and more the principle must be that no institutional control should be necessary.

PRACTICAL IMPLICATIONS

Methods for disposal of nuclear waste are:

- exemption of nuclear waste such as slime from purification plants for water coming from areas outside controlled areas (may be used with other slime as fertilizer), slightly contaminated oil from nuclear power plant turbines (to be burnt with other oil in an oil-fired power station), contaminated condenser tubes or other metal items of great economic value (may be treated as scrap-metal with restrictions on the first use).
- shallow land burial of low level waste on the site (some trash waste and decommisioning waste is appropriate for that method; the requirement is that the waste is covered and the leakage rate from the disposed waste to water is much lower than the limit for reactor sites and that the activity is such that institutional control is necessary for 100 years at the most).
- incineration and subsequent encapsulation of low and intermediate level waste from reactor operation in concrete or bitumen. The waste is then stored and eventually disposed in an underground facility in rock (to be effectively isolated from the biosphere for at least 1000 years).
- sea dumping of low and some medium waste (forbidden by Swedish law).
- land disposal of tailings (stabilized and covered by several meters of moraine, to be effective for thousands of years).
- deep geological disposal of spent fuel and high level waste (at about 500 m depth in rock, isolation from the biosphere by several barriers such as the waste matrix itself, glass, metallic capsulation, low-diffusion barriers, rock. The requirement on certainty is greatest for the quality and integrity of the barriers followed by the models for dispersion in rock, water and the biosphere).

THE NEED FOR INTERNATIONAL COOPERATION

International agreement and cooperation should be encouraged concerning:

- a minimum α-value ("cost" of collective dose)
- methods for the calculation of collective doses (factors of concern are the size of individual doses, time and space)
- upperbounds as limits for various sources
- criteria for choice of sites for waste disposal
- criteria for conditioning of waste
- technical solutions for the treatment, storage and disposal of waste
- control measures
- criteria for an overall optimized solution for the disposal of spent fuel and radioactive waste on a regional and global scale.

THE OBJECTIVES OF RADIOLOGICAL PROTECTION IN THE LONG-TERM MANAGEMENT OF RADIOACTIVE WASTE

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Introduction

An important element in the selection of radioactive waste management strategies is the consideration of radiological protection objectives and requirements specifically derived from them.

Existing recommendations and national regulations on radiological protection generally apply to situations in which exposure of the public can be influenced by control of the source of radiation or radionuclides, by control of operating procedures, or by control of environmental transfer routes. These possibilities cannot be assumed to exist when considering the disposal of radioactive wastes, for which some radionuclides have decay times much longer than any likely period of institutional control. Beyond this period, a regulatory control of radiation exposures, based on continuing surveillance of the source of radionuclides, is no longer possible. Present decisions can, however, have an influence on predicted radiation exposures of populations in the far future. For this reason, it appears necessary to use an approach to radiation protection in which the authorisation of particular waste disposal practices is conditional on predictive radiological safety assessments based on the assumption that control of the source itself or of environmental transfers no longer exists.

Recognising this need, the NEA Committee on Radiation Protection and Public Health and the Radioactive Waste Management Committee set up jointly, in 1982, a group of experts to study and develop consistent general radiological protection objectives for the long-term aspects of radioactive waste disposal. This paper reflects discussions within this group on the key radiological protection issues. Particular attention has been given to discussion of the extent to which the ICRP system of dose limitation may be applied to the long-term aspects of waste disposal.

The Application of Radiological Protection Principles to Radioactive Waste Disposal

The radiological protection principles recommended by the ICRP represent a well established and widely accepted basis for the regulation of current activities. They provide a consistent mechanism for dealing with radiation exposures to workers and members of the public. There are good reasons for adopting the same principles when dealing with hypothetical exposures to the public in the future from today's waste disposal practices. These are a desire for equity in that future generations should be given the same degree of protection that is given to the present population, and a need for consistency in practical application.

However, while the current radiological protection principles can certainly apply to radioactive waste disposal, there is a need for a different emphasis in their practical implementation. Very long half-lives, the possible persistence of low levels of radiation exposure over long periods of time, and the large numbers of people potentially involved throughout the generations suggest that particular attention should be given to the collective impact of a waste disposal practice, which is a fundamental element for the application of the principle of optimisation of protection. However, any assessment of collective radiological impact in the far future is affected by large intrinsic uncertainties, which make very precarious the possibility of a clear-cut discrimination between different waste disposal options based on an optimisation analysis as suggested by the ICRP. Moreover, several factors other than radiation detriment, such as political, social, economical and technological are likely to be predominant in decision-making concerning waste management.

For these reasons, optimisation of protection in waste management is likely to play a role less important and decisive than envisaged by the ICRP in its recommendations. On the contrary, the protection of individuals may give rise to clear and easily assessed constraints on the choice of disposal practices. Therefore, the limitation of individual risks will be the dominant element in the radiological protection systems for radioactive waste disposal.

Objectives for Restriction of Individual Detriment

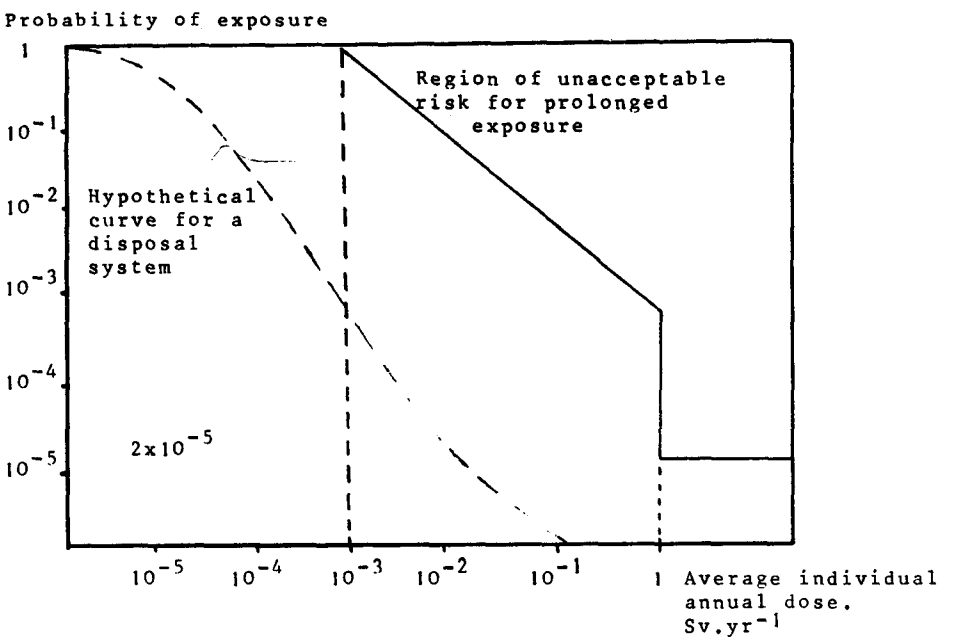
The concept of dose limitation as defined in ICRP recommendations and national regulations requires the implementation of a system of compliance and enforcement. This concept is not directly applicable to the management of long-lived radioactive waste. Present decisions on waste disposal options will have an influence on radiation exposure of populations in the far future, when compliance with any present dose limitation system cannot be enforced or directly demonstrated. For this reason, it appears necessary to use a different concept of a limit, where it is understood simply as a restriction for the planning, design and licensing of waste disposal systems. The authorisation for a particular disposal option should then be conditional on a predictive radiological safety assessment, the results of which are consistent with this limit or objective.

The radiological impact of radioactive waste disposal depends on events and processes which may cause a release of radionuclides into the environment or influence the rate of release or transport through it. Some of these events and processes are certain to occur, others have time-dependent probabilities of occurrence. In order that all events and processes be taken into account on a rational basis, it is suggested that the individual's protection should be based on a limitation of the risk to health, rather than of the level of radiation dose. The risk to health would be defined as the product of the probability of exposure at a particular annual dose and the probability of health effects arising from that annual dose. From the point of view of the protection of the individuals, waste disposal practices should then be judged againt an individual risk limit corresponding to the risk associated with current ICRP dose limits. For future exposures of

limited duration, a risk limit objective of 10^{-4} per year corresponds to the individual dose limit of 5 mSv per year. A maximum risk limit of 2×10^{-5} per year corresponds to the objective of 1 mSv per year suggested by the ICRP for scenarios where the exposure is expected to persist over a decade or more in the lifetime of the individuals concerned.

The practical interpretation of a risk limit objective is relatively straightforward in the range of annual doses associated only with stochastic health effects. The relationship between dose and the probability of health effects can be taken as 2×10^{-2} Sv^{-1}, a rounded value consistent with ICRP assumptions. It is extremely unlikely that an individual would receive from a waste disposal practice such a high annual dose that non-stochastic health effects would be of concern, but for completeness and prudence the assumption can be made that annual doses of 1 $Sv.yr^{-1}$ or greater would give rise to non-stochastic health effects. A risk limit would then be interpreted as a limit on the probability of prolonged exposures at 1 $Sv.yr^{-1}$ or above.

An indication of the boundary of unacceptable individual risks can be represented graphically as in the figure below. The boundary in this figure represents the limit on probability of exposures at various annual dose rates, consistent with a maximum risk objective of 2×10^{-5} yr^{-1} for prolonged exposures. The discontinuity in the boundary of unacceptable risk at 1 $Sv.yr^{-1}$ arises simply because the adopted conversion factor from annual dose to health risk changes from 2×10^{-2} Sv^{-1} for stochastic health effects below 1 $Sv.yr^{-1}$, to unity for non-stochastic effects above this value.



The figure also shows a cumulative probability of exposure-versus-individual dose rate relationship that can be generated in a predictive assessment of the radiological impact of a disposal facility. A disposal option would not be acceptable if any part of the calculated cumulative probability-versus-dose curve lay within the area of unacceptable risk.

Optimisation of Radiological Protection

The principle that all exposures should be kept as low as reasonably achievable can also apply to radioactive waste disposal. However, in practice, the techniques of optimisation of radiation protection may not lead to clear preferences between alternative options. Rather, they are likely to represent just one input in a decision-making process using multi-attribute analysis, where social, economical and other factors play a more decisive role.

A major limitation of radiological protection optimisation using cost-benefit analysis in waste disposal is the impossibility of making truly realistic estimates of collective doses in the far future. Apart from the calculation of uncertainties, there are diverging views among experts about the best approach to assess incomplete collective dose commitments on a very long time scale. Some experts suggest an individual dose cut-off to exclude low individual dose components, others favour a truncation in time. It is to be noted that the former approach would be only in line with current ICRP recommendations if the component of collective dose which is excluded is small and quantifiable. A wider agreement exists about the validity of truncating the integration of collective dose in time at a point beyond which calculational and other uncertainties do not allow alternative disposal options to be distinghished.

As far as the valuation of radiation detriment is concerned, there are proposals to apply discounting techniques to the cost of detriments in the far future. Other experts seem to prefer a less formal approach to valuation of far future detriments, whereby the intrisic and substantial uncertainties attached to future collective detriments would justifiy the allocation of less resources to ensure protection against future detriments than might be justified for present detriments.

The debate is still open on all these questions and it may well be that different solutions are eventually adopted by different countries in the detailed application of the general radiological protection objectives briefly described in this paper.

WASTE MANAGEMENT AND ALARA; A PRAGMATIC VIEW

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The principle of ALARA is a cornerstone of radiological protection. In the workplace, implementation is usually straightforward, is well estabilshed by tradition and often can be formalized. In the environment, applied to the management of long-lived radioactive wastes, implementation presents many problems some of which appear to be insoluble. Large contributions to population doses may be received many millenia in the future and may comprise mainly individual doses received at very low rates. Uncertainties in the dose assessments obscure real differences in the short term that might exist between alternative management strategies. There is little expectation that the reliability of dose prediction over long time scales can be significantly improved in the foreseeable future. Unless resolved, these difficulties prevent a strong radiological input to decisions on waste management that have large economic implications.

This paper appraises a pragmatic approach to implementing ALARA. A reasonable measure of radiological impact for distinguishing between options in a cost-benefit analysis is taken as that part of the population dose received over a short enough time scale (i.e., a few hundred years) that sufficiently reliable predictions are permitted. Also excluded for this purpose (though not necessarily in other considerations) would be that part assessed for populations where individual rates are insufficient to be considered at all significant by individuals.

The difficulties in this approach and qualifications on its application are discussed, as are the implications for waste management strategies in general and for environmental research. Concluded is that an approach along the lines suggested is a way of ensuring that requirements for protection from risks associated with radiation can have a significant input to the choice of management options.

FRG-PROJECT FOR RADIOLOGICAL ACCIDENT RISK ASSESSMENTS OF THE BACK-END OF THE FUEL CYCLE

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The German concept for the back-end of the fuel cycle is based as in many other countries on completing the nuclear fuel cycle by reprocessing, refabrication of MOX-fuel elements and disposal of radioactive waste. The produced radioactive waste and specifically the HLW is to be disposed of in the FRG in geological formations such as salt. At the same time pursuant to the decision of the Federal Government and of the State's Governments dated September 28, 1979, also the direct disposal of spent fuel is to be investigated as a variation to the concept. A political decision on both disposal possibilities will follow in the middle of the 1980's.

For the performance of these assessments, safety related questions will play an important role. Therefore, as early as 1978, the Federal Minister for Research and Technology decided to complete his activities in the field of safety studies for the back-end and to gather them in one project, called PROJECT SAFETY-STUDIES ENTSORGUNG (PSE).

Within PSE the following pathways of radionuclides under other-than-normal conditions will be considered:

1. Release of radionuclides under extreme conditions in on-surface facilities and during transportation of radioactive waste.

2. Release of radionuclides after water intrusion in a deep underground repository and migration of contaminated solutions in groundwater-bearing overlaying rocks.

3. Transport of radionuclides in the atmosphere; exposure by inhalation and external radiation from clouds.

4. Soil contamination by radionuclides carried by rain and wind with partial seapage into the water supply.

5. Transport of radionuclides within the food chain; exposure by oral consumption and external radiation from soil.

6. External radiation from transportation of radioactive waste under non-accident conditions.

Figure 1 contains these six pathways.

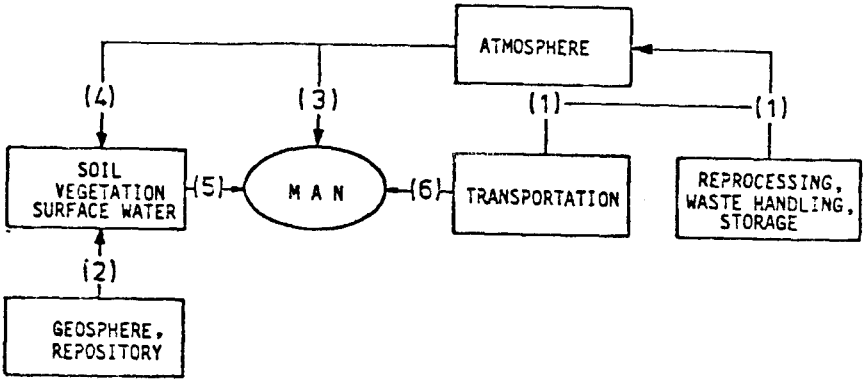

Fig. 1

The PSE objectives to aid the Federal Government in preparing decisions as regards the waste management strategies differ from other safety analytical work which is being carried out within the framework of specific licensing procedures by the future implementing organizations.

The PSE-project is scheduled for two periods. The first one, until June, 1981, was focused to develop and to implement the methodology for a safety analysis of waste management installations. These developments were required because no commercial facilities are in existence. So the already available methodology for reactor safety studies was to be adapted for the analysis of designed plans and technical concepts. In the second period of the project (PSE II) the developed analytical technics will be tested, based upon available concepts and data, and will be deployed for a safety assessment of the overall back-end system.

About 50 men per year are working on the seven-years-project. The yearly budget is approximately seven millions West-German Marks. More than 20 research groups from universities, research centres and the industry are directed by a board of 9 scientists. They contracted a staff at the Hahn-Meitner-Institute at Berlin with carrying out the project planning, project coordination and management. The project will be finished by the end of 1984.

Subprojects

a) Safety analysis of reprocessing, waste treatment and storage

In a preliminary study those accidents were identified by a deterministic approach which could lead to major consequences due to possible physical processes and/or to the considered nuclear inventories, as for example critical accidents in the desolver or red oil explosion. These accidents are subsequently analysed by probabilistic risk analysis methods (modelling the sequence of physical and chemical processes, fault trees and release trees).
The technological basis for the study are concept papers from 1981 for a German reprocessing plant with a capacity of 350 tons HM per year. The results will be expressed as expectation values for release quantities and frequencies in Bq/year. The multitude of accident possibilities in the less sensitive parts of the reprocessing technology can only be qualitatively estimated as a rule, as regards its contribution to overall risk.

The release quantities of all relevant accidents are input data for the calculation of the total human radiation exposure on an individual or collective dose basis. This calculation is carried out according to the International Commission on Radiological Protection (ICRP) guideline No 30. The transport models for atmospheric and food-chain migration of radionuclides take influences of several parameters statistically into consideration. These parameters are for example weather conditions and seasons. The calculation will be done with real data from three different model sites within the FGR so that an evaluation of the influence of different sites on the calculated risks may be possible.

The results of PSE will be comprehensively expressed in Farmer diagramms in the form of a complementary cummulative individual or collective dose distribution function completed by qualitative statements.

b) Safety analyses of the transportation of radioactive materials

The methodical approach in this subproject is similar to that for accidents in surface facilities. Additionally the risk by transportation without accidente will be submitted. For the calculation of radiological consequences of transport accidents so called LAGRANGE-models will be used simultaneously with GAUSS-models as well. LAGRANGE-models simulate the migration of singular radionuclides in the atmosphere and therefore make it possible to consider all the conditions of releases near to soil.

c) Long-term safety analysis of a repository for radioactive waste

Endangering environment near to a repository after sealing the mine is thinkable only if groundwater will get into contact with the disposed waste. Therefore the safety analysis will start with the identification of possible events which may lead to an intrusion of solutions to the waste canisters. An evaluation of the probability that such events could happen, were forgone because the evaluation of a geological system, as opposed to a technical system, is connected with considerable uncertainties.

Therefore we assume this initial event to be given. Subsequently the consequences of the water intrusion will be calculated. In doing this the behaviour of the various technical and natural barriers of the disposal system will be mathematically described for the case of water intrusion as realistically as possible. If input data or if the knowledge of considered effects are insufficient the safety of a barrier is, in a conservative approach, underestimated rather than overestimated.

The possible radiation exposure for an individual has been determined using the directives included in ICRP 30. Possible pathways for exposure are assumed to be drinking water and food-chain, taking into consideration artificial irrigation and consumption of these products at the production site. The results are a 50-years-committed dose equivalent dosis following a one-year on-site drinking water and food consumption where the maximum nuclide concentration is found.

The basis for an assessment of the results will be the natural background radiation of the site as well as the values of section 45 of the German radiation protection regulations.

Status of the project

The results from the various sub-studies are expected during the first semester of 1984. They will show examples concerning the instrumentarium for safety analysis for the planned back-end systems and the weak points of where further development would seem appropriate. Despite the difficulties which arrise primarily in the choice of suitable data such as for the reliability of barriers, the results can be used for the assessment of the different barrier's behaviour during improper operation. The statements concerning waste management safety, as a whole, can be regarded to be sure enough to be used in the political assessment of waste management strategies.

The preliminary results at hand have shown in the mean time that

- the radiation risk resulting from accidents during the reprocessing is no larger than that from normal operation. This would be approximately 10 mrem/year in a reprocessing plant dealing with 350 metric tons HM per year respectively at approximately 1 mrem/GWyear,

- the radiation risk during normal transportation conditions with radioactive wastes on road or by rail will be approximately the same,

- in assessing long-term safety of a disposal site in a salt dome there is a proved instrumentarium available. Statements concerning the safety at the Gorleben site will become even more sure the more site specific data will be accumulated in the next few years during the underground exploration of the salt dome.

THE ESTABLISHMENT OF DE MINIMIS LEVELS OF RADIOACTIVE WASTES

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INTRODUCTION

The working assumption currently accepted for radiological protection purposes, is that the probability of occurrence of stochastic health effects is directly proportional to the level of radiation exposure received, without threshold. This implies that a finite level of risk is associated with any increment of dose no matter how small and that the incidence of stochastic effects in irradiated populations is proportional to the collective dose received, irrespective of its distribution among affected individuals. This notwithstanding, there has been a protracted debate within the radiological protection community, as to whether or not it is a reasonable procedure to sum very small individual doses, additional to those accumulated from natural background, in the calculation and use of collective dose estimates.

This issue has assumed considerable importance in the field of radioactive waste management, as much of the collective dose arising from the dispersion of effluents or low-level solid waste disposal, is delivered at very low individual dose rates. A widespread feeling has emerged that this procedure may over-emphasise the significance of low level exposure; promoting undue public anxieties to waste management practices resulting in essentially trivial levels of risk, and diverting limited regulatory resources which could be better employed in other, higher risk, areas. It has therefore been suggested as an alternative, that a de minimis level of individual dose or risk should be established, below which, there would be no further need for concern on the part of regulatory authorities. This concept of de minimis is under active consideration internationally, by a number of competent authorities and, for example, seems likely to feature in forthcoming regulations by the US Nuclear Regulatory Commission[1]. The purpose of this paper is to examine this concept and its implications, and to describe NRPB proposals for establishing de minimis dose levels in the UK[2].

PREVIOUS APPROACHES TO DE MINIMIS

Individual Assessment

The concept of an insignificant level of risk is by no means new, and proposals on this topic were first considered at NRPB in 1977[3]. In a survey of comparative risks experienced by the population, Webb and McLean concluded that an annual probability of death of the order of 10^{-6} y^{-1} is not taken into account by individuals in arriving at decisions as to their actions. Considering the possible multiplicity of sources of radiation exposure, the annual individual dose of the order of 10^{-4} Sv y^{-1} corresponding roughly to this insignificant level of risk was reduced by a factor of 100 to 10^{-6} Sv y^{-1}, so that if an individual was exposed to a large number of different sources each of which had been ignored, his aggregated dose would not exceed the de minimis level of 10^{-4} Sv y^{-1}.

Similar conclusions based on risk comparisons with other activities or variations in natural background exposure have subsequently been reached by other authors and bodies[4,5]. A general consensus appears to have emerged, that a risk level of the order of 10^{-6} to 10^{-7} y^{-1} is of no significance from the point of view of an affected individual. However, even where due allowance is made for potential multiplicity of radiation sources, there remains the question of application of an

individual related de minimis dose level, by itself, to source related assessments involving irradiated population groups.

Source Related Assessment

Source related aspects of population exposure are treated by ICRP within its system of dose limitation, by application of a cost benefit analysis procedure[6]. The benefits from reduced radiation exposure have to be compared directly with the costs of protective measures, requiring a monetary valuation of radiation-induced health detriment, which is generally expressed as a cost per unit collective dose. The adoption of de minimis individual dose levels within cost benefit analysis could then be achieved either by excluding doses below this level from assessments of collective dose, or by assigning this portion of collective dose estimates a zero valuation for use in cost benefit analysis. However, the net effect of either mechanism would be to ignore small individual doses for decision-making purposes; reflecting the view that if the doses involved are of no significance to affected individuals then it does not matter how many such doses are added up, the resultant collective dose is of no significance to society.

The ICRP, among others, do not accept this view. As implied by their concept of "objective health detriment"[6], they suggest that the societal health consequences of say, one man sievert, is the same whether it arises in a population of one hundred thousand people each exposed to 10^{-5} Sv, or a population of one hundred million people exposed to 10^{-8} Sv. NRPB has adopted a similar line in developing its formal advice on the optimisation of protection of the public[7]. For very small individual doses, below the insignificant levels defined above, a number of alternative valuations of collective dose were initially considered, including a 'de minimis' option of a zero cost. However, this option was rejected in favour of a finite value, based on lost economic output and health care costs associated with the health detriment statistically predicted per unit collective dose, at all individual dose levels. This is reflected in the formal costing scheme by a baseline valuation of £2000 per man Sv, which may be taken to correspond with the ICRP formulation for the cost of objective health detriment, and is applicable to all annual individual doses in the range from zero to 5.10^{-5} Sv[7].

NRPB PROPOSALS

Although NRPB have rejected the idea of applying de minimis individual dose levels in isolation, it has been increasingly recognised that the provision of data for optimisation studies may be costly to obtain in terms of time and resources. It might therefore be argued that where in the absence of further protective measures, the residual individual doses and the collective dose commitment are sufficiently small, the cost of performing the optimisation may in itself, outweigh any potential reduction in health detriment costs. In such situations the rigorous use of cost benefit analysis would not be warranted, and the initial levels of exposure may be excluded from further consideration, not because they are of no concern per se, but because they are optimal. Thus in this context, the concept of de minimis reduces to nothing more than a specific form of ALARA appropriate for trivial radiological problems.

A primary implication of this approach is that each radiological problem must initially be assessed as if it were to be formally optimised. Practical experience in the UK indicates that the minimum cost of a formal optimisation exercise concerning routine releases of radioactivity or disposal of solid wastes, is likely to be in the range of £10^3 - £10^4, and this may be used as a basis to define a de minimis collective dose commitment. Using the NRPB valuation of £2000 per man Sv this leads to a de minimis collective dose of the order of one man Sv. However, the

criterion of de minimis collective dose alone is insufficient and must be accompanied by a de minimis individual dose proposed at 5 10^{-6} Sv y^{-1}, and applicable to the maximum future annual dose to members of a critical group arising over the release lifetime of the source. This will require some care in defining the radiation source under consideration, so that the totality of the operation or source is addressed, with particular regard to the build up of doses over time.

Future Detriment

In its formal advice on optimisation, NRPB advocated the principle of discounting future detriment costs arising from present-day practices, recommending a range of annual discount rates from zero, which assigns the same weight to doses whenever received, up to a value of 3% per year[7]. This precludes the unique definition of a de minimis collective dose commitment since it requires the minimum cost of optimisation to be compared with a discounted 'present value' of health detriment costs which is a function of both the temporal distribution of collective dose and the discount rate applied. If cautiously a zero discount rate is employed, a collective dose commitment of less than one man Sv, arising over the operating lifetime of a given radiation source, will be automatically regarded as de minimis. Alternatively, where finite discount rates up to 3% per year are applied, the present value of health detriment costs should be compared against a de minimis level of £2000. In both cases, as before, the individual levels of dose must additionally be less than 5 10^{-6} Sv y^{-1}.

PRACTICAL IMPLICATIONS

The intent of these proposals is to reduce regulatory efforts in the optimisation of protection for trivial radiological problems. Having specified the radiation source under consideration, there will be a sequential examination of the associated individual and collective doses and possibly, the discounted present value of health detriment costs; all of which would be required in any case for optimisation purposes. It is envisaged that simplifying and cautious assumptions will initially be employed in dose calculations, to be replaced by more rigorous environment modelling results if necessary. But in many practical situations, having determined that the maximum future annual dose to the critical group is less than the de minimis individual dose of 5 10^{-6} Sv, and with knowledge of the half-life and residence time in the biosphere of the nuclides involved, it may be readily established that the collective dose commitment does not exceed one man Sv. Both of the de minimis dose conditions will then be fulfilled, and no additional regulatory consideration will be required.

Furthermore, in such cases where the overriding constraint involves individual doses to critical groups, it will be possible to establish de minimis levels of activity or quantities of nuclides, in an analogous manner to the calculation of generalised derived limits corresponding to ICRP dose limits. The release or disposal to the environment of such derived de minimis quantities might then be authorised without the need for explicit dose calculations, further reducing demands on regulatory resources. This procedure and a range of possible outcomes can be notionally illustrated with regard to the four nuclides listed in the table below, drawing on recent NRPB calculations of the collective dose commitment associated with generalised derived limits of discharge to atmosphere[8].

Nuclide	30 year operating lifetime discharge level (Bq s^{-1})	collective dose commitment (man Sv)	Cost of health detriment (£) zero discount rate	Discounted at 3% per year
Kr-85	$4.05\ 10^{8}$	$2.86\ 10^{1}$	$5.72\ 10^{4}$	$3.22\ 10^{4}$
I-129	3.70	$2.26\ 10^{1}$	$4.52\ 10^{4}$	$2.95\ 10^{2}$
Cs-137	$3.46\ 10^{1}$	$4.38\ 10^{-1}$	$8.76\ 10^{2}$	-
Pu-239	$3.52\ 10^{-1}$	$1.60\ 10^{-1}$	$3.20\ 10^{2}$	-

The table presents results of a derived level of discharge for each nuclide corresponding to the de minimis individual dose of $5\ 10^{-6}$ Sv y^{-1}. The total collective dose commitment is for the discharge continuing over 30 years, and of health detriment costs valued at £2000 per man Sv. It can be seen from this that such discharges of caesium-137 and plutonium-239 clearly satisfy the de minimis collective dose commitment of 1 man Sv, while those of krypton-85 and iodine-129, both of which are globally dispersed, exceed this criterion by approximately a factor of 25. However, given its extremely long half-life, the discounting of health detriment costs associated with iodine-129 discharges has a substantial impact; reducing the present value relative to undiscounted costs for annual discount rates in the range of 1 - 10% by about two orders of magnitude[9]. On this basis, the discounted health detriment cost for iodine-129 is found to be well below the proposed de minimis health detriment cost of £2000, and would therefore preclude the need for any further optimisation where finite discount rates are to be applied. In contrast to this, the results for the short-lived krypton-85 are relatively unaffected by discounting[9] and exceed the de minimis health detriment cost, although further reduction of the discharge may still not be warranted on the basis of an optimisation study.

REFERENCES

1. Baker R.E., Cool W.S. and Mills W.A., NRC revision to 10 CFR Part 20 (1983).

2. The Establishment of De Minimis Levels of Dose. Chilton, NRPB (to be published).

3. Webb G.A.M. and McLean A.S., Insignificant levels of dose : A practical suggestion for decision making. Harwell, NRPB-R62 (1977).

4. See Davis J.P., The feasibility of establishing a "de minimis" level of radiation dose and a regulatory cut-off policy for nuclear regulation. General Physics Corp., Maryland, US, GP-R-33040 (1981).

5. IAEA, De minimis concepts in radioactive waste disposal, IAEA-TECHDOC-282, Vienna (1983).

6. ICRP, Cost Benefit Analysis in the Optimisation of Radiation Protection. ICRP Publication 37, Annals of the ICRP, 10, (2/3) (1983).

7. Cost Benefit Analysis in Optimising the Radiological Protection of the Public. A Provisional Framework. Chilton, NRPB, ASP4 (1981).

8. Jones J.A. and Kelly G.N., The estimation of collective dose from airborne effluents and its relationship to critical group doses (to be published).

9. Fleishman A.B. and Clark M.J., Evaluating future detriment from radioactive discharges. Chilton, NRPB-R132 (1982).

UNE METHODOLOGIE POUR L'EVALUATION DES EXPOSITIONS DU PUBLIC LIEES AU STOCKAGE A FAIBLE PROFONDEUR DES DECHETS RADIOACTIFS DE FAIBLE ACTIVITE

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1 - Introduction

La France a choisi de stocker définitivement dans le sol, à faible profondeur, les déchets radioactifs de faible ou moyenne activité à vie courte. Depuis 1969, ces déchets sont stockés au Centre de la Manche, placé sous la responsabilité de l'Agence Nationale pour la Gestion des Déchets Radioactifs (ANDRA) qui est chargée d'appliquer au plan industriel la politique de gestion des déchets radioactifs décidée par le gouvernement.

Le présent mémoire a pour objet d'exposer la méthode mise au point par l'ANDRA avec le concours de l'Institut de Protection et de Sûreté Nucléaire et d'autres organismes spécialisés (notamment le Bureau de Recherches Géologiques et Minières et l'Ecole des Mines de Paris) pour évaluer les conséquences radiologiques potentielles maximales pour le public du Centre de la Manche et, d'une façon générale, de toute installation de ce genre.

2 - Objectifs et options techniques de sûreté

Les objectifs de sûreté assignés à tout centre de stockage sont :

- la protection des travailleurs et de la population actuelle et future
- la limitation de la durée de surveillance

Ceci conduit à diviser la vie d'un stockage en trois phases successives :

- la phase d'exploitation, pendant laquelle le stockage est construit, rempli, puis fermé
- la phase de surveillance, au cours de laquelle le centre est surveillé et a un accès contrôlé
- la phase de banalisation, pendant laquelle le centre ne fait plus l'objet de contrôles et est rendu à un usage normal. Le début de cette phase doit intervenir au plus tard 300 ans après celui de la phase précédente.

Les options techniques retenues pour atteindre les objectifs consistent à :

- limiter les quantités de radioéléments admises dans le stockage
- isoler les matières radioactives de l'environnement humain jusqu'à ce que leur radioactivité (due principalement à des émetteurs bêta ou gamma à période courte ou moyenne) ait pratiquement disparu par décroissance.

3 - Caractéristiques du stockage

Le stockage contient actuellement 200.000 m^3 de déchets. Au moment de sa fermeture, il pourrait en contenir 400.000 m^3 dont 20 % de très faible activité non enrobés en fûts métalliques et emballages assimilés et 80 % de faible ou moyenne

activité enrobés ou insolubilisés, en fûts métalliques, blocs ou monolithes de béton. Les colis de déchets sont placés soit dans des tumulus (empilements de colis dont les vides sont comblés par un matériau de remplissage) soit dans des monolithes de béton. Les ouvrages sont construits sur des dalles en béton comportant des caniveaux qui permettent de collecter les eaux d'infiltration éventuelle. Une fois remplis, ils sont protégés contre l'eau de pluie d'abord par une couverture provisoire de terre argileuse, puis par une couverture définitive imperméable comprenant, de l'extérieur vers l'intérieur, une couche de terre arable recouverte de végétation, une couche drainante et une couche d'argile. Deux réseaux indépendants permettent de collecter les eaux superficielles : le réseau pluvial qui recueille les eaux de ruissellement et le réseau séparatif qui collecte les eaux d'infiltration éventuelle dans les ouvrages jusqu'à des puisards et à un bassin où elles sont contrôlées.

La sûreté du stockage repose essentiellement

- sur les propriétés de confinement des colis et des ouvrages et dispositifs annexes pendant les phases d'exploitation et de surveillance
- sur la limitation des activités stockées et les propriétés du site (capacité de rétention et de dilution) pendant la phase de banalisation.

4 - Evaluation des risques radiologiques

Les conséquences radiologiques du centre concernent l'exposition externe des travailleurs pendant l'exploitation et l'exposition interne éventuelle du public liée au transfert des radionucléides par l'eau ou par l'air dont il sera question ci-après.

4.1. - Transfert des radionucléides par l'eau

Le transfert des radionucléides par l'eau est permanent pendant la phase de banalisation, à partir de laquelle on suppose que les ouvrages de stockage et leurs dispositifs annexes perdent brusquement leurs propriétés mécaniques et leur fonction d'isolation. Il a été modélisé en combinant un ensemble de paramètres explicités ci-après et ses conséquences radiologiques ont été évaluées.

Pendant les phases d'exploitation et de surveillance, le transfert des radionucléides par l'eau ne peut être du qu'à des évènements accidentels. Les conséquences de ces derniers peuvent être calculées à l'aide du modèle précité à partir d'hypothèses sur les débits et les durées des fuites.

Les paramètres qui interviennent dans le modèle sont

a) l'inventaire radioactif du stockage
En vue de définir la capacité radiologique du centre, un inventaire radioactif maximal hypothétique du stockage terminé compatible avec les prescriptions règlementaires a été retenu comme base de calcul.

b) l'infiltration de l'eau de pluie
Le taux d'infiltration maximum de la couverture a été pris égal à celui du terrain d'origine pendant les phases d'exploitation et de banalisation et cent fois plus faible pendant la phase de surveillance.

c) le vieillissement des colis
Un colis est considéré comme lixiviable dès que son conteneur est détérioré. Le retard à la lixiviation ("temps de latence") a été évalué pour deux familles de conteneurs : les fûts métalliques et emballages assimilés, les blocs et les monolithes de béton. Le vieillissement des colis ayant un caractère aléatoire du fait de la multiplicité des facteurs dont il dépend, la

distribution des temps de latence, de chaque famille de conteneurs a été représentée par une loi gaussienne.

Pour tenir compte d'aléas de fabrication et de manutention, on a considéré par ailleurs qu'un petit nombre de colis pouvaient être endommagés dès l'origine.

d) taux de lixiviation

Le taux de lixiviation d'un colis est le rapport de l'activité entraînée par l'eau en un an à l'activité résiduelle du colis au moment de la lixiviation. Les taux de lixiviation ont été supposés constants et proportionnels à la quantité d'eau infiltrée pendant chaque phase correspondant à un état différent de la couverture. Ils ont été déterminés en fonction des caractéristiques physico-chimiques des déchets en tenant compte,

- pour les déchets contenant des radioéléments naturels, des solubilités des combinaisons chimiques concernées
- pour les déchets non enrobés, des observations sur l'entraînement par l'eau de la radioactivité non fixée (première phase) et des coefficients de partage des radionucléides entre l'eau et la terre mélangée aux déchets (dernière phase)
- pour les déchets enrobés, des résultats d'expérience en vraie grandeur (première phase) et des coefficients de partage des radionucléides entre l'eau et le béton réduit en poudre (dernière phase).

e) transport dans le terrain

Les quantités des radionucléides atteignant l'exutoire (ruisseau Sainte Hélène) ont été calculés en fonction du temps au moyen d'un modèle utilisant les valeurs moyennes des paramètres mesurées sur le site. Quelques mesures ayant suggéré la possibilité d'une circulation rapide de l'eau à certains endroits, on a supposé q'une petite fraction de l'activité quittant le stockage arrivait directement dans le ruisseau sans rétention par le terrain.

f) utilisation de l'eau

On a considéré d'une part la situation actuelle (abreuvement de bétail), d'autre part l'utilisation hypothétique maximale (eau de boisson, poissons, produits agricoles, lait).

g) conséquences radiologiques

Les conséquences radiologiques du transfert pour l'individu le plus exposé ont été calculées en utilisant les facteurs de dose de la CIPR 30. Avec les hypothèses adoptées, l'exposition est maximale au début de la phase de banalisation et est directement liée aux activités totales stockées.

Une étude de sensibilité a montré que les paramètres qui influent plus particulièrement sur l'exposition sont, outre les activités totales stockées, la durée de vie des blocs de béton, les taux de lixiviation, la fraction de l'eau circulant à vitesse rapide et l'utilisation de l'eau.

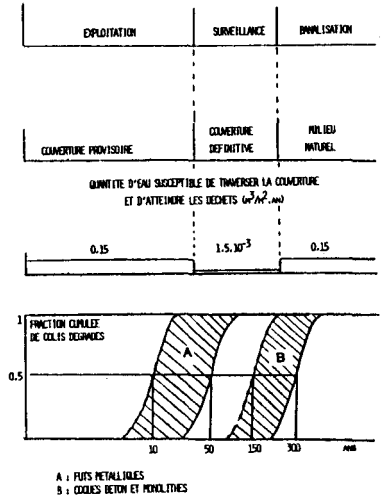
4.2. - Transfert des radionucléides par l'air

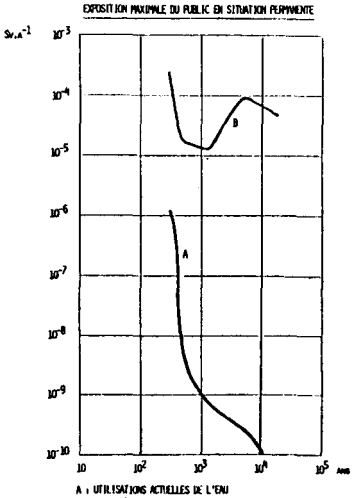
Deux scénarios-types ont été étudiés après la phase de surveillance du site : la construction d'une route et le séjour d'une personne dans une habitation construite sur le lieu du stockage. Le premier scénario est un évènement accidentel de faible probabilité intéressant un petit nombre de travailleurs ; le second est un évènement répétitif concernant quelques personnes du public. Les expositions correspondantes, liées à l'inhalation de poussières radioactives, seraient maximales au début de la phase de banalisation. Elles dépendent essentiellement de l'activité spécifique des déchets en émetteurs à vie longue. L'inventaire hypothétique maximal considéré correspondant à une teneur moyenne d'environ 0,05 Ci/t d'émetteurs alpha à vie longue pour l'ensemble du stockage, les doses engagées sur 50 ans seraient de l'ordre de 4 mSv pour le scénario routier et de 1 m Sv pour le séjour dans une habitation.

5 - Conclusion

Un modèle tenant compte de l'évolution au cours du temps et dans toutes les circonstances raisonnablement prévisibles des performances du système d'isolement multibarrière des matières radioactives constitué par les colis de déchets, les ouvrages de stockage et le milieu naturel a été mis au point pour évaluer l'impact et la capacité radiologique du Centre de la Manche.

La méthodologie utilisée est applicable à tout centre de stockage de déchets radioactifs à faible profondeur.





EXPOSURE MODELS FOR THE DISPOSAL OF WASTE RESULTING FROM HANDLING OF RADIOACTIVE SUBSTANCES AT DUMP SITES *

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Introduction

In the classification of radioactive waste, the question arises as to whether a limiting value can be proposed, such that material with less activity than this limit does not necessitate radiation protection measures. This would imply that such material may be disposed at public dumps sites together with conventional waste without any further requirements.

For determining appropriate limits it is necessary to estimate the radiation exposure resulting from the disposal of substances contaminated by low level radioactivity. Therefore the possible exposure pathways from dump sites must be analysed and their respective contribution to the exposure of man must be determined.

This study is concerned with the above topics and limiting values for 7 different radionuclides are estimated.

Model of a Dump Site

A typical dump site is proposed schematically by fig. 1.

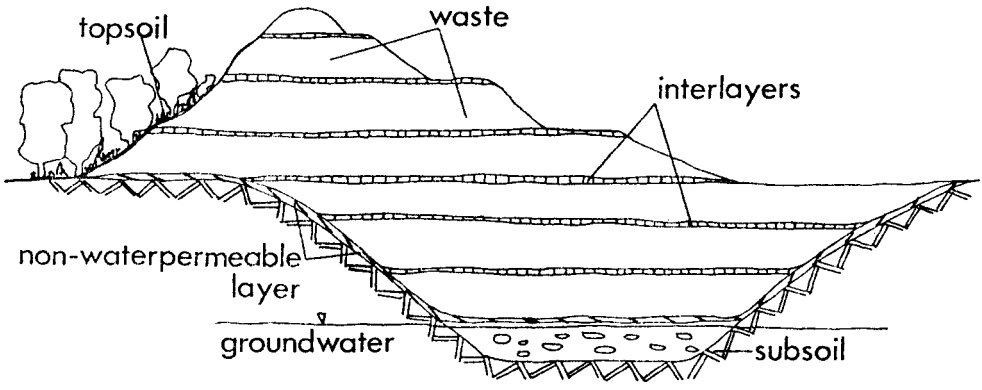

Fig. 1: A schematic representation of a reference dump site, according to (1).

* Supported by Bundesminister des Innern

The area, capacity and height of a dump site may be different (2). The first analysis has shown that for large dump sites with a considerable volume of waste disposal per year, more restrictive limits are calculated than for smaller sites. In the following calculations the reference dump site used was consequently taken as being 33 ha, and is filled to capacity within 20 years. 500 000 t waste are disposed per year, which corresponds to an annual disposal height of 2.5 m. Public dump sites must be insulated against ground water by non water permeable layers.

Exposure Pathways

While the facility is in operation, dump site workers are exposed to direct radiation and inhalation of suspended particles. Another exposure pathway is over the water drained from the dump and piped to a sewage. The assumption made for other models can also be made here in so far that the resultant purified and diluted water is consumed as drinking water by man and animals, used for irrigating agricultural land and in fish hatcheries. A fourth exposure pathway results from the German guideline (3), which specifies that abandoned dump sites must be recultivated and may be used for agricultural purposes. It must be assumed that the roots penetrate through a 30 cm layer of natural top soil and reach the waste where radionuclides can be uptaken.

The above exposure pathways, inhalation, direct radiation drainage-water and use of dump sites as agricultural land have been mathematically formulated. In the first calculations the critical exposure pathways were determined and possible limiting values calculated. Model parameters were chosen corresponding to the German regulatory guide (4) and recent references. Steady state conditions were assumed for all models.

The following conditions are assumed for the individual exposure pathways:

- The radionuclides are in a water soluable from and homogenously distributed in the waste material.
- For inhalation it is assumed that the concentration of a radionuclide in dust particles in the air is equal to that in waste material. One cubic meter of air contains 0.15 mg dust particles of a diameter less than 10 μm.
- For direct radiation, the shielding provided by waste according to Blatz (5) was taken into consideration.
- For the drainage water model an annual precipitation of 600 mm is assumed, 40 % of which is removed as drainage water. The migration rates of the radionuclides are determined in relation to their chemical characteristics. In the sewage the drainage water is diluted at a proportion of 1:100.
- When calculating activity limiting values for the exposure pathway resulting from the agricultural use of a dump site after its closure, the transfer factors soil/plant are reduced by a factor 2 or 10, according to root depth.

Results and Discussion

Using these models activity limiting values were calculated for 7 radionuclides. For this, an annual exposure of 1 mrem ≙ 0.01 mSv was assumed for a reference man. Table 1 shows the limits calculated for each exposure pathway together with the limits for all four pathways; Table 2 reflects the relative significance of each individual pathway.

	All Pathways	Inhalation	Direct Radiation	Drainage Water	Agricultural Use of the Dump
P 32	9.2E-5	3.6E-1	-	9.2E-5	-
Co 60	5.0E-8	3.6E-1	5.0E-8	9.1E-4	1.8E-4
Sr 90	2.1E-8	2.5E-3	-	3.2E-8	6.1E-8
Tc 99	7.0E-7	3.6E-1	5.9E-1	7.8E-7	7.0E-6
J 131	1.0E-6	1.1E-0	1.2E-6	6.2E-6	-
Cs 137	6.2E-7	7.1E-2	6.9E-7	7.4E-5	6.6E-6
Ce 144	1.5E.5	3.6E-2	1.5E-5	4.4E-1	5.3E-1

Table 1: Limiting values for waste in Ci/t according to models for inhalation soil radiation, seep-water and agricultural use together with the limit for all pathways, assuming an annual exposure of 1 mrem per year to man.

	Inhalation (%)	Direct Radiation (%)	Drainage Water (%)	Agricultural Use of the Dump (%)
P 32	<1	0	>99	0
Co 60	0	>99	< 1	< 1
Sr 90	0	0	65	35
Tc 99	0	0	90	10
J 131	0	84	16	0
Cs 137	0	90	1	9
Ce 144	<1	>99	0	0

Table 2: Proportions of the individual exposure pathways to total exposure.

These calculations show that the pathways direct radiation and drainage water are of particular significance for the considered nuclides. In assessing these results, it must be realised that only direct radiation was calculated under sufficiently realistic conditions. For drainage water, inhalation and for agricultural use of the closed dump site, some conservative assumptions were made to correspond with the German Regulatory Guide (4). It is remarkable that even making these assumptions the most restrictive activity limits are calculated for Sr-90 and Co-60. The limiting values for the remaining five radionuclides are greater by at least one order of magnitude. It is of particular interest

to see how these activity limits change when it is assumed that radionuclides are not homogeneously distributed in all the waste material, but are disposed in discrete amounts.

Literature

(1) M. Müller, "Städtebau", B.G. Teubner Verlag, Stuttgart, 1970

(2) W. Schenkel., "Die geordnete Deponie von festen Abfallstoffen" Fachzeitschrift für die Behandlung und Beseitigung von Abfällen, Heft 9, 1974

(3) Arbeitsgemeinschaft für industrielle und gewerbliche Abfallfragen, "Die geordnete und kontrollierte Ablagerung von industriellen und gewerblichen Abfällen" Merkblatt G 7

(4) Bundesminister des Innern, "Allgemeine Berechnungsgrundlagen für die Strahlenexposition bei radioaktiven Ableitungen mit der Abluft oder in Oberflächengewässer", Gemeinsames Ministerialblatt 30 (1979) 369-436

(5) H. Blatz, "Radiation Hygiene Handbook", Mc-Graw-Hill Book Company, New York, 1959

ASSESSMENT OF RADIATION DOSES DUE TO RELEASES TO THE BIOSPHERE FROM NUCLEAR WASTE REPOSITORIES

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INTRODUCTION

Estimation of radiation doses due to releases from nuclear waste repositories comprises phases covering first the near-field, i.e. the behaviour of radioactive waste products, canisters, buffer materials and rock and ground-water close to the repository, and second, radionuclide migration in the geosphere and, third, migration in the biosphere and accumulation of radiation doses. A full analysis chain is performed in Reference /1/, and in this paper some details of the last phase are discussed for the basic scenario of the Reference /1/. The biospheric model is described shortly and the calculational behaviour of two important radionuclides, Ra-226 and U-238 is shown. The total results for the basic spent fuel scenario are presented as an example of the type of information obtained from analyses.

The waste type analysed comprises the spent fuel of two BWR plant operations (1200 tU). The waste is emplaced in copper canisters and the repository is assumed to be located on the coast of the Baltic Sea at the depth of 500 m in hard crystalline bedrock /1/.

BIOSPHERIC MODEL

The release from the repository is assumed to flow via ground water to well, lake or coastal sea. The transfer of radionuclides in the biosphere is described employing a dynamic compartment model, and in each compartment the radioactive material is assumed to be distributed in two phases, solid phase or liquid phase, which is described with the distribution coefficient K_d. The transfer of radionuclides through the part of the biosphere where time-constants are short, as in plants and animals, is described with the employment of concentration factors. Radioactive decay and build-up of daughter-nuclides are taken into account in the model. The exposure pathways which are included in the model are radiation from ground or shoreline, swimming, boating, and ingestion of water, fish, milk, meat and grain. /2/ Figure 1 shows a typical compartment model employed in calculations for the local scale and Figure 2 for the regional (Baltic) and global scales.

The parameters which are needed to calculate the transfer of radionuclides in the biosphere can be divided into two groups. The first group includes parameters which are used to describe the compartments (masses and volumes) and connections between them (flows of water and solids from one compartment to another or outside of the compartment system). The second group of parameters are the nuclide- or element-dependent parameters, which cause that different radionuclides behave in a very different way in the biosphere. These parameters are half-life $T_{1/2}$ and distribution coefficient K_d.

RESULTS AND DISCUSSION

The radioactive half-lifes $T_{1/2}$ of the release nuclides considered vary between 10^3 - 10^{11} a, and the distribution coeffient K_d between 1 - $5 \cdot 10^6$ dm^3/kg. In the following, the behaviour of two in this respect very different radionuclides is studied. These nuclides, U-238 and Ra-226, belong according to calculations to the most important radionuclides in the discussed release scenario.

Table I shows the Ra-226 activity decaying in the local, regional and global scale compartments due to the Ra-226 activity release Q on the local scale. Table II presents the corresponding values for U-238 decay due to U-238 release. The greater K_d-value of the nuclide Ra-226 causes that a greater part of the Ra-226 activity than of the U-238 activity decays on the local scale. Also on the regional scale the accumulation of radionuclide Ra-226 to sediments is stronger in all compartments than the accumulation of U-238. In addition to the situation shown in Table I, the Ra-226 activity is, naturally, also formed by the decay of U-238, U-234 and Th-230 releases in the biosphere.

In Table III the main results of spent fuel dose calculations are shown. The estimated maximum dose rate for a critical person in the basic scenario is 0.02 mSv/a on the local scale and $3 \cdot 10^{-9}$ mSv/a on the global scale. The collective dose commitment in local scenario is $1 \cdot 10^3$ manSv and in global scenario $1 \cdot 10^5$ manSv. The accumulation rate of the collective dose on the global scale is, however, extremely small. The integrated collective dose over those 500 a when the dose rate is highest is $2 \cdot 10^{-2}$ manSv on the local scale and 1 manSv on the global scale. The most important pathways are drinking water from a local well and ingestion of fish. In the main case, where releases to the biosphere begin 10^6 years after disposal, the most important radionuclides belong to the decay chains 4N+2 (U-238 chain) and 4N+1 (Np-237 chain).

On the local scale the most important nuclides are Np-237 and Ra-226. In the case of Ra-226 the doses are in calculations mainly caused by the daughter-nuclide Pb-210. The sedimentation removes more Np-237 and Ra-226 activity from the lake compartment than the water flow from the lake. This is valid also in the case of the most important nuclides Th-229 and Ra-226 on the regional scale, because the K_d-values are relatively high. On the global scale the nuclide U-238 accumulates slowly in the Ocean (residence time 10^6 a). Because its half-life is very long, the sedimentation is also in this case a more important process in removing activity from the Ocean than the radioactive decay.

The accumulation of radioactive material in the sediments seems to be an important removal factor on the scales considered. Therefore it is essential to be able to model the transfer to sediments and the feedback from sediments correctly. The calculated behaviour of radionuclides can be compared with the behaviour of natural radionuclides. The ratios of the concentrations of daughter-nuclides to the concentration of U-238 mother-nuclide are in relatively good agreement in the calculated and measured situations /2/.

REFERENCES

/1/ KORHONEN, R., PELTONEN, E.K., SAVOLAINEN, I., HEINONEN, O.J., MUURINEN, A., VALKIAINEN, M., A comparative assessment of radiological impact due to nuclear waste disposal in hard crystalline bedrock. IAEA conference on radioactive waste management. Seattle, USA, May 16-20, 1983.

/2/ KORHONEN, R., SAVOLAINEN, I., Assessment of biospheric behaviour of releases from nuclear waste repositories. IAEA and CEC seminar on the environmental transfer to man of radionuclides released from nuclear installations, Brussels, Belgium, 17-21 Oct. 1983.

Table I Fractions of Ra-226 release Q decaying in the local, regional (Baltic) and global (Ocean) scale compartments ($T_{1/2}$ = 1600 a, $K_d = 3 \cdot 10^4$ dm^3/kg)

Scale	Lake/sea	Surface sediment 0-5 cm	Buried sediment 5-35 cm	Sediment sink	$\sum$
Local lake	$3.9 \cdot 10^{-5}$ Q	$5.0 \cdot 10^{-3}$ Q	$7.3 \cdot 10^{-2}$ Q	$7.2 \cdot 10^{-1}$ Q	$8.0 \cdot 10^{-1}$ Q
Baltic	$7.4 \cdot 10^{-4}$ Q	$4.6 \cdot 10^{-3}$ Q	$4.6 \cdot 10^{-2}$ Q	$6.8 \cdot 10^{-2}$ Q	$1.2 \cdot 10^{-1}$ Q
Ocean	$6.4 \cdot 10^{-2}$ Q	$5.8 \cdot 10^{-3}$ Q	$7.8 \cdot 10^{-3}$ Q	$7.6 \cdot 10^{-4}$ Q	$8.0 \cdot 10^{-2}$ Q
$\sum$	$6.5 \cdot 10^{-2}$ Q	$1.5 \cdot 10^{-2}$Q	$1.3 \cdot 10^{-1}$Q	$7.9 \cdot 10^{-1}$ Q	1.0 Q

Table II The fractions of a U-238 release Q decaying in various compartments ($T_{1/2} = 4{,}5 \cdot 10^9$ a, K_d = 500 dm^3/kg)

Scale	Lake/sea	Surface sediment 0-5 cm	Buried sediment 5-35 cm	Sediment sink	$\sum$
Local lake	$7.0 \cdot 10^{-11}$ Q	$2.0 \cdot 10^{-10}$ Q	$3.3 \cdot 10^{-9}$ Q	$8.9 \cdot 10^{-2}$ Q	$8.9 \cdot 10^{-2}$ Q
Baltic	$2.8 \cdot 10^{-9}$ Q	$3.6 \cdot 10^{-10}$ Q	$6.1 \cdot 10^{-9}$ Q	$2.5 \cdot 10^{-2}$ Q	$2.5 \cdot 10^{-2}$ Q
Ocean	$1.0 \cdot 10^{-4}$ Q	$2.0 \cdot 10^{-7}$ Q	$3.4 \cdot 10^{-6}$ Q	$8.8 \cdot 10^{-1}$ Q	$8.8 \cdot 10^{-1}$ Q
$\sum$	$1.0 \cdot 10^{-4}$ Q	$2.0 \cdot 10^{-7}$ Q	$3.4 \cdot 10^{-6}$ Q	1.0 Q	1.0 Q

Table III Calculated individual dose rates and collective doses due to releases from a spent fuel repository

Scale	Maximum individual dose rate (Sv/a) and the most important nuclides	Collective dose during the most exposed 500 a (manSv)	Collective dose 0-10^8 years (manSv)
Local	$2.0 \cdot 10^{-5}$ Np-237 , Ra-226	$1.8 \cdot 10^{-2}$ Np-237 , Ra-226	$1.2 \cdot 10^{3}$ Ra-226 , Np-237
Regional	$2.2 \cdot 10^{-12}$ Th-229 , Ra-226	$2.2 \cdot 10^{-2}$ Th-229 , Ra-226	$2.0 \cdot 10^{3}$ Ra-226 , U-238
Global	$2.6 \cdot 10^{-12}$ U-238 , Th-229	$1.3 \cdot 10^{0}$ U-238 , Th-229	$1.2 \cdot 10^{5}$ U-238

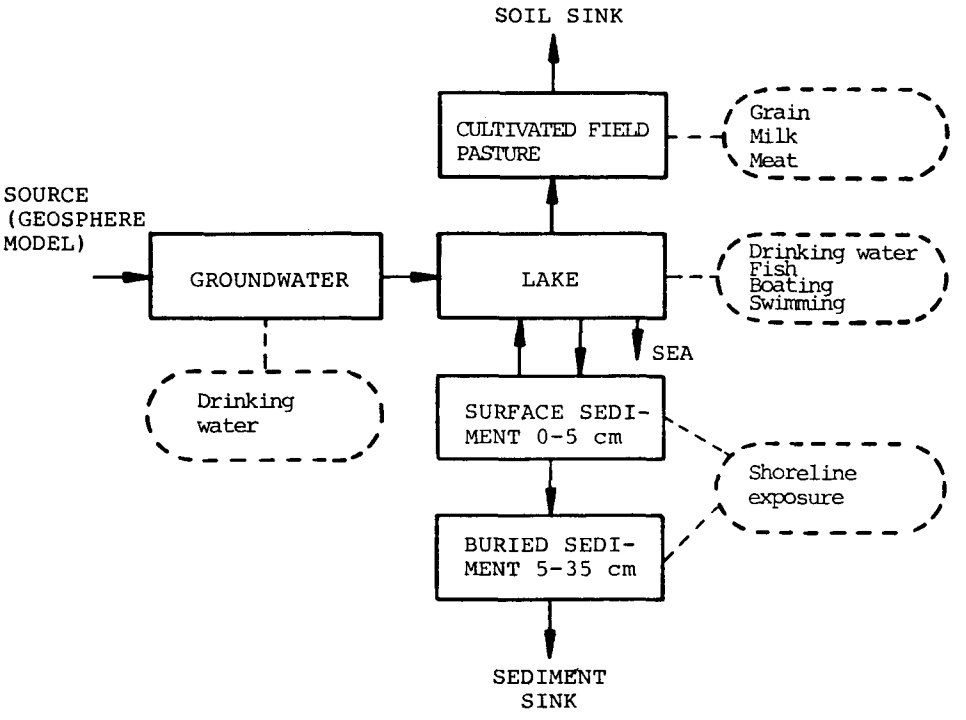

Fig. 1 Compartment model for local scale calculations.

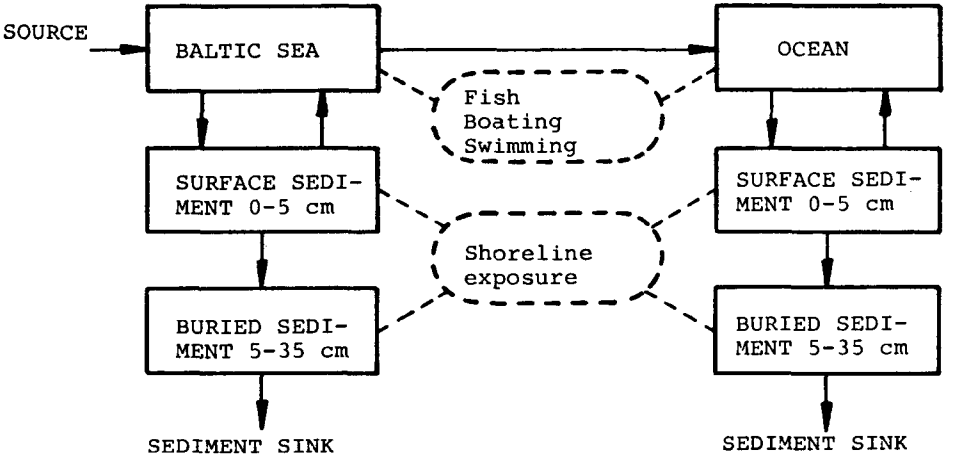

Fig. 2 Compartment model for regional and global scales.

RADIATION PROTECTION CONCEPT FOR THE PLANNED RADIOACTIVE WASTE REPOSITORY IN THE KONRAD MINE

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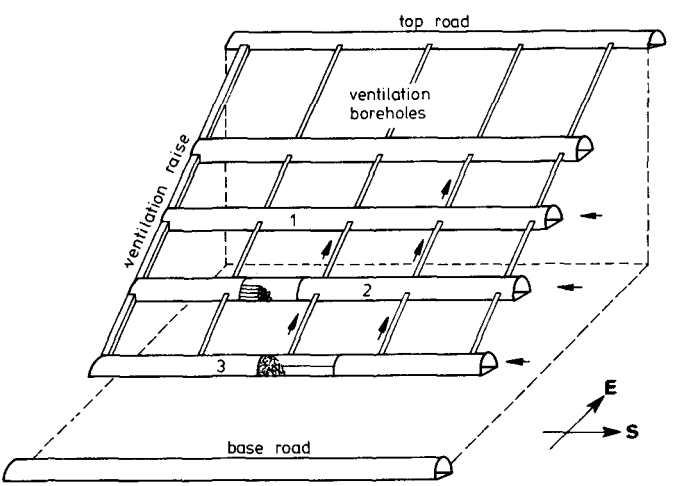
INTRODUCTION, BASES

It is intended to establish the planned repository in the former Konrad iron ore mine near Salzgitter-Bleckenstedt. With a dip of approx. 23°, the iron ore bed proposed for the deposition of radioactive waste is situated at a depth of approx. 1300 m to 800 m. The deposition cavities to be constructed during the operating phase form horizontal galleries with a cross section approx. 7 m in width and 6 m in heigth (Fig. 1). The different waste packages will be transported below ground with trackless diesel vehicles and stacked over the whole cross section of the gallery. The hollow spaces between the waste packages will then be backfilled.



1 - gallery used as air course prior to use for disposal
2 - upper storage gallery
3 - lower storage gallery

Fig. 1: Principle of a storage-field (Sectional area of the storage-rooms: 40 m²)

The planning bases relevant to radiation protection are in agreement with /1/:
- separation in space of the waste package and debris transports,
- parallel ventilation of the deposition and driving areas and, moreover, the supply of the waste transport roadways with fresh air,
- limitation to 5 mSv/a on an average of the exposure of the staff to direct and stray radiation which is largely unavoidable when handling and depositing the

waste; limitation of the largely avoidable exposure of the staff to radiation to the amount of a whole-body resp. effective dose of 0.5 mSv/a as a result of inhalation, submersion etc. of released radioactive substances or of detached, non-adhesive surface contamination (reference dose rate values taken as a basis of the planning work),
- limitation of the potential exposure to radiation in the surroundings as a result of a discharge of released radioactive substances, to values of approx. 20 % of the limiting values defined in § 45 of the Radiation Protection Ordinance /2/.

The causes of the staff's exposure to radiation are to be found on the one hand in the uranium/radium and thorium content of the iron ore and on the other, in the direct radiation of the waste as well as in the release of radioactive substances from the waste packages or their surfaces. The basic exposure (referred to 2000 working hours per year) is composed of the direct radiation in the individual galleries (additional dose up to 0.18 mSv/a as compared with the dose above ground), the inhalation of the iron ore dust (whole body dose 0.04 mSv/a) as well as the inhalation of radon-220/222, including their short-lived decay products (effective equivalent dose up to 12 mSv/a) /3, 4, 5/.

The radon exposure can be strongly influenced by the way of ventilation, in particular when allowing for goafs. It will be ensured that persons working below ground who do not belong to the staff exposed to radiation will get a basic exposure not exceeding a whole-body resp. effective dose of 5 mSv/a. For staff exposed to radiation, the basic exposure together with the radiation emanating from the waste form the professional exposure to radiation for which the limiting values defined in the Radiation Protection Ordinance are to be complied with.

To reach the objectives "compliance with the reference values of the exposure of staff and environment to radiation due to waste taken as a basis of the planning work" and to control the unavoidable effects, the radiation protection concept described in the following is applied /6/. The basis of the preparation of such a concept is formed by a detailed and complete description of the operational and technical sequences within a repository.

PRINCIPLES OF THE RADIATION PROTECTION CONCEPT

The paramount functional criteria of mine operation are the preponderantly "mobile activities"; here the transports between varying starting and final points are to be referred to in the main. The combination of conventional mine operation and handling of radioactive waste packages, the environmental conditions below ground which are unfavourable from the point of view of radiation protection (dust, distances to be covered, unfavourable surfaces) and influence to a high degree the character of the work and the technologies applied, were taken into account when defining the principles of the radiation protection concept set forth in the following.

The whole mine will be subdivided into a conventional and a nonconventional area. This subdivision refers not only to the whole mine structure but also to the technical operations (preparation, transport, deposition, supply, repair), the organizational services (executive services such as mining, electrotechnical, ventilation and radiation protection service) with their functional sequences and the equipment used with their respective locations. For practical reasons, it is recommended to declare the whole non-conventional area including the upcast ventilation shaft and the above ground installations for the handling of waste packages to be a controlled area in the sense of radiation protection. All overlapping activities are to be restricted to a minimum and monitored.

Another principle of the radiation protection concept results from the fact that - in contrast to other nuclear installations in which the handling of radioactive substances determines the planning and sequence of operations to a large extent - the conditions below ground and the mining requirements determine the character of plant and operation of a mine used as a repository. This means that without abandoning radiation protection principles, the radiation protection measures are to be adapted to the largest possible extent to the operating conditions. This objective can be reached best by decentralized radiation protection.

A continuous feedback between radiation protection and operation will be of fundamental importance. "Special" and "long-term supervision programmes" will therefore be practised in a well-defined way in order to profit from the experience gained by reducing too rigorous radiation protection measures delaying mine operation and by analyzing all points and sequences of operation relevant to radiation protection so that long-term non-conventional operating disorders (such as, for example, the slow contamination of a major gallery section) can be prevented.

The last principle to be set forth is that of how to act in the case of an incident. Incidents in a mine used as a repository are subdivided into incidents with mechanical, with thermal and with mechanical and subsequent thermal impact. The incidents themselves and their effects are characterized by the considerable size of the mine, the resulting diversity of the places where an incident can take place, the passive nature of the barriers and the large number of staff members potentially involved, due to the "open ventilation" (no activity inclusion using progressively lower pressure). The largest complex of problems resulting from a release of activity due to an incident comprises the intervention in the actual ventilation with a view to limiting the propagation of the activity. This can be achieved technically by sealing off the zone where the incident has taken place from the rest of the mine structure by establishing a sufficient number of ventilation doors ("ventilation barriers").

TASKS AND PERFORMANCE OF RADIATION PROTECTION

In the following the main criteria of the most important measurement tasks and control measures will be set forth.

The general examination of persons for contamination will be carried out in a decentralized way, the persons working in the mine being examined below ground. All accesses to the controlled area are provided with monitored contamination detectors. Should a contamination be ascertained, the contaminated persons will be decontaminated - in dependence upon the extent of the contamination - either locally by means of a decontamination vehicle or - after appropriate isolating measures have been taken - above ground in the decontamination room for persons. Equipment and working areas are monitored for contamination as well, particular importance being attached to operating points with increased dust exposure and sealed storage chambers.

The monitoring of the repository for radiation is made by measuring the local dose in the whole installation with thermoluminescence dosimeters and the local dose rate with stationary (above ground), temporarily stationary (chambers, transport roadways) and mobile (radiation protection missions) devices. The storage chambers are an essential source of radiation to which the staff is exposed. At a distance of 10 m, a stack of packages filling the whole cross section causes a dose rate of up to 0.1 mSv/h, the component of the radiation scattered at the rock being of the order of almost 50 % /7/. Without additional shielding, a person staying here only for approx. 50 hours would already receive the mean annual dose of 5 mSv/a which serves as a reference value for the planning. The storing machines and vehicles are therefore equipped with shielding systems. For the control of the persons exposed to radiation, official and self-reading personal dosimeters are used. Incorporation

controls depend upon the actually measured concentrations of the activity in the ventilation and ambient air and will be carried out only in special cases.

The kind and extent of the air activity and emission control will be determined substantially by the expected - though small-scale - release of volatile nuclides from the waste packages. The radionuclides H-3, C-14, traces of J-129 as well as Rn-220/222 (from iron ore and waste) require special control methods. In addition, aerosol traces from released surface activity are expected. The activity concentrations at the workplaces will be substantially determined in a discontinuous way, whereas the discharge via the exhaust air and the sewage/residual water path (decontamination waters, mine waters) will be determined continuously.

The environmental control programme is characterized by an emission spectrum differing from that of nuclear power plants and by a low emission level. The measurement programme which is to be established in agreement with the plan approval authority will therefore refer mainly to the operating site as well as to the immediate environment.

REALIZATION OF THE CONCEPT

Within the scope of the preparation of documents for the plan approval procedure, the mine, the operating conditions and the waste packages are submitted to a safety analysis /8/. The following criteria are allowed for in the analysis as parameters which are characteristic of the Konrad mine and relevant to radiation protection: relatively high radiological basic exposure (radon), accumulation of mine waters (approx. 15000 m^3/a) which are pumped above ground and discharged in a controlled way (water path), "open ventilation", size of the mine (some kilometers), as well as dust. Up to the plan approval decision which can be taken in the year 1987 at the earliest, the final quantitative requirements on the mine, operating conditions and waste packages will be defined iteratively /8/. The realization of the final radiation protection concept can thus be commenced in 1987 at the earliest. In the course of the "change-over" of the Konrad mine into a repository which is proposed to be carried out in 1987 to 1989, the radiation protection requirements on the technical installations and equipment must be complied with and the organizational and administrative preconditions for the carrying out of the radiation protection tasks provided.

/1/ Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk. Bundesanzeiger, Jahrgang 35, Nummer 2, 05. Januar 1983.
/2/ Verordnung über den Schutz vor Schäden durch ionisierende Strahlen. (Strahlenschutzverordnung - StrlSchV) vom 13.10.1976, BGBl. I, S. 2905.
/3/ Gesellschaft für Strahlen- und Umweltforschung, München: Abschlußbericht Eignungsprüfung der Schachtanlage Konrad für die Endlagerung radioaktiver Abfälle. GSF - T 136, Juni 1982.
/4/ Physikalisch-Technische Bundesanstalt, Braunschweig: Radon-Messungen in der Grube Konrad. Aktenvermerk Eh/Ko vom 02.03.1983.
/5/ Gesellschaft für Strahlen- und Umweltforschung, München: Durchführung von Beweissicherungsmaßnahmen: Radon- und Thoron-Aktivitätskonzentration im Grubengebäude Schachtanlage Konrad.
/6/ Gesellschaft für Umweltüberwachung mbH, Aldenhoven, 1981: Strahlenschutzkonzept für die zu errichtende Anlage des Bundes zur Endlagerung radioaktiver Abfälle. Abschlußbericht, GUW - 0780-01.
/7/ D. Ehrlich et al.: Dose Planning and Calculations for Radioactive Waste Repository Plants in Mines. Proc. of the Sixth Internat. Conf. on Radiation Shielding, Tokyo, May 16-20, 1983.
/8/ E. Warnecke et al., in R. Odoj, E. Merz (Hrsg.): Proceedings of the International Seminar on Chemistry and Process Engineering for High-Level Liquid Waste Solidification. Jul-Conf-42 (Vol.2) 792-815 (1981).

STATISTICAL ANALYSIS OF THE ADEQUACY OF EXISTING RISK ASSESSMENT RESULTS FOR HIGH LEVEL NUCLEAR WASTE REPOSITORIES

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Introduction

The importance of radiological performance assessment to the successful licensing of any nuclear facility can hardly be overstated. This is particularly true for a high-level nuclear waste repository because of its being the first deep geologic nuclear facility and the first with such a long time span of concern. The ONWI performance assessment approach relative to postclosure safety focuses on continuous processes and superimposes the discrete events on process behavior. This approach is used because nuclear waste isolation systems in salt are composed of components which degrade slowly rather than fail suddenly, and isolation system performance at suitable sites is likely to be dominated by continuous processes rather than discrete events.

The primary purpose of this study was to determine whether the present literature contains risk analyses of discrete events of scope and depth sufficient for licensing preparation. The sufficiency criterion used was the presence of enough completely developed events to represent, in sum, essentially the total risk posed by the repository as evaluated using a full-range consequence/probability (risk) matrix technique. Data were extracted from literature citing assessments based both on scenario development and fault-tree techniques.[1,2]

Following a brief description of the postclosure risk matrix, the available assessment literature will be discussed and its sufficiency determined. Similar data for the preclosure phase will then be developed and combined with that from the postclosure to make possible a comparison of relative risk of postclosure vs. that from preclosure. Finally, results from these analyses are used as a basis of recommending priorities for future scenario development work.

Associated with the primary purpose of this study were several technical objectives. These included the provision of guidance for future scenario development work; design basis accident scenarios, in particular; and secondly, estimation of the total risk associated with the operation and existence of a high level nuclear waste repository. Historically, both the total risk and design basis accident concepts have been extremely important in nuclear power plant licensing activities in the U.S. Present indications are that similar importance will be given these concepts at least in the public sector of nuclear waste repository licensing.

Risk

Risk is classically defined as a consequence times a probability. A quick scan of postclosure performance assessment literature indicates that analyses have been made for events of probabilities 10^{-8} to 10^{-1} and consequences ranging over seven orders of magnitude.

Thus, to a first approximation, the postclosure risk posed by a repository can be represented by a 7x8 matrix. Quantitatively, the total risk is the sum of the probability/consequence products of all elements of the matrix. Undoubtedly, multiple events will fall into many of the elements, but this is expected to happen most often in the lower consequence events where hundreds of events would be needed to affect the total risk estimate.

Figure 1 indicates the percent contribution each matrix element makes to the total risk estimate. Elements, for which events have been analyzed in the literature, are also indicated. The initial intent was to analyze the entire probability/consequence matrix and quantify the uncertainty in risk as well as developing a risk estimate. Classical and/or Bayesian techniques could be used if the matrix was not so sparse. Due to the paucity of scenarios presently analyzed, such methodological approaches are not feasible. Therefore, a more basic approach must be utilized. It might be noted that the sum of the percent contributions of those elements which have been analyzed is only 0.81% which can be increased substantially if high risk elements are engineered out of the system.

FIGURE 1

Contribution of Matrix Elements to Total Risk [%]

Postclosure

Relative Consequence of Occurrence	Probability of Occurrence 10^{-8}	10^{-7}	10^{-6}	10^{-5}	10^{-4}	10^{-3}	10^{-2}	10^{-1}	10^{-0}
10^{7}	8.1E-6	8.1E-5	8.1E-4	8.1E-3	8.1E-2	8.1E-1	8.1E-0	8.1E-1
10^{6}	8.1E-7	8.1E-6	8.1E-5	8.1E-4	8.1E-3	8.1E-2	8.1E-1	8.1E-0
10^{5}	8.1E-8	8.1E-7	8.1E-6	8.1E-5	8.1E-4	8.1E-3	8.1E-2	8.1E-1
10^{4}	8.1E-9	8.1E-8	8.1E-7	8.1E-6	8.1E-5	8.1E-4	8.1E-3	8.1E-2
10^{3}	8.1E-10	8.1E-9	8.1E-8	8.1E-7	8.1E-6	8.1E-5	8.1E-4	8.1E-3
10^{2}	8.1E-11	8.1E-10	8.1E-9	8.1E-8	8.1E-7	8.1E-6	8.1E-5	8.1E-4
10^{1}	8.1E-12	8.1E-11	8.1E-10	8.1E-9	8.1E-8	8.1E-7	8.1E-6	8.1E-5
10^{0}								

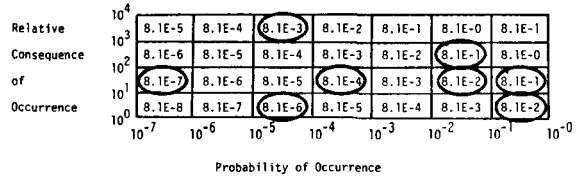
Figure 2 shows the risk matrix for the preclosure repository phase. In this case the sum of the contributions from those elements which have been analyzed is 1.8%. Here, as in the case of the postclosure matrix, the total risk is dominated by the highest risk matrix element. It is highly likely that events portrayed in this portion of the matrix can be effectively precluded through engineering design. If such means were used to eliminate any contributions from the top three risk elements, the 1.8% sum would rise significantly to 64%.

Examination of both the preclosure and postclosure risk matrices indicates that 97% of the total risk comes from events characterized in the top three risk elements. This fact would tend to indicate the need to identify events, if any, which fall into these elements and engineer them out of the system. The next tier of elements may be either treated similarly to the higher risk elements or may be developed into the "design basis accidents" which were referred to earlier as a need for successful licensing. Major efforts are needed in this area in the postclosure phase since few events in

FIGURE 2

Contribution of Matrix Elements to Total Risk [%]

Preclosure



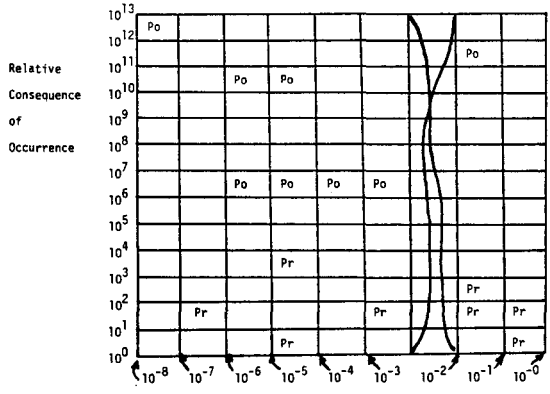
this region of the matrix have been analyzed. In both postclosure and preclosure cases, significant progress toward meeting these needs are anticipated as repository designs develop over the next few years. Since it may be difficult to determine the proper cell of the probability/consequence matrix for a given discrete event, the theory of fuzzy sets may be utilized in the development of cell membership in the database. When sufficient data have been developed by the application of fuzzy set theory and/or other decision theoretic approaches, sensitivity/uncertainty analyses can be performed to determine which areas still need further development.

The question might be asked which repository phase warrants greater emphasis if characterization of risk is the objective. Figure 3 shows the preclosure and postclosrue matrices merged with common probability and consequence scales. Similarly to the development of the matrices in Figures 1 and 2, this merger was accomplished by normalizing the combined pre and postclosure consequence set to the magnitude of its lowest member. It is immediately evident that the total risk over all time from a repository is dominated by the postclosure phase risks. Therefore, it is only logical

FIGURE 3

Available Repository Risk Data

Preclosure and Postclosure



Po = postclosure
Pr = preclosure

that this phase receive the priority attention. However, to reiterate an earlier point, in both phases the spectrum of accidents in the probability/consequence context must be expanded if the true picture of repository performance is to be projected. The specific scenarios which are needed are those with matrix coordinates such that they contribute significantly to overall risk. Preliminary indications are that this type of guidance makes the development of needed scenarios much more efficient than is ordinarily the case with a more random approach. Perhaps the efficiency of this process could be even further enhanced if any development efforts were delayed until the appropriate designs were available. However, history has shown that many times the scenario development can have a more significant effect on design than vice versa. Therefore, the strategy to be followed in the ONWI program is to go as far as possible with the information now in hand and refine the results and interpretations as more detailed data become available.

Conclusions

The low overall percentages of risk represented by the scenarios which have been analyzed could be interpreted to mean that little work has been done in this area; such is definitely not the case. On the contrary, such results are what would be expected from the random selection/reanalysis mode in which we have been operating for the last several years. Only if a more systematic approach to scenario development is employed in the next few years will a sufficient database be available on which to do the statistical analyses required for licensing.

References

(1) Koplik, C.M., M.F. Kaplan and B. Ross, 1982. "The Safety of Repositories for Highly Radioactive Wastes", Review of Modern Physics, 54, 269, January.

(2) Science Applications, Inc., 1980. Repository Accident Scenarios, ONWI/Sub/80/E513-02105 R-1, Office of Nuclear Waste Isolation, Battelle Memorial Institute, September.

INCIDENCE DE LA VARIABILITE DE PARAMETRES BIOPHYSICO-CHIMIQUES SUR LES CONSEQUENCES SANITAIRES EVENTUELLES DE LA LIBERATION DANS LE MILIEU DE RADIONUCLEIDES A PARTIR D'UN SITE DE STOCKAGE

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INTRODUCTION

Malgré une très faible probabilité, des radionucléides peuvent s'échapper des barrières ouvrées (ou artificielles) vers les milieux constituant l'environnement de l'homme. Il importe qu'à tout instant, après le dépôt des déchets, les doses d'irradiation soient les plus faibles possibles, conformément à la CIPR. Les radionucléides tests envisagés sont ^{241}Am et ^{237}Np, transuraniens particulièrement importants dans le cadre des études sanitaires sur les déchets.

VARIABILITE DES PARAMETRES BIOPHYSICO-CHIMIQUES

La variabilité de ces paramètres ne peut être envisagée dans son ensemble ; on se limitera donc à quelques exemples, dans le cadre de stockage de déchets.

Terme source

Lors de leur apparition dans le milieu, Np et Am sont très probablement à l'état de NpO_2^+ et Am^{3+} qui ont respectivement des comportements voisins d'ions alcalins et alcalinoterreux (ou Fe^{3+}) (NpO_2^+ très mobile, Am^{3+} moyennement mobile à peu mobile) [1]. La mobilité est définie par rapport à l'eau ou l'un de ses bons marqueurs.

Aspects hydrologiques

Les caractéristiques du transfert de radionucléides, en particulier les caractéristiques cinétiques dépendent de paramètres microbiologiques (surtout en zones non saturées et à saturation temporaire) et de paramètres structuraux, physico-chimiques et pédochimiques des milieux traversés.

Aspects microbiologiques

Les micro-organismes interviennent notamment par la production de composés complexants. Leur incidence sera donc beaucoup plus grande sur la mobilité chimique de l'ion trivalent Am^{3+} (facteur de variabilité.10^1) qur sur celle de l'ion monovalent NpO_2^+ (facteur de variabilité << 10^1) [2].

Aspects structuraux

En zone non saturée, interviennent tous les paramètres agronomiques de la structure des sols [3]. En zone saturée, l'accent doit être mis sur l'incidence de la fissuration. L'incidence sur les valeurs du "retard mécaniue" dans le mouvement des radionucléides peut être très grande, mais cette notion ne peut être séparée des "effets de parcours" qui, outre les aspects cinétiques, intègrent des modifications dans les états physico-chimiques des radionucléides [4] (surtout Am^{3+}).

Aspects physico-chimiques

Outre les insolubilisations et les solubilisations, interviennent les complexations et les conséquences de variations souvent simultanées du pH et du redox.

L'incidence des complexations paraît souvent surestimée, la formation de complexes avec les traces pondérales des radionucléides étant très fortement concurrencée in situ par l'existence de quantités pondérales relativement énormes des macro-éléments (Na, K, Ca, Fe, etc...). L'incidence des complexations est cependant plus forte pour Am que pour Np [1]. Les phénomènes de sorption sur les phases solides sont fonction de divers paramètres minéralogiques et physico-chimiques. Par exemple, pour l'argile kaolinite, la sorption à pH 7 est environ 20 fois moins grande que pour la montmorillonite. Toutes choses étale, NpO_2^+ est beaucoup moins intensément sorbé que Am^{3+} (facteur de l'ordre de 10^2) [1].

COEFFICIENTS DE DISTRIBUTION

Il est commode, dans les modélisations, de quantifier les échanges de matières entre phases solide et liquide par 2 paramètres :

$$k_d = \frac{\text{activité massique solide}}{\text{activité volumique liquide}} \text{ (dimension : } L^3M^{-1})$$

K_d Surfacique $K_{ds} = \frac{\text{activité spécifique surface solide}}{\text{activité volumique liquide}}$ (dimension : L)

En ce qui concerne les aspects cinétiques, on recourt au facteur de retard

$R = \frac{\text{vitesse du mouvement de l'eau}}{\text{vitesse de transfert du radionucléide}}$ K_d, K_s et R sont, en fait des paramètres globaux intégrant un certain nombre de paramètres primaires [5,6] dont ceux qui viennent d'être évoqués. Leur variabilité dépend donc, d'une manière complexe, de la variabilité de ces paramètres primaires.

PRISE EN COMPTE DE L'INCERTITUDE NUMERIQUE ET DE LA VARIABILITE SPATIALE DES PARAMETRES BIO-PHYSICOCHIMIQUES

Lorsqu'on étudie les conséquences sanitaires dues au relâchement dans le sous-sol de radioéléments provenant d'un stockage de déchets radioactifs, les équations plus ou moins sophistiquées qui sont utilisées, pour simuler le transfert de ces radioéléments, nécessitent la connaissance numérique précise des paramètres. Si tout ou partie des paramètres est muni d'une incertitude, ceux-ci seront choisis plus ou moins arbitrairement et les résultats avancés seront munis d'une incertitude difficilement estimable.

Dans notre démarche, nous remplaçons la notion de paramètre déterministe par la notion de paramètre heuristique : cela veut dire que l'on remplace la valeur d'un paramètre incertain ou de variabilité incertaine par une distribution de valeurs déterminée à partir d'un échantillonnage de mesures in-situ, ou d'expérimentations. Si, dans l'environnement d'un site de stockage, on effectue N mesures P_i d'un paramètre, il est possible en faisant une analyse fréquentielle par histogramme de déduire une distribution de probabilité pour ce paramètre.

Nous simulons, ensuite, le transfert des radioéléments à travers l'environnement par la méthoe dite de Monte-Carlo. Lorsqu'une particule d'Am 241 ou de Np 237 se trouve dans un domaine D, au lieu de prendre une valeur déterminée pour chaque paramètre, on choisit une valeur compte tenu de la distribution de probabilité de ce paramètre [7].

Cette méthode nous a permis de faire un certain progrès dans l'étude des risques radiologiques d'un stockage. Il est en effet tenu compte de la dispersion numérique des résultats expérimentaux, ce qui est plus satisfaisant que la prise

en compte de la valeur dite "la plus pénalisante" ou de la valeur dite "moyenne". Ensuite, cela nous a permis de correler les paramètres, c'est-à-dire d'éviter d'avoir un jeu de paramètres incompatibles : le choix de la valeur d'un paramètre doit induire la valeur des paramètres qui lui sont corrélés. A titre d'exemple, le choix de la valeur d'un coefficient de distribution surfacique K_{ds} du Np 237 ou de l'Am 241 ne peut se faire indépendamment de la connaissance des paramètres décrivant les formes physico-chimiques du Np 237 ou de l'Am 241 et des paramètres décrivant les aspects hydrologiques [8].

Enfin, cela nous a permis de faire des analyses de sensibilité paramétrique plus simple. Au lieu de fixer les paramètres à leur valeur et de faire varier le paramètre, dont on veut étudier la sensibilité, pour une suite de valeurs données, il suffit d'imposer, au paramètre en question, une distribution de probabilité uniforme à la place de sa distribution de probabilité expérimentale [9].

REFERENCES

[1] BITTEL R.
Radioprotection 1982, 17, (4) 217-224

[2] WILDUNG R.E., GARLAND T.R.
In : Transuranic Elements in the Environment
HANSON W.C. Ed., US Techn. Inform. Center, 1980, 300-335

[3] HENIN S. et al
Le profil cultural
Paris, MASSON, 1969

[4] DIEULIN et al
In : AIEA, Environmental migration of longlived Radionuclides
Vienne, 1982, 659-668

[5] SERNE R.J.
In : OWI-WISAP, 1977, 57-90

[6] ROCHON et al
In : AIEA, Underground Disposal of Radioactive Wastes
Otaniemi, 1979, 1980, II, 271-314

[7] EDWARDS L., HARVEY T.
Limiting Individual dose form an extended source - High level waste repositivy model
Lawrence Livermore National Laboratory UCRL-86849 (May 1982)

[8] CHEUNG F., EDWARDS L., HARVEY T., REVELLI M.
Post closure risk of alternative SRP nuclear waste forms in geologic reposotories
Lawrence Livermore National Laboratory UCRL-53269 (May 1982)

[9] IMAN R. and al
Risk methodology for geological disposal of radioactive waste. A distribution free approach to include rank correlation among input variables for simulation studies
SAND-80-0157 (1980)

ANALYSIS OF DIFFERENT VITREOUS MATRICES OF THE BOROSILICATE TYPE

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INTRODUCTION

Presently, the Argentine Nuclear Plan (1) contemplates the installation of four nuclear power plants, in addition to Atucha I and Embalse, which are already in operation, before the end of the present century.

These six NPPs will involve an installed capacity of 3.5 GW_e. The programme is based on natural uranium reactors and includes an industrial plant for the production of heavy water, whose construction is already well-advanced, as well as all the necessary installations aimed at completing the fuel cycle, including plutonium recycling.

Advanced studies are also being performed concerning the construction of a repository for the disposal of high-activity radioactive wastes in a granitic rock formation, with the objective of avoiding the occurrence of non-stochastic effects and of limiting the probability of occurrence of stochastic effects, while collective detriment is reduced as far as it is reasonable.(2)

In order to reach these goals, the eliminated wastes must be kept isolated from mankind for as long as it is necessary for the original radioactivity to decay sufficiently.

Considering that, in order to limit the individual risk, the container of these vitrified wastes must allow for an isolation of an order of 1,000 years, while disregarding other engineering barriers, several studies were performed concerning the capacity of some vitreous matrices to retain a simulated composition of high level wastes. Additionally, an analysis was made of the glass properties by means of evaluation techniques.

EXPERIMENTAL WORK PERFORMED

Preparation of Glass Containing Simulated Wastes

In a first step, the parent glasses were made, for their employment as frit in the final formulae. They were designed on the basis of three prototype compositions and the final formula was then analyzed after fritting. The various compositions are shown in Table 1. These vitreous matrices, after their grinding and sieving through a 120-mesh screen, were added 10% in weight of simulated wastes. The composition of the latter is shown in Table 2. This is a mixture of oxides that was obtained by denitration with formaldehyde and later drying and calcination at 700°C of a 6N nitric solution of the various elements. The simulated waste glasses (further on "waste glasses") were ellaborated in a muffle furnace at 1100°C for 8 hours inside ceramic crucibles.

Table 1

Composition of the parent glasses
(in weight percentages)

Type	SiO_2	B_2O_3	Al_2O_3	CaO	Na_2O	TiO_2	Fe_2O_3
A	42.2	20.0	14.43	0.73	14.49	0.10	1.20
B	39.4	22.0	14.90	8.67	8.58	0.10	0.22
C	49.7	12.4	2.23	3.62	25.72	4.70	0.30

Table 2

Composition of the simulated waste oxides
(in weight percentages)

RuO_2	6.89	La_2O_3	22.68	SrO	2.45	Cr_2O_3	0.71
BaO	4.42	MnO_2	2.00	TeO_2	1.82	NiO	0.71
CeO_2	7.95	MoO_3	13.39	ZrO_2	13.18	P_2O_5	6.46
Cs_2O	6.39	Rb_2O	0.99	Fe_2O_3	1.68	UO_2	8.28

Leaching Resistance Test

The method applied consisted in treating a 2 gr sample of ground glass, its granulometry being between 0.297 and 0.595 screen meshs, with 50 cm^3 distilled water at 98°C during one hour. The assessment of alcalinity was performed by means of a potentiometric titration with HCl 0.01 N. Additionally, a quantitative analysis was made of the leachates resulting from tests performed on the parent glasses. Table 3 shows the results obtained in the leaching tests.

Table 3

Results of the leaching tests

Sample	Parent glasses		Waste glasses	
	HCl 0.01 N (cm^3/g)	$meq/m^2 \cdot h$	HCl 0.01 N (cm^3/g)	$meq/m^2 \cdot h$
A	1.10	0.42	0.92	0.37
B	0.50	0.19	0.42	0.16
C	16.20	6.20	19.30	7.30

Table 4 shows the concentrations of the various parent glass components found in the leachates.

Table 4

Chemical analysis of the leachates
($\mu g/cm^3$)

Sample	Na	Ca	Mg	Zn*	Al	Fe*	Si	B
A	7	0.1	0.05	0	1.2	0	14	8.9
B	2	2.2	0.03	0	2.1	0	13	4.1
C	79	0.2	0	0	0.6	0	107	19.5

* Results below the detection limits.

Differential thermal analysis

By means of this technique and during the gradual heating of the glasses, the endothermal and exothermal peaks were assessed, corresponding with the transition and maximum crystal growth temperatures (Tg and Tk)(3,4). Each sample weighed 30 mg and a temperature of 900°C was reached at a speed of 10°C/min. The results obtained are shown in Table 5.

Table 5

Sample	Tg (°C)	Tk (°C)
A	560	710
B	590	750
C	530	850

X-ray diffraction

The diffractometric technique was applied on 400-mesh glass powder, with $K_{\alpha Cu}$ radiation. The studies were performed on parent and waste glasses and the crystalization degree of the various samples was verified.

The parent glasses were annealed, their temperature was rapidly lowered to 500° and then slowly lowered to room temperature, thus simulating the actual cooling of a container filled with waste glass.

The simulated waste glasses were annealed at 700°C for 120 hours in order to attain their devitrification.

The diffraction results obtained from parent glasses show a vitual amorphous condition, which is an indication of the actual vitreous quality of the analyzed material before simulated waste addition. The tests performed on the annealed waste glasses showed marked devitrification, while it was detected that the magnitude of this phenomenon is directly proportionate to the glass sodium content.

EVALUATION OF THE RESULTS AND CONCLUSIONS

As it had already been assumed, the studies on the leaching rate served to verify that there is a direct relation between this parameter and the glass alkali content. The samples that showed the lowest leaching rate are those of the B type, whose Na_2O content is also the lowest. There is also a direct relation between most of the component concentrations in the parent glasses and those in the leachates. The exception is boron, for which an inverse relation was observed.

It was also noticed that the addition of simulated wastes to the parent glasses does not modify substantially their solubility. The X-ray diffraction tests showed that the samples with the lowest crystallization tendency are the ones with the lowest alkali content and, in turn, those less liable to be leached. This is the case of samples of the B type.

An inverse and almost linear relation was verified between the transition temperatures, Tg, and the glass sodium content, as well as between the former and viscosity.

An analysis of the results allows for detecting clear advantages of sample B, as compared with samples A and C and, therefore, future studies will be aimed at the qualification of vitreous compounds similar to the B type.

BIBLIOGRAPHY

1. Castro Madero, C. "El Plan Nuclear Argentino y su proyección futura". Futurable (Buenos Aires) v.10: 25-39, 1981.

2. Migliori de Beninson, A. and Cancio, D. 1983 IAEA-CN-43/118

3. Mallow, G. and Schiewer, E. 1977 HMI-B217, Part I.

4. Jones, G.O. "Glass". 2. ed. London, Chapman & Hall, 1971. 128 p.

STATUS OF U.S. DEPARTMENT OF ENERGY RADIOACTIVE WASTE MANAGEMENT PROCEDURES

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INTRODUCTION

The Department of Energy (DOE) is preparing an Order which defines policies and guidelines for managing radioactive waste, waste byproducts, and surplus facilities. The broad policy framework of the Order is intended to cover all types of radioactive waste management by DOE and will help to provide assurance that the entire nuclear energy fuel cycle of the Department has been closed. The policy is also designed to aid in defining the transition to the disposal mode from the current practice of storing high-level waste (HLW) and TRU waste.

Historically, a uniform policy for waste management was first defined by the Atomic Energy Commission in 1973 with the issuance of a manual chapter on radioactive waste management.[1] The 1973 policy formally specified that waste contaminated with transuranium nuclides in excess of 10 nCi/g was to be separated from other radioactive waste and stored pending a decision on disposal (a practice which was first implemented in 1970) and provided operating criteria for managing HLW and radioactive solid waste other than HLW. Prior to that each government site developed its own radioactive waste management practices. In 1982, the DOE established 100 nCi/g as the level above which disposal in a geologic repository is necessary for TRU waste.[2]

The objective of the Order is to assure that all operations involving management of radioactive waste, waste byproducts, and surplus facilities are conducted to protect the public health and safety and according to the same nationally and internationally recognized radiation protection standards[3,4] that apply to all DOE nuclear operations. The Order is intended to provide protection comparable to that provided by Nuclear Regulatory Commission (NRC) regulations for licensed activities and conforms to applicable Environmental Protection Agency (EPA) standards.

Table I is a list of basic definitions used in the Order. In determining what constitutes radioactive waste, the "threshold quantity" concept is used and individual sites are permitted to establish these values based on guidance provided separately from the Order.[5] Figure 1 is a flow diagram for DOE's radioactive waste streams.

REQUIREMENTS

The requirements of the Order apply to all DOE nuclear operations that manage radioactive waste and surplus facilities including defense-related, nuclear energy development, and basic energy research programs and address waste storage, treatment and disposal techniques and decommissioning. Through the requirements, a framework is established within which individual sites can develop and implement radioactive waste management procedures. The procedures can take into consideration the waste generating operations as well as local climatological, hydrogeological, and demographic characteristics.

High-level Waste

High-level waste results from irradiated fuel reprocessing and contains at least 100 nCi/g of transuranium radionuclides as well as high concentrations of fission products. Three distinguishing characteristics of HLW are high toxicity, longevity, and high heat generating capacity. Storage and transfer

TABLE I. DEFINITIONS

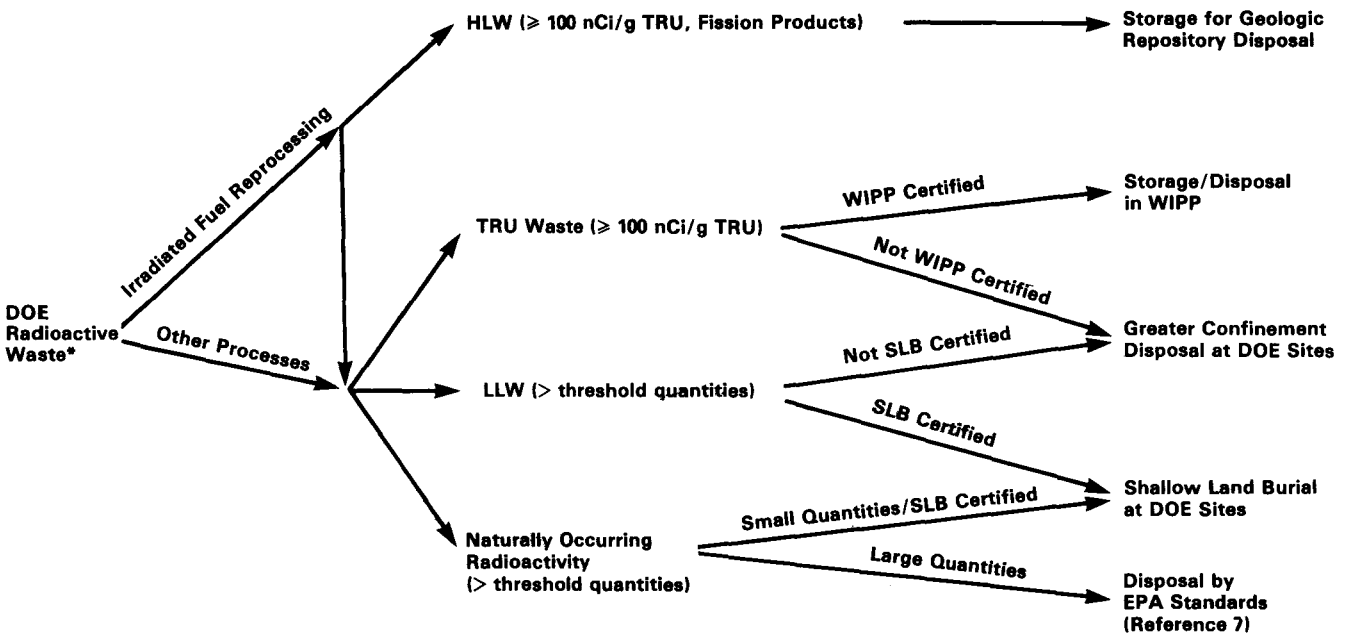
Byproduct Material. Waste produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content, including discrete surface waste resulting from uranium solution extraction processes.
Geological Repository. A deep underground mined cavity used for the disposal of radioactive waste.
Greater Confinement. A technique for waste disposal that employs natural and/or engineered barriers to provide a degree of isolation greater than shallow land burial but less than a geologic repository.
High-level Waste (HLW). The highly radioactive waste material that results from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid waste derived from the liquid, that contains a combination of TRU waste and fission products in concentrations as to require permanent isolation.
Low-level Waste (LLW). Any radioactive waste not classified as high-level waste, TRU waste, spent nuclear fuel, or byproduct material as defined by this Order.
Naturally Occurring Radionuclides. Radionuclides and their decay products which occur as part of the natural environment.
Radioactive Waste. Solid, liquid, or gaseous material of insignificant value which contains radionuclides in excess of threshold quantities.
Shallow Land Burial (SLB). Disposal of waste in near-surface excavations which are covered with a protective overburden.
Surplus Facility. Any facility or site (including equipment) that has no potential programmatic use and is radioactively contaminated to levels that require controlled access.
Threshold Quantity. A quantity or concentration of radioactivity above which waste must be managed as radioactive waste and below which the waste may be disposed of as nonradioactive waste at a DOE sanitary landfill.
TRU Waste. Without regard to source or form, radioactive waste that at the end of institutional control periods is contaminated with alpha-emitting transuranium radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/g.
Waste Byproducts. Material, other than special nuclear material, that can be separated and recovered from radioactive waste streams and made available for safe, environmentally acceptable, and cost-effective applications.

systems for liquid HLW must be doubly contained and have adequate monitoring and leak detection systems, and spare tank capacity for emergency situations. Recovery of waste byproducts and waste volume reduction are encouraged. Disposal shall be in a geologic repository established under the provisions of the Nuclear Waste Policy Act.[6] In exceptional cases, other disposal techniques that are technically and economically feasible and present lower public health risks may be considered after requisite environmental documentation.

TRU Waste

TRU waste is characterized by its high toxicity and longevity and contains at least 100 nCi/g of transuranium radionuclides at the end of institutional control periods (about 100 years). This concentration limit has gone into effect in the past 2 years at DOE facilities. The requirements specify that TRU waste is to be certified to meet disposal site acceptance criteria and that it shall be stored in preparation for disposal at the Waste Isolation Pilot Plant (WIPP), a facility located in New Mexico for conducting research and development to demonstrate safe disposal of radioactive waste from defense activities and programs of the United States exempted from regulation by the NRC.[7] Small

FIGURE 1. DOE RADIOACTIVE WASTE STORAGE AND DISPOSAL



*Waste that is below threshold quantities may be disposed of as nonradioactive waste.

quantities of TRU waste that do not meet waste acceptance criteria for WIPP may be disposed of by greater confinement. TRU waste that in the past was disposed of by shallow land burial is to be monitored and periodically assessed to determine the need for corrective action. Waste with less than 100 nCi/g of transuranium nuclides will be managed as LLW.

Low-level Waste

As defined, LLW means any radioactive waste not included in the other categories and therefore encompasses a wide range of materials, radioactivity concentrations and radionuclides. Disposal is by shallow land burial or in some cases by greater confinement. Each disposal site is to issue waste acceptance criteria which address topics such as allowable quantities and concentrations of radioactivity, mechanical stability of waste packages, restrictions for harmful gases, vapors, free liquids, hazardous chemicals, respirable substances, chelating and complexing agents, and other safety related characteristics. Each site shall also issue operating criteria which address topics such as protecting the public health and safety and the environment, site security, housekeeping and volume reduction practices, as well as operator training, first aid, emergency response plans, recordkeeping, access control, quality assurance, and other items. Sites shall also issue closure/post-closure criteria which specify final site stabilization techniques, residual soil contamination levels, identification markers, and long-term security, surveillance and maintenance. Guidance is also given for preparing selection and design criteria for new disposal sites.

Discharge of liquid LLW directly to the environment or on natural soil columns is to be replaced by other techniques such as solidification or in situ immobilization. Wastes containing only naturally occurring radionuclides including uranium or thorium and their decay products and byproduct material shall also be disposed of as LLW by shallow land burial or according to the intent of EPA standards.[8]

Decontamination and Decommissioning of Surplus Facilities

Surplus facilities are to be managed in a safe, cost-effective manner and structures, equipment, and valuable materials recovered for reuse as practical. The requirements specify procedures for conducting surveillance and maintenance, developing plans and priorities, implementing projects and releasing facilities for other use. They also include design features for new nuclear facilities that facilitate decommissioning.

REFERENCES

1. U.S. Atomic Energy Commission Manual Chapter 0511, "Radioactive Waste Management," September 19, 1973.
2. U.S. Department of Energy Order 5820.1, "Management of Transuranium Contaminated Material," September 30, 1982.
3. National Council on Radiation Protection and Measurements, "Basic Radiation Protection Criteria," NCRP Report No. 39, January 15, 1971.
4. "Recommendations of the International Commission on Radiological Protection," ICRP Publication 26, Vol. 1, No. 3, Pergamon Press, New York, NY (1977).
5. To be published, U.S. Department of Energy, Idaho Operations Office.
6. U.S. Public Law 97-429, Nuclear Waste Policy Act of 1982, January 7, 1983.
7. U.S. Public Law 96-164, Department of Energy National Security and Military Applications of Nuclear Energy Authorization Act of 1980, December 29, 1979.
8. "Standards for Remedial Actions at Inactive Uranium Processing Sites," U.S. Environmental Protection Agency, January 5, 1983, Federal Register, p. 590.

PROBLEMS CONNECTED WITH CHEMICAL AND PHYSICAL ANALYSIS OF LOW LEVEL RADIOACTIVE WASTE

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INTRODUCTION

Shallow land burial is the current method for the disposal and containment of low level radioactive waste. By far, the major portion of this waste consists of large volumes of nonradioactive inorganic and organic matrices. These matrices may be toxic, explosive, ignitable, pathogenic, or may contain complexing agents. Because of the potential for contamination of groundwater and possible health and safety hazards, it is necessary to chemically and physically analyze the inorganic and organic wastes prior to disposal. It is also necesaary to monitor these sites to confirm integrity of containment. Concern for the health and safety of waste site personnel and possible inadvertent intruder has prompted recent United States regulations such as 10CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Wastes."[1] These regulations formalized the necessity for chemical and physical assessment of radioactive waste. Other federal and state regulations detail the testing to be performed at land burial sites. Of particular interest, are old cocontaminated sites which come under the Comprehensive Environmental Response, Compensation and Liability Act of 1980 (Superfund). [2]

Sources of low level wastes are both from fuel and non-fuel cycle. Fuel cycle waste comprise 43% of the total and consist of uranium contaminated CaF_2, filters, ion exchange resins, filter sludges, contaminated clothing, evaporator concentrates and equipment. The other major constituent comprising 49% of the total comes from non-fuel cycle sources such as academic, medical and commercial sources.[3]

Card, et al, [4], has estimated that a two step decontamination of the primary system of a commercial reactor would require 36,000 kg's of chemicals. Among the chemicals employed are a number displaying a variety of hazards. These are oxidizing agents, reducing agents, sequestrants, inhibitors, and surfactants. Since these wastes are radioactively contaminated, it is necessary for analysts to institute special handling and precautions, in addition to those procedures required for carcinogenic, toxic or pathogenic materials.

Low level burial sites have, as part of their license, agreed not to accept waste which is more hazardous than the radioactive component. The hazard has been further defined in 10CFR61. As a result, laboratory determinations are required to analyze for and quantify possible radioactive and chemically hazardous components for declaration on the shipping manifest. Commercial sources licensed to receive radioactive samples, with the capability for performing the special analyses, are limited.

HEALTH PHYSICS

Prior to receipt of the waste samples, the submitting group provides EAL with information about the composition of the matrix material and the approximate activity levels of the radionuclides in the sample. Samples with surface external radiation levels less than 100 mR and alpha activity levels less than 1 µCi are accepted and require minimum health physics. Instructions on how to ship liquids and solids to conform to Department of Transporation regulations are provided. Special requirements for preservation and storage may be necessary. The samples are surveyed by a health physicist upon receipt for external contamination and to confirm the external radiation level. This procedure also assists the radiochemist in deciding on dilution factors with regard to handling and analyzing these samples under normal laboratory procedures, or by separate instruments in a segregated

laboratory. A separate consideration is the dilution required for conventional organic and inorganic analysis with regard to special handling due to hazardous physical, or chemical properties.

Glassware, electrodes, cuvettes, nebulizers, syringes, injectors and columns may be exposed to radioactive contamination. Whenever practical, disposable components are used. This includes pipettes, volumetric apparatus, and syringe tips. In most cases cleaning procedures prior to sample measurement will remove the major portion of any radioactive or conventional contaminant. Many of the instrument components which come in direct contact with the samples, such as GC columns, GC injector ports, nebulizer and flameless atomizer are readily replaceable. Instruments such as the total carbon analyzer, atomic absorption analyzer and the various gas chromatographs may accumulate radioactivity in their various components. Radiation surveys and wipe tests will provide the necessary hazard information. Other precautions to prevent the spread of undesirable contamination involve segregation of the residual and rinse solutions, checking of gaseous and particulate emissions, shielding as necessary, and protective clothing.

CHEMICAL/PHYSICAL ANALYSIS

In addition to their radioactivity, the samples that are submitted to the laboratory can be expected to possess such other hazardous properties as toxicity, reactivity, corrosivity, etc. Many waste materials are highly concentrated. The need for personnel protection is quite clear and is spelled out in such documents as National Research Council's "Prudent Practices for Handling Hazardous Chemicals in Laboratories."[5]

Analytical tests may require one or more of the instruments employed in an environmental test laboratory. The instruments commonly used in water and wastewater analysis include the following: analytical balance, pH/specific ion electrode (SIE) meter, conductivity meter, turbidimeter, atomic absorption spectrophotometer, spectrometers (visible, UV, IR), total carbon analyzer, gas chromatograph (GC), and gas chromatograph/mass spectrometer (GC/MS).

Instrumentation is quite sophisticated. The atomic absorption spectrophotometer is used for trace element analysis, the gas chromatograph for organics and gas chromatograph/mass spectrometer for priority pollutants. Tests for the various minor and major constituents must be according to approved methods. Quality control procedures ensure the credibility of the analyses.

Laboratory Facilities

The regulated access laboratory is designed to handle highly toxic and carcinogenic materials. This laboratory is equipped to permit samples to be examined, screened for various parameters (including radioactivity) which indicate hazard level and extracted and diluted, as required, so that the prepared extracts can be analyzed safely in an environmental laboratory.

The regulated access laboratory contains laminar flow hoods, glove boxes, explosion proof refrigerators and a few basic pieces of instrumentation for the screening tests mentioned above. Analytical balances and an assortment of graduated glassware are also available.

Laboratory air is exhausted exclusively through the hoods. Incoming air is limited so as to produce a net negative air pressure within the laboratory. Access is through an air lock which also serves as a personal decontamination and change area.

All glassware and equipment is thoroughly cleaned after each use to prevent cross contamination between samples.

Protective clothing is worn at all times while hazardous waste samples are being processed. The personnel that have been assigned to this function are medically screened once a year and received training in first aid and laboratory safety and safe handling of chemical carcinogens.

Laboratory Protocols

Laboratory tests are performed according to RCRA procedures as required. While pathogenicity, infectiousness and radioactivity were not addressed by RCRA, these parameters in waste materials are regulated under CERCLA and NRC regulations. Several parameters, including some defined under RCRA, i.e., reactivity and ignitability of solids, lack definitive test protocols. The Environmental Protection Agency is presently developing test procedures for toxic gas evolution and for solids ignitability. Action levels that define hazardous levels of these attributes will follow. The Nuclear Regulatory Commission defined parameters have neither test procedures or action levels at this time. The chemical toxic constituents under CERCLA and the levels of radioactivity are readily measured with the existing analytical methodology discussed above, but action levels defining hazardous class, have only been promulgated for a few highly toxic materials.

A comprehensive (and expensive) analytical protocol is always required when the waste material and its origins are unknown. Much testing can be avoided by a few simple steps:

- Segregation of waste types at their point of origin.
- Using separate containers for each waste type.
- Labeling at the point of origin of each waste type with all its components shown on the label.
- Proper manifesting and documentation during removal and disposal of the waste.

Before a waste is selected for analysis, an attempt should be made to determine the origin of the material. By establishing the process, or the activity that created a given waste type, one can learn a great deal about its probable constituents. Rather than having to screen for all hazard parameters, the laboratory can then concentrate on those that are known to be present and properly enumerate the concentrations and levels of the critical parts of these.

Individual chemical and instrumental tests are also performed. This could involve any procedure used in organic and inorganic analysis. In most cases these involve analysis of many parameters besides the specific hazardous waste tests. This is due to various state and federal laws regarding effluents, drinking water and groundwater.

COCONTAMINATED WASTE STUDIES

Cocontaminated waste studies have involved working with samples of wastes prior to disposal of corrosion products in water and sludges from nuclear utilities, spent catalyst from chemical firms, oils and solids from waste disposal firms, surface water, discharge water, leachates and groundwater from mine sites, laboratories, and low level burial sites. In most cases, these samples have not had significant amounts of radioactivity associated with them.

Typically, the samples do not have sufficient radioactive or hazardous waste information connected with them. Samples are usually soils, sludges, or liquids and arrive in a variety of packages. Our attitude is that every sample is hazardous. Our standard procedure is to survey the incoming samples externally with a beta-gamma survey meter. This determines if special handling is required. We then do a gross alpha and beta count on an aliquot. With regard to the hazardous waste content, our first indication as to hazardous properties is in appearance and odor.

The sample is then stored in a secondary container, either a plastic bag or can, in an explosion proof storage area. Aliquoting is performed in a hood.

Case I: Low Level Burial Ground

A number of well and ground water samples from a low level burial ground were received with a requirement that the EDTA, DTPA, volatile organics, ferrous and ferric iron, TOC, pH, and complete GC/MS and priority pollutants be determined. Screening of the samples for radioactivity indicated low external gamma and internal alpha and beta content. Chemical analysis indicated volatile organics primarily industrial solvents and metals ranging from 20 ppb to 10 ppm. We found that there were few procedures in the literature for the analysis of chelates, so we developed a new measurement technique. We equilibrated the EDTA and DTPA with an excess of copper to form the stable Cu chelate compound and then were able to determine the chelate content with the use of a Dionex Ion Chromatograph. [6] The chelate content was found to be negligible. Gloves for handling samples and other disposable equipment were used throughout.

CASE II: Nuclear Utility

Corrosion products in sludges and cooling water from a nuclear utility were analyzed for Cu, Cr, Co, Ni, Zn, Na, K, etc. The samples were surveyed for gamma and gross alpha and beta. Graphite furnace atomic absorption was chosen as the analysis method because of higher sensitivity over that of flame AA. The sample was diluted prior to aliquoting a few microliters for analysis. Additional cover was placed over the exhaust gas to enhance the venting. Dry residue tests involving total dissolved solids, oil and grease, total solids and volatile solids were handled with typical radiochemistry precautions.

Case III: Research Firm

Simulated wastes from a breeder coolant system were submitted. This comprised silicon oil and solids for U and gross alpha and Sr, Ce, Pb, Si, Na and trace elements. The sodium was metallic and reacted violently with water. Methods for handling the silicon oil were developed and the elemental analysis performed by graphite furnace AA as above. Isotopic uranium was determined by mass spectrometry.

CONCLUSIONS

1. Each sample presents its own special problems and requires individual care.
2. With appropriate health physics and industrial hygiene techniques there is no reason why chemically hazardous low level radioactive wastes cannot be successfully analyzed in a properly equipped laboratory.

REFERENCES

1. Licensing Requirements for Land Disposal of Nuclear Waste, 10CFR61. Federal Register, Vol. 47, No. 248, December 27, 1982.
2. Comprehensive Environmental Response, Compensation and Liability Act of 1980, Public Law 96-510.
3. Study of Chemical Toxicity of Low Level Waste, NUREG/CR1793, General Research Corp - Santa Barbara, California, March 1980, U.S. NRC Washington, D.C.
4. Card, C.J., Munson, L.F., Hoenes, G.R., Halseth, G.G., "Health Physics and Industrial Hygiene Aspects of Decontamination as a Precursor to Decommissioning," Proceedings, International Conference on Decontamination of Nuclear Facilities, Niagara Falls, CN, September 19-22, 1982.
5. Prudent Practices for Handling Chemicals in Laboratories, National Research Council, National Academy Press, Washington, D.C., 1980.
6. Ion Chromatograph, Dionex Model 12, Dionex Corporation, Sunnyvale, California.

INTENDED RETENTION OF RADIONUCLIDES FROM THE WASTE AIR OF REPROCESSING PLANTS

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The low emission rates of the nuclear power plants actually in operation are, besides other reasons, based on the fact that practically no defects of fuel elements occur. In reprocessing plants, during the dissolution all radioactive materials contained in the fuel are transferred either to the fuel-solution or to the dissolver-off gas (DOG).

The second column of table I shows the expected airborne emission rates of a reprocessing plant with an annual throughput of 1500 t of heavy metal, if no special retention facilities are installed. During chopping and leaching of the fuel elements, practically 100 % of the Kr 85 and more than 99 % of the C 14 (in the form of CO_2) are released into the DOG. More than 60 % of the tritium produced in the fuel is fixed in the fuel-rod cannings in the form of zirconiumhydrid. If the liquid effluents of tritium shall be kept low, the main release path way of tritium - including H 3 from the highly active waste (HAW) storage tanks - will be the waste air. Without any additional measures, only a small part of the I 129 is released into the DOG. This release raises up to more than 99 %, if a sufficient quantity of carrier gas is passed through the dissolver. Residual iodine remaining in the solution will be distributed in the whole process. Major sources of radionuclides bound on aerosol particles are the dissolution, the transport of process-fluids by airlifts or steamjets, the off gas of the process vessels (e.g. washers, pulsed columns, mixer settlers, centrifuges) and the gas produced by radiolysis. Assuming 10 mg of the process fluid being carried away with each m^3 of DOG and VOG, the emission rates without any additional retention facilities beside the off-gas condensers - which are installed in any case - will be about 37 TBq.

Table 1 shows the annual dose equivalents due to these emission rates assuming an emission height of 200 m in the northern part of the Federal Republic of Germany. The calculations are based on the data given in the guidelines of the Minister of the interior /1/. Only the emission of I 129 and aerosols results in annual doses which exceed the maximum permissible radiation exposure of the public in the Federal Republic of Germany, if we compare the effective dose with the limits of the total body. The maximum permissible annual dose equivalents in the radiation protection ordinance of the Federal Republic of Germany are 0.3 mSv for the total body, 1.8 mSv for bones and skin, and 0.9 mSv via the food chains for the thyroid. The doses due to H 3, C 14 and Kr 85 are about a factor of 25, 60 and 10, respectively, smaller than the limits. The doses due to globally distributed radionuclides are small, too. Assuming e.g. the reprocessing of fuel corresponding to a world wide installation of 1000 GW_e (which means about 1000 big power stations with light water reactors), the total emission of H 3 over a period of 50 years would result in a total body dose equivalent less than 10^{-5} mSv. Under the same conditions, the dose due to C 14 for the total body would be less than 0.001 mSv and the dose due to Kr 85 for the skin would be less than 0.01 mSv /2/. Mixing the total amount of I 129 with the surface water of the oceans would result in an annual dose equivalent for the thyroid due to globally distributed I 129 of about 10^{-4} mSv /2/.

For a first big reprocessing plant, which certainly will be necessary in Germany in this century, the following questions have to be answered: Should any retention facilities for H 3, C 14 and Kr 85 be installed; and which efforts should be made to retain I 129 and the aerosols? Normally, the permissible emission rates result from extensive discussions between the applicant, the authority, the experts, the radiation protection commission (SSK) and the public. In 1977 and 1983, the radiation protection commission made recommendations concerning the emission of radioactive materials from the now deferred "Entsorgungszentrum" /3/ and a smaller reprocessing plant /4/.

Meanwhile the ICRP recommended the use of a cost-benefit analysis to determine the retention levels for radioactive materials /5/. Following this method, it was

tried to find the optimum retention facilities. The tables 2 to 6 show the marginal cost-effectiveness values for the four radionuclides and the aerosols. The valuation of 1 man-Sv with 20 000 DM /6/ results in no retention of H 3 and Kr 85. 99 % of the C 14 and about 99.3 % of the Iodine 129 should be retained. Surprisingly, for the aerosol retention only washers and mist eliminators together with the off-gas condensers should be installed. However, safety considerations in the case of incidents and accidents make HEPA-filtration necessary in every off-gas system. The marginal cost effectiveness of the C 14-retention, for integration times of 500 as well as 10^6 years, is very small.

Due to the future reduction of the radiation exposure due to C 14 caused by the combustion of fossile fuel and due to the very small additional individual doses, the retention of C 14 seems to be unnecessary.

The question of tritium retention is strongly related to the site of the reprocessing plant. As the reprocessing plant planned in the Federal Republic of Germany will be situated far away from the coast, the emission rates with the waste water must be reduced with the aid of a H 3-scrubber. Following the German radiation protection ordinance, it is not allowed to emit the tritium retained by the scrubber by means of active measures.

The considerations made above are related to a big reprocessing plant with an annual throughput of 1500 t. At the moment, a smaller plant with an annual throughput of 350 t is under discussion in the Federal Republic of Germany /4/. The statements concerning the intended retention facilities are transferable to this smaller plant /7/.

/1/ Bundesminister des Innern:
Allgemeine Berechnungsgrundlagen für die Strahlenexposition bei radioaktiven Ableitungen mit der Abluft oder in Oberflächengewässer.
Gemeinsames Ministerialblatt 15.8.1979, p. 371 ff.

/2/ Bonka, H.:
Strahlenexposition durch radioaktive Emissionen aus kerntechnischen Anlagen im Normalbetrieb.
Verlag TÜV Rheinland (1982)

/3/ SSK, RSK:
Grundsätzliche sicherheitstechnische Realisierbarkeit des Entsorgungszentrums.
GRS, Köln (1977)

/4/ SSK:
Empfehlung der SSK zur Rückhaltung radioaktiver Stoffe bei einer Wiederaufarbeitungsanlage.
Bundesanzeiger 128 vom 14. Juli 1983

/5/ ICRP:
Publication 26.
Pergamon Press (1977)

/6/ Horn, H.G., et al.:
Kosten-Nutzen-Analyse zur Rückhaltung radioaktiver Stoffe in Kernkraftwerken und Wiederaufarbeitungsanlagen.
(publication in preparation)

/7/ Bonka, H., Horn, H.-G.:
Optimization of the Retention of Radioactive Material from the Airborne Effluents of Reprocessing Plants.
CEC-Seminar, Luxemburg, 8. und 9. 11. 1983

Table 1: Emission rates without additional retention facilities, individual dose equivalents and effective collective dose equivalents due to airborne emissions of a 1500 t/a reprocessing plant
Cooling time 1 a, composition of aerosols according to /6/

nuclide	annual emission without retention [TBq]	maximum annual dose equivalent [mSv]				total effective collective dose equivalent [man-Sv]		monetary valuation of the effective collective dose equivalent for α = 20 000 DM/man-Sv [DM]	
		effective	skin	thyroid	bone-surface	t = 500 a	t = 1 E6 a	t = 500 a	t = 1 E6 a
H 3	1.5 E4	0.012	0.012	0.012	0.012	55	55	1.1 E6	1.1 E6
C 14	26	0.005	0.005	0.002	0.003	410	3700	8.2 E6	7.4 E7
Kr 85	7.4 E5	0.015	0.16	0.013	0.013	290	290	5.8 E6	5.8 E6
J 129	2.2	1.4	<	41	<	1800	5700	3.6 E7	1.1 E8
aerosols	37	0.47	0.14	0.15	1.4	400	400	8 E6	8 E6

Table 2: Cost-benefit analysis for the retention of H 3

No.	retention process	retention efficiency [%]	annual costs for retention [DM/a]			total annual costs for retention and final disposal [DM/a]		marginal cost effectiveness $\frac{\Delta K}{\Delta S}$ [DM/man-Sv]	
			capital	operating	total	storage in geological formations	deep sea dumping	storage in geological formations	deep sea dumping
1	tritium scrubber and recycling	81	1 E5	1.6 E5	2.6 E5	2.7 E6	8.4 E6	61 000	1.9 E5
2	case 1 and separation from DOG	82	1.5 E5	5.1 E5	6.6 E5	3.1 E6	8.8 E6	7.3 E5	7.3 E5
3	case 1 and separation from VOG	84	1.5 E5	5.1 E5	6.6 E5	3.1 E6	8.8 E6	2.4 E5	2.4 E5
4	case 1 and separation from HAW-depot off-gas	95	3.2 E5	7.7 E5	1.1 E6	3.5 E6	9.2 E6	1 E5	1 E5

Table 3: Cost-benefit analysis for the retention of C 14

No.	with or without Kr 85 retention	retention-process	retention efficiency [%]	annual costs [DM/a]				marginal cost effectiveness $\Delta K/\Delta S$ [DM/man-Sv]	
				retention		final disposal	total		
				capital	operating	operating		t = 500 a	t = 1 E6 a
1	without	highly efficient scrubbing	99	4 E5	3.5 E5	0.6 E5	8.1 E5	2 000	220
2	with	simple scrubbing	81	4 E5	3.5 E5	0.6 E5	8.1 E5	2 500	280
3		highly efficient scrubbing	89	5 E5	3.5 E5	0.6 E5	9.1 E5	3 000	340
4		oxidation of CO and C_nH_m and highly efficient scrubbing	98	6 E5	1 E6	0.6 E5	1.7 E6	21 000	2 400

Table 4: Cost-benefit analysis for the retention of Kr 85

No.	retention and final disposal process	retention efficiency [%]	annual costs [DM/a]					cost effectiveness $\frac{\Delta K}{\Delta S}$ [DM/man-Sv]
			retention		final disposal		total	
			capital	operating	capital	operating		
1	low temperature rectification and above ground storage	96	2 E6	2 E6	5 E6	2 E6	1.1 E7	40 000
2	low temperature rectification and deap sea dumping	96	2 E6	2 E6	6 E5	2.5 E6	7.1 E6	25 000
3	acc. to 2, capital costs following an industrial estimation	96	5 E6	2 E6	6 E5	2.5 E6	1 E7	36 000

Table 5: Cost-benefit analysis for the retention of I 129

No.	retention process	retention efficiency [%]	annual costs [DM/a]				marginal cost effectiveness $\Delta K / \Delta S$ [DM/man-Sv]	
			retention		final disposal	total		
			capital	operating	operating		t = 500 a	t = 1 E6 a
1	retention from DOG	99.3	0.8 E6	1.2 E6	0.1 E6	2.1 E6	1200	370
2	retention from DOG and VOG from extraction stages	99.6	1.6 E6	1.3 E6	0.1 E6	3.0 E6	1.7 E5	5.3 E4
3	retention from DOG, VOG from extraction stages and VOG from waste systems	99.9	3.5 E6	1.4 E6	0.1 E6	5.0 E6	3.7 E5	1.2 E5

Table 6: Cost-benefit analysis for the retention of radionuclides bound on aerosol particles

No.	retention process	retention efficiency [%]		$\frac{S}{S_0}$	annual costs [DM/a]				marginal cost effectiveness $\frac{\Delta K}{\Delta S}$ [DM/man-Sv]
					retention		final disposal	total	
		β-aerosols	α-aerosols		capital	operating	operating		
0	condensor, washer and mist-eliminator in all off-gas systems (necessary in any case)	99	99	1	3.0 E6	0.5 E6	<	3.5 E6	88
1	measure 0 +prefilter and HEPA in DOG	99.4	99.3	0.6	4.0 E6	0.5 E6	5 E4	4.6 E6	6900
2	measure 1 +prefilter and 1 HEPA in VOG of waste vitrification	99.8	99.3	0.25	7.0 E6	1.2 E6	8 E 4	8.3 E6	26 000
3	measure 2 +prefilter and 1 HEPA in VOG of 1st extraction	99.9	99.9	0.1	9.0 E6	1.25 E6	9 E4	10.4 E6	35 000
4	measure 3 +prefilter and 1 HEPA in VOG of 2nd/3rd extraction	99.92	99.999	0.05	10 E6	1.3 E6	9 E4	11.4 E6	50 000
5	measure 0 +prefilter and 1 HEPA in all off-gas systems	99.999	99.999	0.001	12 E6	1.7 E6	1 E5	13.8 E6	120 000

DISPOSAL OF THORIUM AND URANIUM WASTES

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INTRODUCTION

Some of the sites formerly used for processing thorium and uranium in the United States are known today to be contaminated with residual radioactive materials. In many cases, the total amount of contaminated soil is large, but the activity concentrations of radioactive materials present are sufficiently low as to justify their disposal on privately owned lands or storage onsite rather than transport them offsite to a licensed radioactive materials disposal (commercial) site. In many instances, packaging and transporting these wastes to a licensed disposal site would be too costly and not justified from the standpoints of risk to the public health or cost-benefit.

The U.S. Nuclear Regulatory Commission (USNRC) in late 1981, issued a Branch Technical Position which discussed options for approving license applications for disposal of uranium and thorium under USNRC regulations. In the following sections, the technical bases, rationale and dose calculations for establishing the limiting concentrations in wastes under each disposal option are discussed. The Branch Technical Position has been used since 1981 for reviewing the acceptability of disposal methods proposed by U.S. organizations.

DISPOSAL AND STORAGE OPTIONS

The following describes the established disposal and storage options for uranium and thorium wastes. A summary of the maximum concentrations permitted under each disposal option is provided in Table I.

1. Disposal of acceptably low concentrations of natural thorium, depleted or enriched uranium, and uranium ores with no restriction on burial method.

The concentrations specified for this option are believed acceptably low without restricting the method of burial. It is expected, however, that facility operations will be conducted in such a manner as to minimize the possibility of soil contamination and when such occurs the contamination will be reduced to levels as low as reasonably achievable.

2. Disposal of certain low concentrations of natural thorium and depleted or enriched uranium (with essentially no daughters present) when buried under prescribed conditions with no land use restrictions and no continuing NRC licensing of the material.

Under this option, burial will be permitted only if it can be demonstrated that the buried materials will be stabilized in place and not be transported away from the site. Acceptability of the site for disposal will depend on topographical, geological, hydrological and meteorological characteristics of the site. At a minimum, burial depth will be at least four feet below the surface. Also, recorded title documents are expected to state that the specified land contains buried radioactive materials.

3. Disposal of low concentrations of natural uranium ores with daughters in equilibrium, when buried under prescribed conditions in areas zoned for industrial use, and the recorded title documents are amended to state that the specified land contains buried radioactive materials and are conditioned in the manner acceptable under state law to impose a covenant running with the land that the specified land may not be used for residential building. (There is no continuing NRC licensing of the material.)

Disposal will be approved if the burial criteria outlined in Option 2 (including burial at a minimum of four feet) are met. Under this option, no residential building would be permitted over land where natural uranium ore residues have been buried.

4. Disposal of land-use-limited concentrations of natural thorium, natural uranium and depleted or enriched uranium when buried under prescribed conditions in areas zoned for industrial use and the recorded title documents are amended to state that the land contains buried radioactive materials, and are conditioned in the manner required by state law to impose a covenant running with the land that the land (1) may not be excavated below stated depths in specified areas unless cleared by appropriate health authorities, (2) may not be used for residential or industrial building, and (3) may not be used for agricultural purposes. (There is no continuing NRC licensing of the disposal site.)

Criteria for disposal under these conditions is predicated upon the assumption that intentional intrusion is less likely to occur if appropriate warnings are given in land documents of record. In addition to meeting the burial criteria in Option 2, recorded title documents must be amended to impose these land use restrictions. Also, it is expected that disposal under this option will normally be approved only if disposal is carried out on publicly owned land.

5. Storage of licensed concentrations of thorium and uranium onsite pending the availability of an appropriate disposal site.

When concentrations exceed those specified in Option 4 disposal is normally carried out on publicly owned land, which is subject to continuing monitoring and government inspection. For an interim period, thorium and uranium may be stored onsite under an NRC license until a suitable method of disposal is found.

DOSE LIMITS FOR THE DISPOSAL OPTIONS

The dose limits in Option 1 are based on the clean-up guidance from the U.S. Environmental Protection Agency (EPA) for similar situations. For natural uranium, EPA's disposal standard is 5 picocuries of radium 226 per gram of soil at the soil surface and 15 picocuries per gram at soil depths below 15 centimeters. In addition, concentration limits for uranium and thorium were calculated based on limiting the annual intakes from the inhalation and ingestion pathways to maximum doses of 20 millirems to lung and 60 millirems to bone as recommended in an EPA standard for protection against transuranium elements present in the environment. For direct radiation, an external dose limit of 10 microroentgens per hour above background was used. The selection of these dose guidelines was influenced by the NRC policy of maintaining radiation exposures as low as reasonably achievable.

The dose limits in Option 2 are such that (1) no member of the public should receive a radiation dose exceeding those discussed in Option 1 when the wastes are buried in an approved manner absent intrusion into the burial grounds and (2) no member of the public would likely receive a dose in excess of 170 millirems to a critical organ in the event of a possible intrusion into the burial ground.

The dose limit in Option 3 involving the disposal of low level natural uranium wastes (with daughters in equilibrium) is based on appropriately limiting indoor exposure of radon-222 and its daughters. Under the limited land use restriction of this option, no individual will likely be exposed from indoor radon-222 and its daughters to an annual average of 0.5 working level month (WLM).

The dose limit in Option 4 is based on a limited exposure under a restrictive land use requirement such that an individual is not expected to receive a dose in excess of 500 millirems to a critical organ and radon doses will not exceed 0.5 WLM as a result of intruding into the disposal location.

DOSE CALCULATION AND DERIVED MAXIMUM ALLOWABLE CONCENTRATIONS FOR VARIOUS DISPOSAL OPTIONS

From the above dose limits, calculations were performed to derive the concentration limits for each disposal option. With regard to concentration limits based on external exposure, calculations were based on the methodology and dose conversion factors taken from ORNL-4992 "A Methodology for Calculating Radiation Dose from Radioactivity Released to the Environment."

The calculations assumed (1) distribution of radioactive material over an infinite plane (2) a structural shielding factor of 0.5, (3) a maximum occupancy factor of 80 percent and (4) an exposure dose delivered at one meter from the surface.

With regard to concentration limits based on internal exposure, the same methodology as in ORNL-4992 was used. For inhalation, the dose conversion factors for the Task Group Lung Model were taken

from ORNL/ NUREG/TM-190. In the case of inhalation of radon and its daughters, the limiting radium concentrations were based on a radon dose less than that given by continuous exposure to 0.02 working level. The calculations assumed (1) a maximum dry soil density of 2.5 gm/cc, (2) a resuspension factor of $5 \times 10^{-9} m^{-1}$ in the case of airborne particulates and an average particle size of one micron (AMAD=1um), (3) a quality factor of 20 for alpha particles, (4) an occupancy factor of 80 percent and (5) 60 percent of an individual's food intake could be harvested from contaminated soil.

CONCLUSION

The disposal and storage options for uranium and thorium discussed in this paper are believed appropriate to assure proper protection of the general public and are believed also cost effective. The disposal options provide viable and safe alternatives to continued storage of low-level radioactive wastes and permit reuse of land that would otherwise be unavailable for productive purposes.

TABLE 1

SUMMARY OF MAXIMUM CONCENTRATIONS PERMITTED UNDER DISPOSAL OPTIONS

	Disposal Options (pCi/gm)			
Kind of Material	1[a]	2[b]	3[c]	4[d]
Natural Thorium (Th-232 + Th-228) with daughters present and in equilibrium	10	50	--	500[e]
Natural Uranium (U-238 + U-234) with daughters present and in equilibrium	10	--	40	200[e]
Depleted Uranium				
o Soluble	35	100	--	1000[e]
o Insoluble	35	300	--	3000
Enriched Uranium				
o Soluble	30	100	--	1000[e]
o Insoluble	30	250	--	2500

a. Based on U.S. Environmental Protection Agency (USEPA) cleanup standards.
b. Concentrations based on limiting individual doses to 170 mrem from an annual exposure.
c. Concentrations based on limiting equivalent exposure to 0.02 working level or less.
d. Concentrations based on limiting individual doses to 500 mrems and, in case of natural uranium, limiting exposure to 0.02 working level or less.
e. We are considering limitation of the concentrations of soluble materials to the same concentrations specified in Options 2 and 3 in order to minimize potential leaching of materials into groundwater.

IMPACT OF 20 YEARS LIQUID WASTE MANAGEMENT PRACTICES ON ENVIRONMENTAL SAFETY AT BRADWELL NUCLEAR POWER STATION

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Introduction

Bradwell Nuclear Power Station is one of the first of the UK commercial Gas Cooled Magnox Stations and began operation in 1962. Even then it was policy in the UK to control radioactive waste discharges so that not only were doses within a specified upper limit but also conformed to the principle of being as low as reasonably achievable. This has resulted in a very close liaison between the operator, the Central Electricity Generating Board (CEGB), and the authorising mininstries, particularly the Ministry of Agriculture Fisheries and Food (MAFF) in relation to discharges of liquid radioactive effluents into the marine environment, in the case of Bradwell the tidal Blackwater Estuary.

Exposure Pathways

Assessment of exposure pathways provides the basis for both discharge control and environmental monitoring. The composition of the critical group and the nature of the discharge may change with time and require a modification of the monitoring programme. This is illustrated by the following analysis of the changes which have occurred at Bradwell over 20 years. This paper updates an earlier paper covering the first 10 years. (Wasson & Mitchell, 1973).

Environmental Monitoring

Environmental monitoring is regularly carried out by both CEGB and MAFF working independently. The major locally caught foodstuffs are analysed for total beta radioactivity, and by gamma spectrometry. In addition seaweed and silt are also analysed as indicators, and gamma dose rate measurements are made on the foreshore to assess the external dose to members of the public. Habits surveys are conducted by MAFF at intervals to identify the critical group, which initially consisted of oyster consumers, to be replaced by fish eaters from 1975.

Fig.1 shows the concentrations of the relevant radionuclides measured by MAFF in oysters from 1964 and fish from 1975, all from within the Blackwater Estuary. Successive peak concentrations of ^{65}Zn, ^{134} + ^{137}Cs and ^{110m}Ag occurred in oyster flesh during 1967, 1968 and 1971 respectively and are attributed primarily to the Station discharges. By 1976 the concentration of ^{110m}Ag had fallen to zero and that of ^{65}Zn had also become low. These trends together with an increase in radioactive caesium in fish flesh after 1975 resulted in the fish eaters becoming the critical group from then onwards. This increase coincided not only with an increase in the amount of radiocaesium discharged from the station but came at a time when the concentrations in the southern North Sea were beginning to increase as a result of discharges from the fuel reprocessing plant at Sellafield being

carried round the north of Scotland by the prevailing current system. The picture is additionally complicated by the presence of ^{137}Cs from bomb testing fallout.

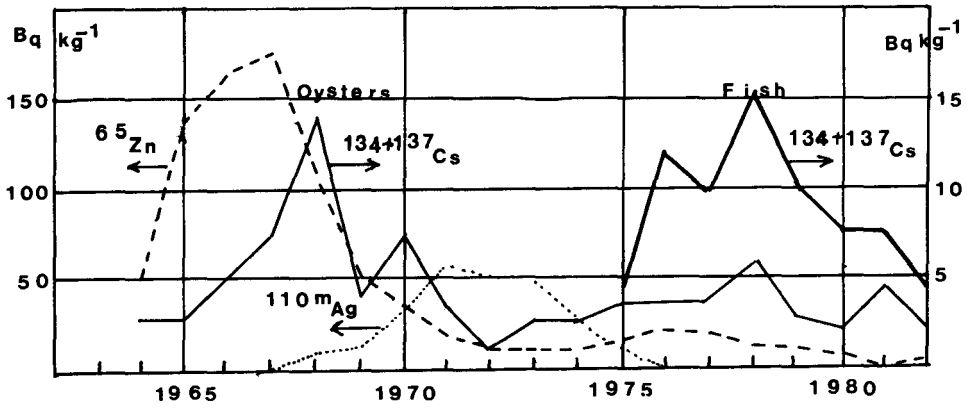

Fig.1:The concentration of ^{65}Zn, ^{110m}Ag and $^{137}+ ^{134}Cs$ in oyster flesh and $^{137} + ^{134}$ Cs in fish flesh from the Blackwater Estuary. Bq kg $^{-1}$.

Apportionment of the nuclides in oyster flesh is relatively easy since the species is non mobile. No Sellafield derived ^{65}Zn or ^{110m}Ag would be measurable in the Blackwater but a MAFF study of caesium in the estuary and the adjacent waters has indicated that during 1973-79 approximately 80%, 17% and 3% would have come from the Power Station, Sellafield and fallout respectively. In the case of fish, migration of individuals between the estuary and the southern North Sea makes apportionment almost impossible but a considerable fraction of the caesium measured has probably come from Sellafield.

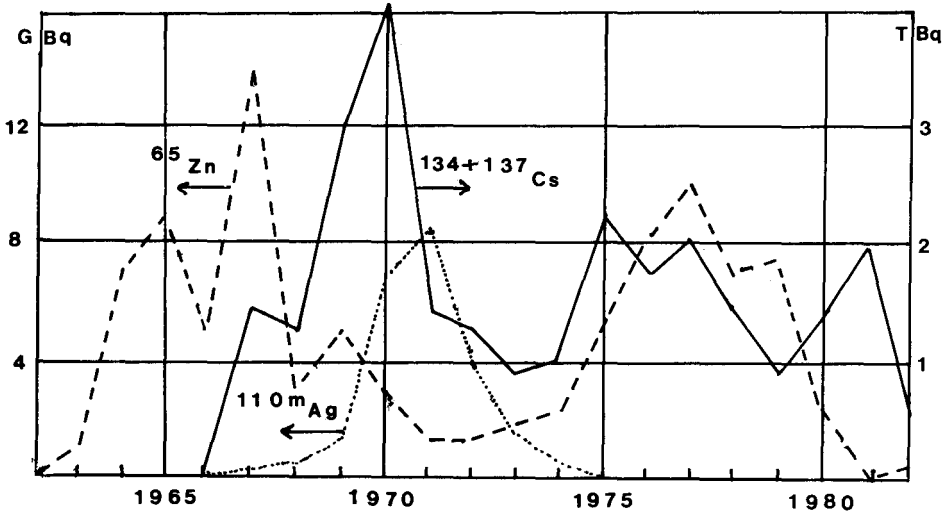

Fig.2:The annual discharges of ^{65}Zn, ^{110m}Ag and ^{137}Cs in liquid radioactive waste from Bradwell Power Station. G Bq/TBq.

Control of Discharges

There are three main sources of liquid effluent which are, in order of importance, the irradiated fuel storage pond, decontamination and reactor gas driers. Figure 2 shows the annual discharges for the radionuclides of radiological importance. Although the exposures involved are all significantly below the maximum permitted levels the operator has been able to respond in a way which has kept these as low as reasonably achievable.

The pre-operational assessment identified oyster consumers as the potential critical pathway and ^{65}Zn as the critical radionuclide. A special ion-exchange unit was included in the Pond Water Treatment Plant specifically to reduce the amounts of this nuclide before it reached the effluent system. In practice the levels of ^{65}Zn in the pond were substantially lower than predicted however the plant was operated in its design mode until 1967. In that year there was a significant increase in the levels of $^{134} + ^{137}Cs$ in the pond water and hence in the effluent. Although at that time these were not considered critical nuclides steps were taken to control their release by altering the operational regime for handling irradiated fuel to prevent mechanical damage which was allowing pond water access through the magnox cladding, at the same time replacing the original zinc removal resin in the pre-treatment unit with one which absorbs caesium. It is interesting to note that there was a simultaneous decrease in the amount of ^{65}Zn discharged. At the same time as the caesium was increasing another isotope, ^{110m}Ag, appeared and increased to such a degree that it surplanted ^{65}Zn as the critical nuclide in oysters. Investigations by CEGB identified the source of this nuclide, not as a fission product but as an activation product resulting from the presence of silver in a commercial chromating bath used to treat a batch of fuel element can components. As a result of this investigation the treatment was stopped. Although levels of ^{110m}Ag in the effluent continued to rise for a time they eventually declined as the affected batch of fuel passed through the system.

It should also be noted that while the discharges of ^{65}Zn from the station rose to a second maximum in 1977 this was barely reflected in the levels in oyster flesh. A knowledge of the concentration of stable zinc in the estuary over the whole period would be required before attempting to explain this apparent anomaly.

Public Radiation Exposure

The internal doses to the critical groups from the consumption of oysters and locally caught fish (figure 3) have been derived from ICRP 30 data. Note that Wasson and Mitchell (1973) employed an oyster consumption rate of $75gd^{-1}$ throughout, based on the highest individual consumer. The present critical group parameters have been derived using current MAFF techniques (Hunt et al 1982). The figure shows the decline in the dose to the oyster consumers after 1967 and the increase in the importance of the fish eating group after 1975. The largest dose to the critical group from the nuclides under consideration occurred in 1978, but this was still only 0.45% of the ICRP dose limit of 5 $mSv\ y^{-1}$ to members of the public. Other potential pathways such as external dose or consumption of other shellfish species are constantly reviewed but to date have been considered to be of negligible importance.

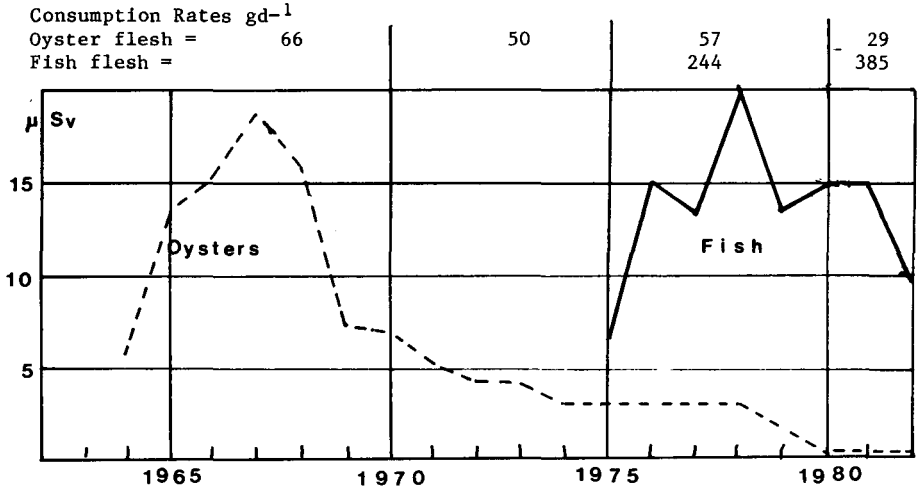

Fig.3 The internal dose to members of the critical group from the consumption of oyster and fish flesh. micro Sv.

Conclusion

The proper application of ALARA requires close liaison to be maintained between the operator and the authorising government departments so that site activities and the resulting waste discharges may be related to their impact on the environment. Inter alia this requires that the environmental monitoring programme and evaluation of pathways be kept under review, coupled with the ability to identify factors on site responsible for changes in dose and to take corrective action where necessary.

Acknowledgement

The authors thank Messrs D. Jefferies and W. Camplin for their assistance for analysing the oysters and calculation of doses.

References

Wasson M.M. and Mitchell N.T.
"Impact of ten years liquid waste management practices on environmental safety at Bradwell Power Station"
Proceedings of the Third IRPA Congress, Washington, 1973.

Hunt G.J., Hewitt C.J. and Shepherd J.G.
"The identification of critical groups and its application to fish and shellfish consumers in the coastal area of the North East Irish Sea"
Health Physics 43 (6) 875-89. 1982

INCINERATION OF LOW-LEVEL RADIOACTIVE WASTE AND SCINTILLATION VIALS UNDER RESOURCE CONSERVATION AND RECOVERY ACT

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Incineration is the most responsible disposal procedure for low-level radioactive waste and scintillation vials. Although scintillation vials containing C-14 and H-3 in amounts less than 0.05 μCi/gm. are not regarded by the Nuclear Regulatory Commission (NRC) as radioactive waste, the toluene and 1,4-dioxane, frequently used as the solvent base for scintillation media, are listed by the Environmental Protection Agency (EPA) as hazardous.

The incinerator was designed to be capable of handling low-level radioactive, pathological and hazardous waste. Ash demonstrated free from residual radioactivity and/or hazardous constituents (scintillation media) can be removed to a sanitary landfill. Incineration reduces the volume of waste and renders it non-radioactive and non-hazardous because the products of combustion are removed thru the stack and by chemical decomposition as defined by the EPA and confirmed by ash analysis. Proper incinerator design parameters achieved this result.

Responsibility for the proper design of the incinerator belonged to the Engineer working with calculations provided by the Health Physicist. Engineers chose the manufacturer, Environmental Control Products Inc. (ECP), based on personal contact with owners of incinerators and visits to several manufacturer's sites. Their final choice was based on capability to dispose of waste without exceeding regulatory limits. The manufacturer guaranteed that the incinerator would meet all Resource Conservation and Recovery Act (RCRA) requirements such as a two second residency time in the upper chamber to achieve 99.99% combustion. They would custom build an incinerator to specifications such as construction of the upper chamber four feet longer than the lower chamber to increase combustion efficiency by increasing the residency time. Before shipment, the manufacturer conducted a complete test of their product to verify that specifications had been met.

To determine the correct size incinerator, a comparison study on two previous years waste records was carried out. This provided an estimated growth in the use of radionuclides. A 10% yearly increase, extrapolated on a linear curve for twenty years was used. 2000°F was chosen for the upper chamber and 1850°F for the lower chamber as the specification temperatures. These higher temperatures with the longer residency time in the upper chamber met all RCRA requirements for complete destruction by incineration of hazardous material. Since plans for the future included applying for a Part B Permit for hazardous waste solvent disposal, a flange was incorporated in the side of the lower chamber so a liquid fuel injection system could be added.

Although the ECP 750T model (750 lbs/hour average burn rate) met all criteria, a larger model, the 1000T (1000 lbs/hour) was chosen and modified to specifications so radioactive waste disposal only required approximately one to two hours of burn time per week. On days when this type of waste was disposed of, the incinerator operated at least eight hours so the required dilution factor could be achieved and not exceed the maximum allowable regulatory release to an unrestricted area (top of the stack). The larger incinerator allowed use of pathological waste as

charging material during these hours of operation. Also, the larger size required only one work shift.

Choosing the 1000T instead of the 750T increased the cost payback by less than one month, from 10.5 months to 11.4 months. This cost payback was based solely on disposal of low-level radioactive waste and not on the significant saving associated with incineration of pathological waste.

The license application preparation began with an architect's drawing showing the location of the incinerator in the new toxicology building and its proximity to the main building and the surrounding area with general information on the incinerator. The decay calculations for short-lived radionuclides included the entire computation for remaining radioactivity at the time of removal from the laboratories, for ease of review by regulatory personnel. Amounts of less than 1.0 μCi either before or after decay were considered non-radioactive and incinerated. Calculations were made for maximum (up to 50 mCi) and minimum (0.25 mCi) amounts received as verified by Burroughs Wellcome Co.(BW Co.) inventory records of receipt of radioactive material. The information thus generated was put in a tabular format to expedite examination.

The following items were included in the license application: a summary of incineration calculations at 2000°F showing maximum activity per burn vs. maximum regulatory limit with fraction of regulatory limit reached for 8, 10, 16 and 24 hour burns for C-14, H-3 and S-35; Pasquill Equation for ground level radiation to hypothetical man; the new radioactive waste management section of the corporate radiation safety handbook; the steady state environment inventory world wide, for Durham County and for the air above BW Co., calculated using surface area of the earth as $4\pi r^2$ with r = 3959 miles and world wide production of H-3 equal to 1.9×10^6 Curies and for C-14 equal to 3.8×10^4 Curies.

There were three State regulatory agencies concerned with the adjudication of the license and permit process. 1) The North Carolina (NC) Dept. of Human Resources, Radiation Protection Branch (Agreement State-license), 2) NC Dept. of Human Resources, Division of Health Services, Solid and Hazardous Waste Management Branch (RCRA requirements) and 3) NC Dept. of Natural Resources and Community Development, Division of Environmental Management, Air Pollution Branch (Emission Control Standards-Air Permit).

Permission to incinerate pathological and low-level radioactive waste was requested, and the required engineering background data to the Air Pollution Branch to obtain a permit to construct and operate an incinerator was submitted. This authorization was received expeditiously. It permitted incineration of 800 lbs. per hour of Types 0, I, III, and IV waste as requested, provided a successful test burn for particulate emissions was passed within 90 days after the unit became operational.

During the interim between submission and receipt of the license and air permit, a significant regulatory change took place. The NRC published a change in 10 CFR Part 20 in which certain radioactive, biomedical wastes were deregulated. This deregulation included scintillation vials containing C-14 and H-3. As a result, the vials, due to their ignitability and for toluene and 1,4-dioxane, also their toxicity, now fell under RCRA as hazardous waste. Because of the uncertainty of what approach would be taken in incinerating the vials as a hazardous waste and the limited

time available to coordinate the process with state representatives, the original particulate test was scheduled with review of the burning of vials to be done at a later date.

The first test burn was conducted on December 4, 1981. To maintain the permitted burning rate of 800 pounds per hour, 200 pounds was charged every 15 minutes. The burn exceeded 1.60 pounds per hour of particulate emissions and was failed. On February 3, 1982, the second test burn was conducted. During this test, 80 pounds were charged every six minutes. At 1.16 pounds per hour of particulate emissions this test was passed. By charging waste every 15 minutes, a larger volume of waste was put into the ignition chamber which was allowed to burn for a longer period of time. The ash on top of the charged waste was burning to a finer particle size. When the next load was charged, the fine ash was entrained in the exhaust gas and a greater amount was vented out the stack.

The covering letter for the license submission stipulated that when complete combustion occurs, all C-14, H-3 and S-35 go up the stack as products of this activity as radioactive CO_2, H_2O and SO_2. The following procedure was outlined and executed to confirm this hypothesis.

For one month before any radioactive waste was incinerated, only pathological waste was burned. After each ash removal, an analyses for residual C-14, H-3 and S-35 was carried out to establish background numbers for this particular incinerator. It is a fact that animal carcasses have naturally occurring K-40 and Ra-226. In addition, the refractory material would also have background activity. After the background numbers were established, radioactive waste was burned and repeat ash analyses done. The results hopefully would show that there was no residual activity remaining and therefore the ash was non-radioactive. What had been hypothesized was confirmed and permission granted in writing for the ash to go to the county landfill. The approval came within twenty four hours after the findings were submitted to the regulators.

With an operating permit in hand and established, acceptable means of disposing of solid and NRC regulated wastes, it was time to look into the incineration of the scintillation vials as a hazardous waste. Because the incinerator was new, it came under the RCRA requirements. In order to incinerate the vials a Part B application would have to be initiated and a trial burn conducted. In addition, the operating permit had to be amended to allow the burning of flammable solvents (Type V wastes). The approach here was to gather all available data and bring representatives from both regulatory agencies (Solid and Hazardous Waste Mangement Branch and Air Pollution Branch) together for a meeting. This was done and, in addition, the person who normally reviews and writes permits was invited. The State reviewed what data had already been collected. It was determined that in as much as the average BTU/pound of the liquid in the vials was greater than 10,000, a request for authorization to burn the vials as an auxiliary fuel source should be submitted.

The meeting was followed up with a written request to the State to burn the vials as an auxiliary fuel. This request was approved. The required data was submitted to the State to amend our initial operating permit. In this letter authorization was requested to burn 80 pounds per hour of scintillation vials, not to exceed 320 pounds per day (2 hours to develop an ash bed, 4

hours of burn time). The solvents in the vials would consist of xylene, toluene and 1,4-dioxane.

The Air Pollution Branch advised us that another test burn was required. A meeting was scheduled with them to ascertain what would be required to be analyzed for during the burn. It was agreed that in as much as this was the first unit of its kind in the area, the test burn should be analyzed for: particulates, heavy metals, vinyl chloride, asbestos and PCB's.

Two test burns would be conducted. On July 20, 1982 only scintillation vials were incinerated, charging 8 pounds every six minutes. On July 21, 1982 the standard solid waste feed (720 pounds per hour) with 80 pounds per hour of vials was incinerated. The results of the test burns were submitted to the State for review. They developed a computer model of the stack based on local meterological data compiled by the U.S. Weather Bureau at the local airport to determine the maximum ground level impact. They compared these data against established ambient air standards, or where these were not available, established OSHA standards.

In accordance with the RCRA regulations, the ash was handled as a hazardous waste until it had been analyzed and delisted. Since there was no on-site capability to conduct an extraction procedure toxicity analysis of the ash, the State Solid and Hazardous Waste Management Branch performed this test. The only test that remained to be done and pass was to analyze the ash for residual xylene, toluene and 1,4-dioxane.

For three consecutive weeks a homogeneous representation of ash was removed after burning vials, pathological and low-level radioactive waste at 1600°F in the lower chamber and 2000°F in the upper chamber. Each sample was thoroughly mixed, then raked smooth on a grid system and an equal sample removed from each grid square, for each weeks burn. On the fourth week, no vials were burned and two samples taken. One as a control and the other was spiked with a known amount of toluene, xylene and 1,4-dioxane. On the following four weeks the same procedure was repeated for burns at 1850°F in the lower chamber and 2000°F in the upper chamber.

Ten GC analyses of each of the five samples for each temperature burns were done by BW Co. analytical laboratories, and two independent outside laboratories, each serving as a verification of the others' work.

The ash from the 1600°F burns contained residual toluene, xylene and 1,4-dioxane. The ash from the 1850°F burns contained no residual solvents.

This laborious process provided the data to prepare a statistical protocol showing that the initial assumptions of efficiency and long residency time with the correct temperature in the lower and upper chamber did in fact result in complete combustion, meeting the stringent 99.99% standard in the RCRA regulations.

RADIOACTIVE WASTE MANAGEMENT IN TAIWAN, ROC.

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Radwaste Administration, Atomic Energy Council
Republic of China

RADIOACTIVE WASTE MANAGEMENT POLICY

Taiwan's first step into the atomic age was an education reactor installed at Tsing Hua University in 1961. The Taiwan Research Reactor (TRR) of the Institute of Nuclear Energy Research (INER) was installed and started its operation in 1973. Meanwhile, Taiwan Power Company (TPC) decided to develop nuclear power to furnish a share of the future electricity supply. To this end, four BWR nuclear power units, at the northern tip of Taiwan with the total output of 3240 MW(e), are in operation. Two PWR units of total 1900 MW(e) at the southern tip will be connected to grid in May 1984 and May 1985 respectively.

Early in 1970, the Atomic Energy Council (AEC), Republic of China decided to invite experts from Taiwan Power Company, Universities and INER to organize a committee to study the waste management policy. One of their conclusions is "Ocean dumping is the most feasible alternative for the ultimate disposal of low level solid radioactive waste in this country. Before it is implemented, radioactive waste to be disposed of will be stored on some islets temporarily".

Since 1973, a joint task group by AEC, INER and TPC has undertaken a feasibility study on shallow land storage. Their preliminary siting criteria are:

(1) will not contaminate drinking water supply.
(2) is sparsely populated.
(3) is fairly easy for transportation.

A few mountain areas and many islets around Taiwan were investigated. Lan-Yu (Orchid Island) was finally choosen as temporary shallow land storage site for low level solidified radwaste.

REGULATORY ORGANIZATION

In order to protect public health and safety to this country, the Radwaste administration of AEC was instituted in January 1, 1981. The administration comprises three section and the Lan-Yu storage facility. Their authorities are:

(1) The development of radwaste management policy.
(2) The establishment and implementation of rules and regulations.
(3) A comprehensive inspection and enforcement program covering radwaste treatment, handling, transportation and storage to ensure compliance with the regulations.
(4) A confirmatory research program to develop the information required to support regulatory functions.
(5) The operation of National Lan-Yu Radwaste Storage Facility.

TRANSPORTATION AND STORAGE

Transportation

Pursuant to Radwaste Management Rules, 5,1, 1981, a radwaste generator should submit Transportation Plan (TP) and Emergency Response Plan (ERP) to Radwaste Administration for review and approval. Once these plans are approved, a shipment request form should be submitted for approval each time. Radwaste Administration will send out inspectors to verify that every phase of the transportation is being implemented safely and in compliance with regulatory requirements. Since May, 1982, vessels and tup barge transport have been used by contractors to ship radwaste package in containers from northern tip of Taiwan to Lan-Yu. The operation is illustrated in Figure 1 and 2.

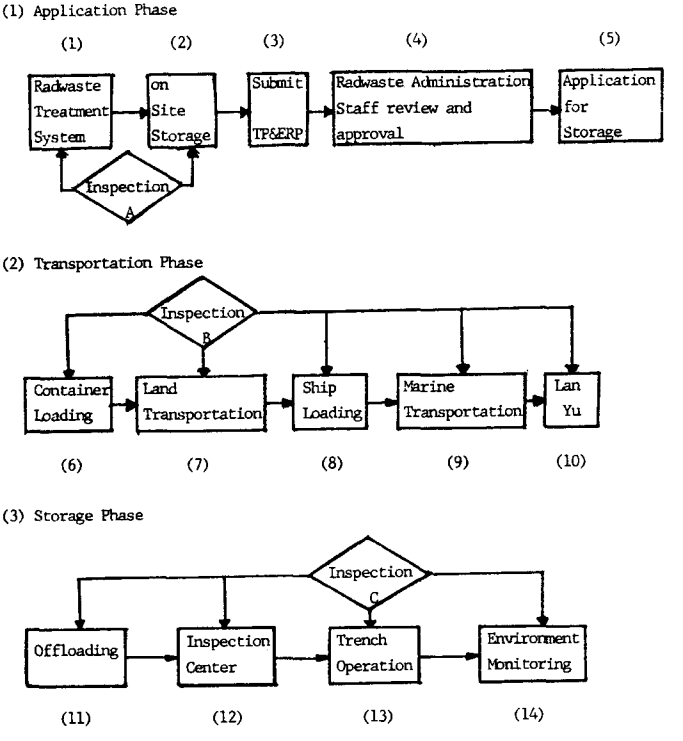

Fig. 1 Transportation Flow Diagram

Storage

Lan-Yu is located at 22°04'N and 121°32'E, and is about 75Km from Taiwan with a size of 46Km. Total of 170 hectares at the southern part of Lan-Yu is designed to accept solidified low level waste for over fifty years. The construction was completed in April 1982. 12,000 drums have been stored since May, 1982. The design of storage facility is shown in Fig. 3 and 4. This method can be classified as shallow land storage. Solidified radwaste encapsulated in steel drums will be stored in near-surface reinforced concrete trenches which are retrievable. These trenches are 5.4m wide, 4.5m deep and 0.35m thick, 3m underground and 1.5m above. Each trench is designed to store three layers of radwaste drums. After the trench is fulled, it will be covered with concrete slabs of 0.35m thick and sealed with asphalt for water proof. In this manner, concrete trenches will provide a high degree of isolation and prevent migration. The radiation level of a full trench without any backfilling is very close to background. Any water flowing out from the trenches will be collected in sumps equipped with a decontamination unit for emergency use. Before release to the ocean, samples will be analysed to ensure that the concentration is below one tenth of the maximum permissible concentration prescribed in the National Safety Standard on Ionizing Radiation Protection of AEC. 18 Thermo-Luminescence dosimeter stations were installed in the storage site and all around the islet. Samples of air, soil, plant and marine life have been taken on a regular basis for radioactity analysis.

RESEARCH AND DEVELOPMENT PROGRAMS

The main projects being carried on or to be set forth in the near future are:

(1) The ultimate disposal study of low level radwaste.
(2) Container Assessment-Corrosion study of low level waste container materials.
(3) The volume reduction study of low level waste.
(4) Environmental and Safety Assessment of National Lan-Yu Storage Facility.
(5) The feasibility study of safe storage of high level waste in Taiwan.

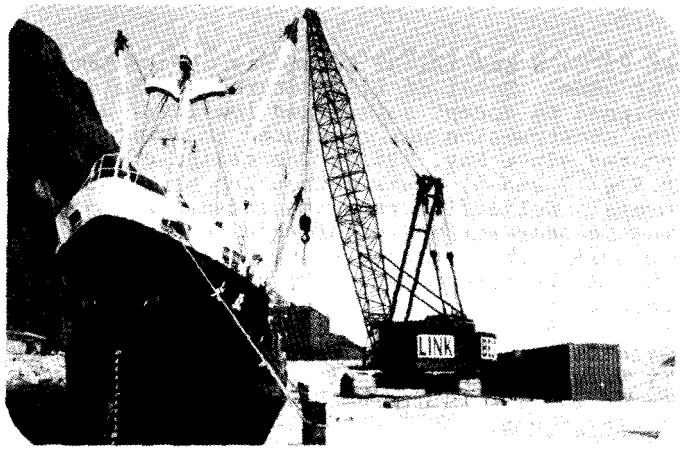

Fig. 2
Offloading at
Lan-Yu Pier

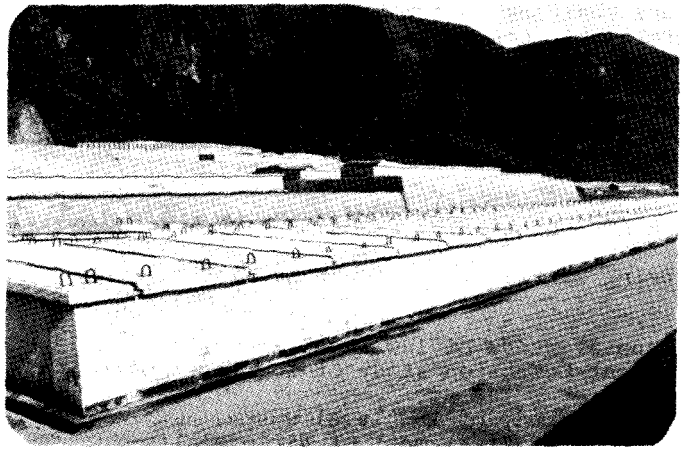

Fig. 3
Outside View of
Concrete Trench

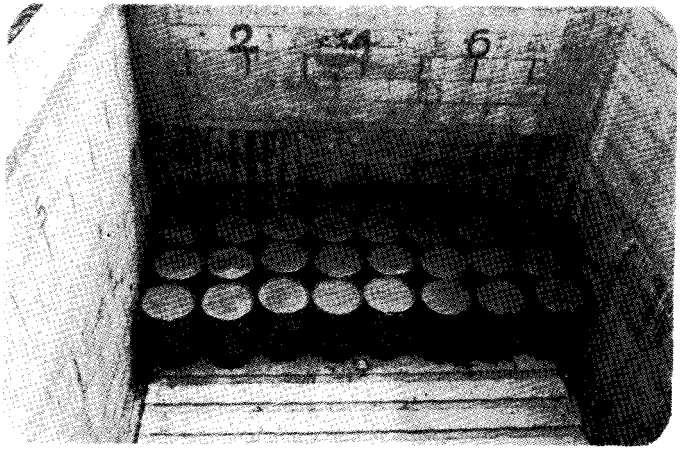

Fig. 4
Inside View of
Concrete Trench

ACTIVITIES IN RADIATION PROTECTION OF THE COMMISSION OF THE EUROPEAN COMMUNITIES.

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The EURATOM TREATY has conferred on the Commission certain competences and obligations to create the conditions necessary to reduce hazards arising from ionizing radiation to the life and health of the general public and of workers. On the one hand, the Commission's responsibilities evolve from a requirement for uniform safety standards, and Chapter III of the Treaty sets out a number of requirements involving regulatory and technical aspects of radiological protection and its implementation. On the other hand, the Commission is also made responsible for promoting research in Member States and for carrying out a research programme into the harmful effects of radiation on living organisms (Annex 1 of the Treaty). This double obligation in radiation protection forms part of the objectives related to health and safety which the Commission pursues in the Community.

I. REGULATORY AND TECHNICAL ACTIONS IN RADIATION PROTECTION.

Articles 30 to 32 of the Euratom Treaty call on the Commission to set up "Basic Safety Standards for the Health Protection of the General Public and Workers against the Dangers of Ionizing Radiation". Additional standards relating to the protection of persons undergoing medical examinations or treatment have been drafted. All of these standards have been formulated after consulting a group of experts set up for this purpose and aim to establish uniformity in the Community in the field of radiation protection.

The Member States are obliged to lay down appropriate legislative, regulative or administrative provisions to comply with the Commission's Basic Safety Standards (Article 33). With the evolution of knowledge on radiation protection, these standards have been subject to several revisions since first issued in 1959; the most recent revision dates from 1980 and brings them into line with the concepts and methodologies set out in ICRP 26.

Guidance on emergency reference levels concerning the accidental exposure of the population is contained in a report prepared by the aforementioned group of experts and published by the Commission in 1982. A further guide dealing with the protection of the public in respect of consumer goods containing radioactive substances awaits publication.

In support of the basic safety standards and in view of their correct application, the Commission has initiated reports on such topics as:

- Dosimetry of external exposure.
- Radon hazards in mines.
- Assessment of internal plutonium contamination.
- Training in radiation protection.
- Analysis of national regulatory provisions.

The continuous monitoring of radioactivity in water, soil and air in each Member State and the communication of the results to the Commission, foreseen by Articles 35 and 36, is to ensure also compliance with the Basic Safety Standards. The Commission publishes an annual compilation of these data.

Under the term of Article 37, the Commission is required to issue an opinion on plans involving the disposal of radioactive waste in any Member State as such procedures could result in contamination in neighbouring Member States. Since to date almost all solid waste has been stored rather than disposed of, special attention is paid to liquid and gaseous effluents from nuclear installations, both in normal and accident conditions. Effluent treatment, dispersion and food chain transfer are considered in this respect, and, where appropriate in collaboration with the research programme presented below, the Commission has initiated studies and/or seminars on

- Charcoal efficiency for radio-iodine capture,
- Aerosol filtration,
- Noble gas retention,
- Mesoscale atmospheric dispersion,
- Cumulative effects of discharges into river basins,
- Environmental transfer between soil, plant and animal,
- Estimation of population dose from hypothetical accidental releases in the E.C.,
- Transfrontier emergency planning.

In addition, the Commission periodically reviews discharges from nuclear power stations and reprocessing plants in the Community and evaluates the resultant doses to the population.

Finally, in order to obtain a more complete view of the range of doses to which the population may be exposed, studies of exposure from natural background have been undertaken.

II. RESEARCH IN RADIATION PROTECTION.

The Radiation Protection Research Programme aims to update the scientific basis for radiation protection standards and to develop methods and techniques to prevent and counteract harmful effects of radiation, and/or to cope with the consequences of radiation accidents. Problems related to the protection of man and his environment against ionizing radiation are investigated with a view to preventative action. Radiation risks are assessed in perspective to other risks arising in modern society, and pertinent and up-to-date information is provided for decision making.

The scientific priorities to which the Radiation Protection Programme adresses itself are conveniently classified in six closely linked sectors:

- Radiation dosimetry and its interpretation,
- Behaviour and control of radionuclides in the environment,
- Non-stochastic effects of ionizing radiation,
- Radiation carcinogenesis,
- Genetic effects of ionizing radiation,
- Evaluation of radiation risks and optimization of protection.

Particular consideration is currently being given to several areas which span these sectors:

NATURAL RADIATION, in particular inhalation of radon and thoron and their decay products, represents the principal contribution to population exposure. Natural radioactivity and its artificial enhancement are assessed by measurements on site and in construction materials.

DIAGNOSTIC RADIOLOGY is the largest contribution to all man-made population exposure and must be optimized to obtain relevant clinical information at the smallest possible exposure level. This is achieved by determining doses and by optimizing equipment, procedures and interpretation.

RADIOACTIVE POLLUTION from all categories of sources is followed along its pathways to man, models are provided to describe and predict this behaviour, and measures are developed to gain control in case of accidental releases. Particular emphasis is given to the long-term behaviour of radionuclides and to the radiological aspects of disposal of radioactive wastes.

EFFECTS OF LOW DOSES, not yet directly amenable to an experimental approach, are attacked by a multidimensional strategy. Theoretical considerations based on microdosimetry, determination of molecular alterations in target molecules and studies on isolated cell systems are combined with the extrapolation of data obtained at higher dose levels from animal experiments or from epidemiological studies of human populations.

CARCINOGENESIS as the principal somatic hazard of radiation to the population is investigated with respect to its mechanisms of action and its actual risk to man. Studies on molecular and cellular mechanisms and experiments in animals provide a basis for extrapolations to man, which must be confirmed by observations on suitable human populations. Carcinogenesis from radionuclides incorporated in the body is studied with respect to the deposition of radionuclides in target tissues and their metabolism.

GENETIC DAMAGE affecting future generations needs to be investigated for its mechanisms and risks to man. Repair mechanisms are crucial factors in the expression of genetic damage. Investigations are designed to determine natural incidence and contribution from

radiation. Particular attention is devoted to the effects on the developing organism.

ACCIDENTS, although rare, must be anticipated by developing strategies for analysis, prediction, and countermeasures. Means for diagnosis and treatment following whole-body, local or internal irradiation are investigated.

RISK ASSESSMENT, based on both benefits and harm from ionizing radiation and its use, is the integrating goal of research in radiation protection. Individual and collective doses are assessed, criteria for optimization of radiation protection are developed, the consequences of accidents are analyzed and the economic and social implications of the use of ionizing radiation are evaluated.

The Commission of the European Communities carries out this research in about 300 cost-shared contracts with national research institutions or universities and with several international scientific groups, contributing 25-40% to their costs. Promotion of cooperation and dissemination of scientific information in radiation protection is achieved by regular symposia, seminars and workshops and by informal study group meetings between contractors. Reviews and reports on topical subjects are also published.

WORLD-WIDE RADIATION PROTECTION STANDARDIZATION BY ISO

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and

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AFNOR Paris/France

After a meeting in London in 1946, delegates from 26 countries followed an initiative of the newly formed United Nations to create a new international organization "whose object shall be to facilitate the international co-ordination and unification of industrial standards". The new organization was named the International Organization for Standardization (ISO). The already existing International Electrotechnical Commission (IEC), which covers electrotechnical matters, was affiliated to ISO and, while preserving its autonomy, has served since then de facto as the Electrical Division of ISO. ISO is now universally recognized as the specialized international agency for standardization, currently comprising the national standards bodies of 86 countries, thus covering over 95% of the world's industry. The work of ISO is aimed at world-wide agreement on international standards with a view to the expansion of trade, the improvement of quality, the increase of productivity and the lowering of costs. ISO's work covers every area of technology. It is decentralized, the Technical Secretariats being located in 29 countries and served by some 400 people, while the ISO Central Secretariat in Geneva has a staff of about 100 persons drawn from 16 countries. At present more than 5000 International Standards are registered at the ISO Central Secretariat. Several regional organizations are active within ISO, for example ASMO for the Arabic countries, ARSO for Africa, CEN for Western Europe, and COPANT for Latin America.

The decision to set up a technical committee is taken by the ISO Council, which also determines the scope of the committee. A technical committee usually creates sub-committees (SC) and working groups (WG) to cover different aspects of the work. More than 100,000 voluntary experts from all over the world are currently engaged in ISO activities. Either international standards without a national equivalent are developed by the experts from different countries, or existing national standards are harmonized into international ones. It is expected that approved international standards increasingly will become or supersede corresponding national standards in all ISO member countries.

Within ISO, Technical Committee 85 "Nuclear Energy" is responsible for world-wide standardization in the nuclear field. The decision to form TC 85 was made by ISO in 1956, and a first plenary session took place in 1957 with delegates representing 17 countries. The secretariat was with the American National Standards Institute, ANSI, for many years. The DIN German Nuclear Standards Committee has served as secretariat for TC 85 as well as its SC 5 "Nuclear Fuel Technology" and several WGs since 1978.

Among the 175 Technical Committees of ISO, TC 85 ranks with regard to the projects in progress and the number of meetings held per year among the top ten to twenty per cent. In 1982, for example, it had fourteen WG and SC meetings dealing with 57 projects. The number of participating (currently 21) and observing (currently 22) member countries is increasing, with all the important nuclear countries represented (Table 1). Similar to the situation in other TCs, the active participation of most developing and socialist countries is, however, marginal. Seven international organizations have liaison arrangements with ISO/TC 85.

Table 1
Participating (P) and Observing (O) Member Countries of ISO/TC 85 "Nuclear Energy"

P-Member	O-Member	Liaison
Belgium	Australia	IAEA
Brazil	Austria	CCC
Canada	Bulgaria	CEC
China, People's Rep.	Chile	CERN
Czechoslovakia	Colombia	EOQC
Finland	Denmark	ICRP
France	Egypt	WHO
Germany, Federal Rep.	Ghana	
Hungary	Greece	
Italy	India	
Japan	Iran	
Netherlands	Ireland	
Poland	Israel	
Romania	Korea, Dem. Peop. Rep.	
Spain	Korea, Rep.	
Sweden	Mexico	
Switzerland	New Zealand	
Turkey	Pakistan	
United Kingdom	Portugal	
U.S.A.	South Africa	
U.S.S.R	Venezuela	
	Yugoslavia	

TC 85 is divided into the following sub-committees:
SC 1 - Terminology, Definitions, Units and Symbols, (secretariat ANSI, USA), with one working group (WG);
SC 2 - Radiation Protection (secretariat AFNOR, France);
SC 3 - Power Reactor Technology (secretariat SIS, Sweden), with four active WGs;
SC 4 - Radioactive Sources (secretariat Poland) has been dissolved because its tasks have been accomplished; and
SC 5 - Nuclear Fuel Technology (secretariat DIN, Germany), with nine active WGs.

Table 2. Working groups of ISO/TC 85/SC 2 "Radiation Protection"

WG	
2	Reference Radiations
4	Apparatus of Gamma Radiography and Irradiators
5	Materials and Devices for Protection against X, Gamma, Beta and Neutron Radiations, and Equipment for Remote Manipulation of Radioactive Materials
6	Protective Clothing against Radioactive Contamination
7	Thermoluminescence Dosemeters
8	Radioelement Gauges
9	Evaluation of Contamination
10	Ease of Decontamination of Surfaces
11	Leak Testing of Sealed Sources

The work of TC 85/SC 2 is relevant to the field of *radiation protection*. Its scope is to take care of standardization in the field of protection against ionizing radiation and radioactive substances mainly in the areas of procedures and equipment for protection against external irradiation and radioactive contamination, and methods for evaluation of external radiation and radioactive contamination. It has nine active WGs (Table 2), which have already produced nine standards (Table 3), and progress on various projects (Table 3) has been mainly smooth. In the following, a brief description is given of the current activities in the working groups.

The active WG 2 *Reference Radiations* is headed by Mr. G. Portal, France (secretariat AFNOR). It consists of individual experts from Belgium, Canada, France, Germany, Italy, Netherlands, Poland, Romania, Sweden, UK and USA, with liaison representatives of CEC, ICRP, ICRU and CERN. WG 2 has four active sub-groups dealing with the preparation of new standards on

- dosimetry of X and γ reference radiations for radiation protection over an energy range 8 MeV to 1.3 MeV,

- methods of production of neutron reference radiations for calibrating neutron measuring devices used for radiation protection purposes and for determining their response as a function of neutron energy,

- procedures of calibration of neutron measuring instruments using the radiations produced by the methods described in the previous document,

- method of production of α and β reference radiations for calibration of surface contamination measuring instruments,

- method of production of γ reference radiations for calibration of surface contamination measuring instruments, and

- γ reference radiations for determining the response of protection level dosemeters and doseratemeters at photon energies of 6 and 8.5 MeV (this will become a further addendum to ISO 4037).

It has to be emphasized that experiments are undertaken in various laboratories to ensure that the specification mentioned in the drafts are realistic.

The secretariat for WG 4 *Apparatus for gamma radiography and irradiators* is now also with AFNOR. WG 4 consists of experts from Austria, Belgium, Brazil, Canada, Poland, USA and USSR. Its scope is to supplement ISO 3999 by requirements on source ejection devices and to prepare a draft on the requirements to be fulfilled by irradiators. A meeting of this working group should be held at the beginning of 1984.

The convenor of WG 5 *Materials and devices for protection against X, gamma, beta and neutron radiations and equipment for remote manipulation of radioactive materials* is Mr. Vertut, France, and the secretariat lies with AFNOR. The WG consists of experts from Austria, Belgium, Germany, Poland, France, Italy and the UK. Further to ISO/DIS 7212 on shielding units of 50 mm and 100 mm under ballot, WG 5 is now working on containment and transfer elements, and on shielding of 150 mm and 200 mm thickness, tongs, small manipulators and gloves.

WG 6 *Protective clothing against radioactive contamination* is headed by Mr. Seran, and the secretariat is AFNOR. It consists of experts from Austria, Belgium, France, Germany, Italy, Sweden, UK and USA. A first draft proposal ISO/DP 8194 describing in particular test methods which enable the measurement of the main characteristics of such clothing to ensure adequate protection against radioactive contamination has been prepared and circulated to SC 2 members for comments. The numerous comments received are at present under consideration and a revised draft proposal will be prepared accordingly.

Table 3. Some ISO radiation protection standards and drafts

Reference	Title
ISO 1757-1980	Personal photographic dosemeters
ISO 1758-1976	Direct-reading electroscope-type pocket exposure meters
ISO 1759-1976	Indirect-reading capacitor-type pocket exposure meters and accessory electrometers
ISO 2889-1975	General principles for sampling airborne radioactive materials
ISO 3999-1977	Apparatus for gamma radiography - Specification
ISO 4037-1979	X and gamma reference radiations for calibrating dosemeters and doseratemeters and for determining their response as a function of photon energy
ISO 4037/AD 1-1983	X and gamma reference radiations for calibrating dosemeters and doseratemeters and for determining their response as a function of photon energy - AD 1: High dose rate series of filtered X-radiations
ISO 4037/AM 1-1983	AM 1: Low dose rate series of filtered X-radiations
ISO 4071-1978	Exposure meters and dosemeters - General methods for testing
Drafts:	
ISO/DIS 6980	Reference beta radiations for calibrating dosemeters and doseratemeters and for determining their response as a function of beta radiation energy
ISO/DIS 7212	Lead shielding units for 50 mm and 100 mm thickness walls
Draft Proposals:	
ISO/DP 7205	Gauges incorporating radioelements - Gauges intended to be installed at a permanent point
ISO/DP 7503	Evaluation of surface contamination
ISO/DP 8034	Personal and environmental thermoluminescence dosemeters
ISO/DP 8065	Evaluation of tritium - Surface contamination
ISO/DP 8194	Clothing for protection against radioactive contamination

WG 7 *Thermoluminescence Dosemeters* is chaired by Mr. E. P. Goldfinch (UK). Experts from Belgium, Finland, France, Germany, Italy, Sweden, the Netherlands, United Kingdom and the USA are active in this working group. It has worked in very close liaison with IEC/SC 45 B/WG 6 which was preparing an IEC publication on the corresponding readout instruments. ISO/TC 85/SC 2/WG 7 had prepared a tenth working document which was submitted as ISO/DP 8034 to the comments of SC 2 members and it seemed that a few remaining problems could be solved rapidly for the preparation of an ISO/DIS. However, during a meeting in September 1982, the IEC group decided that it was not feasible to write separate standards on readers and detectors and that it was necessary to deal in one single document

with the complete system. After a meeting held in London in March 1983, it was agreed that three complementary documents could be prepared, one part dealing with the complete system, and for those items where it was possible to consider the reader or the detector without the other, and one part on each component separately. ISO/TC 85/SC 2/WG 7 is still working on that basis but it seems that the views are split among the experts. ISO/TC 85/SC 2, in connection with IEC, intends to decide on this subject during its next meeting to be held at the beginning of 1984.

In WG 8 *Radioelement Gauges*, Mr. Chatelet is chairman and AFNOR provides the secretariat. Individual experts from Belgium, Finland, France, Germany, Italy, USA and USSR participate in the work. A second draft proposal ISO/DP 7205 has been circulated but in view of the number of comments still received, this document needs further revision and another meeting of WG 8 will be organized.

WG 9 *Evaluation of Contamination*, chaired by Mr. T. Tamberg (secretariat DIN, Germany) consists of experts from Belgium, Canada, Finland, France, Italy, Sweden, Czechoslovakia and the UK. As mentioned above, a first draft proposal (ISO/DP 7503) has been prepared on the measurement of surface contamination by β emitters of energies above 0.4 MeV and by α emitters. It is being finalized to take account of the comments received with the vote of SC 2 members, and in particular of those coming from the convenor of the sub-group within ISO/TC 85/SC 2/WG 2 dealing with the production of reference radiations for calibration of surface contamination monitors with which a close liaison is ensured. Another draft proposal, ISO/DP 8065 on the evaluation of tritium surface contamination, has been circulated within SC 2 and comments received are now under consideration by WG 9. Further work on the contamination by β emitters of low energy (below 0.4 MeV), electron capture and internal conversion emitters requires more time and additional studies.

The work of WG 10 *Ease of Decontamination* (convenor also Mr. T. Tamberg, secretariat DIN) is closely related to that of WG 9. Experts from Austria, Belgium, Brazil, Canada, France, Germany, Italy, Sweden, Switzerland, UK and USA contribute to the work. A BSI and DIN standard serve as the main basis for its work. First, however, interlaboratory tests are carried out to determine which is the best method to be standardized on the international level for determining the ease of decontamination of surface materials.

WG 11 *Leak Testing of Sealed Sources*, with AFNOR acting as secretariat, is a new working group set up by SC 2 during its meeting held in Ghent in May 1982 which has been given the task of continuing the work previously carried out by ISO/TC 85/SC 4/WG 5 (now disbanded). For this purpose WG 11 may have to review also ISO/TR 4826 *Sealed Radioactive Sources - General*, which includes a few statements on leak testing. Experts from the following countries have already expressed their wish to participate in the work: Austria, Belgium, Canada, Finland, France, Germany, Sweden, UK, USA, USSR,and a first meeting of WG 11 will be organized at the beginning of 1984.

An enquiry is still in progress to determine whether, in view of the documents already published by other international organizations on standard models for calculation of environmental exposures from gaseous effluents, standardization work in this field should be carried out by SC 2. It should also be noted that

- this review reflects the status of October 1983 and does consequently not consider the results of a January 1984 meeting of ISO/TC 85/SC 2, a February 1984 meeting of IEC/TC 45 B, etc.

- all published ISO standards and draft standards may be purchased through the national standards institutes.

INTERNATIONAL STANDARDIZATION OF TECHNICAL PRODUCTS IN MEDICAL PRACTICE

Hermann Bertheau
Secretary IEC TC 62
Hamburg

International Standardization of Technical products is mainly performed by the two non-governmental international organizations:

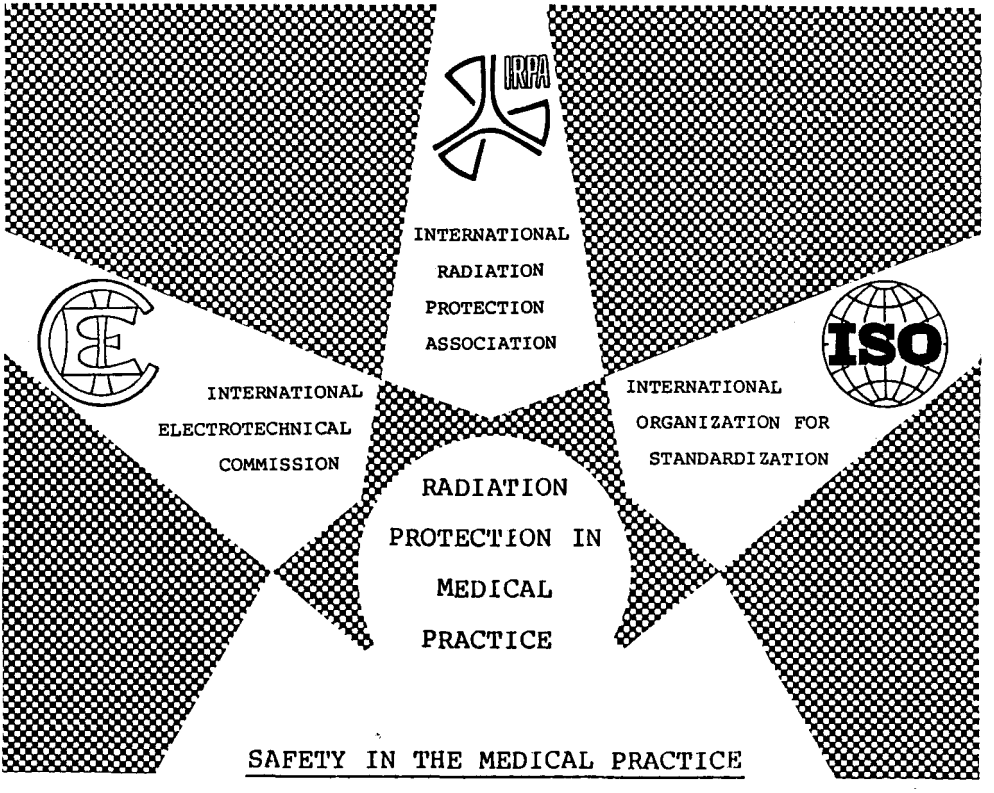
the International Electrotechnical Commission

and the International Organization for Standardization

Both organizations are active in a wide variety of subjects, whereby the activities of the IEC are confined to subjects related to the use of electricity and to phenomena arising from the presence of electricity, such as electrostatic charges.

Both organizations are concerned with aspects of protection against various kinds of radiation, whereby either radioactive materials are involved or the radiation is deliberately produced or the technical products concerned have to be protected against a radiation.

One of the important areas in which both organizations are involved is the safety in the medical practice or more precise the safety of patients and operators due to the use of technical products in the medical environment.



Technical Committees concerned with radiation are for example:

IEC TC 3 Graphical symbols
IEC TC 13 Electrical measuring equipment
Transducers with electrical output
IEC TC 27 Industrial electroheating equipment
IEC TC 29 Electroacoustics
IEC TC 45 Nuclear instrumentation
IEC TC 59 Performance of household electrical appliances
Microwave appliances
IEC TC 61 Safety of household and similar appliances
Household microwave ovens; Ultraviolet and infrared lamps
IEC TC 76 Laser equipment
ISO/TC 85 Nuclear energy
ISO/TC 135 Non-destructive testing

Within the IEC following Technical Committees are dealing with equipment, devices, accessories or system aspects related to the use or presence of electricity in the medical practice:

IEC TC 1 Terminology
Radiology and radiological physics; Electrobiology
IEC SC 14D Small special power transformers
Safety transformers
IEC SC 29D Ultrasonics
Hearing aids; Ultrasonic diagnostic equipment; Ultrasonic therapeutic equipment
IEC SC 45B Health physics instrumentation
IEC TC 61 Safety of household electrical appliances
Ultraviolet and infrared radiating appliances
IEC TC 62 Electrical equipment in medical practice
IEC TC 64 Electrical installations of buildings
Installations in medically used rooms
IEC TC 66 Electronic measuring equipment
Clinical laboratory equipment
IEC TC 76 Laser equipment
IEC TC 77 Electromagnetic compatibility between electrical equipment including networks
IEC CISPR International special committee on radio interference

Following ISO Technical Committees are concerned with equipment, devices, accessories, measuring procedures for medical use and bioengineering including surgical, radiological and dental applications, that are related to electricity or radiation:

ISO/TC 76 Transfusion equipment for medical use
ISO/TC 84 Syringes for medical use and neeles for injections
ISO/TC 85 Nuclear energy
Radiation protection, personal dosemeters
ISO/TC 94 Personal Safety
Protective clothing and equipment
ISO/TC 97 Computer and information processing
ISO/TC 106 Dentistry
ISO/TC 121 Anaesthetic equipment and medical breathing machines
ISO/TC 136 Furniture
Hospital furniture
ISO/TC 150 Surgical implants
ISO/TC 172 Optics and optical instruments
ISO/TC 173 Technical aids for disabled and handicapped persons

For their activities in the medical field the two organizations have made an agreement in the form of 'Guidelines for the collaboration between ISO and IEC in the field of safety aspects of medical electrical equipment'

Liaisons exist also with other non-governmental and governmental organizations working in fields of common interest.

Among those ICRP and ICRU should be mentioned and among the governmental organizations the International Organization for Legal Metrology, OIML. The OIML is concerned with the procedures for legally enforced calibration of instrumentation used in the health care.

At present, work in the IEC involving aspects of radiation protection in medical electrical equipment is done for:

Medical electron accelerators,	Publication 601-1-1 (1980)
High frequency surgical equipment,	Publication 601-1-2 (1982)
Short wave therapy equipment,	Publication 601-1-3 (1982)
Ultrasonic therapy equipment,	Publication 601-2-5 (1984)
Microwave therapy equipment,	Publication 601-2-..(1984), (earlier 62D(Central Office)18)
Baby incubators,	Document 62D(Central Office)24, March 1983
Gamma beam teletherapy equipment,	Document 62C(Central Office)26, January 1984
Therapeutic X-ray generators,	Document 62B(Central Office)49, February 1984
High voltage generators of diagnostic X-ray generators,	Documents 62B(Central Office)50 and .., June 1982 and ... 1984
Capacitor discharge X-ray generators,	Document 62B(Secretariat)80, ... 1984 (earlier: 62B(Secretariat)70)
Dignostic X-ray equipment, General requirements for protection against ionizing radiation,	Document 62B(Secretariat)91, ... 1984 (earlier: 62B(Secretariat)81)
Ultrasonic medical diagnostic equipment,	Document 62D(Central Office).., ... 1984 (earlier: 62D(Secretariat)31)

Information on current activities in IEC TC 62 can be obtained by requesting the submission of the Occasional Report from the Secretariat of IEC TC 62; POB 630121; D - 2000 Hamburg 63 .

Title and Scope of Technical Committee No 62 of the IEC are:

Electrical equipment in medical practice

To prepare international standards concerning the manufacture, installation and application of electrical equipment used in medical practice. This also concerns surgery, dentistry and other specialities of the healing art.

Note.- This scope includes systems, equipments and accessories which are within the scope of other Technical Committees; attention will be confined to aspects in which special requirements for medical use arise, particularly as regards safety.

The IEC works through the National Committees of its member countries. At meetings of the Technical Committees the National Committees are represented by delegations. At meetings no final decisions are made as to the content of a standard to be published. At meetings the decisions of the Technical Committee

are confined to questions of procedure, how and when the drafts will be circulated for comments or balloting.

International safety standards are developed according to the IEC Report, Basic aspects of the safety philosophy of electrical equipment used in medical practice, Publication 513 (1976).

In order to aim at generally useful and acceptable international standards it is important that all interested people or groupes within a country are involved in commenting on, and the decision making about, the draft international standard developed. It is evident that the influence a National Committee will have upon the development of an international standard depends on the prospect of the final international standard being accepted and used in the country.

The desired contribution at the national level can normally only be reached if the structures of the national participation ensure the input from the interested groups and parties concerned in that country. In this connection the need for liaisons at the national level to related activities in the different international organizations and different Technical Committees of IEC and ISO is evident.

In order to care for good international standards on technical products in the medical practice, the national member organizations of the International Radiation Protection Association should be adequately represented with respect to the necessary requirements for the desired radiation protection in the relevant committees of the national bodies being the members of IEC and ISO.

The addresses of the National Committees of the IEC as well as those of sources of supply of Publications of the IEC can be obtained from the Information Officer; IEC Central Office; 3, rue de Varembé; CH - 1211 Genève 20 .

ACTIVITIES OF THE OECD NUCLEAR ENERGY AGENCY IN THE FIELD OF RADIOLOGICAL PROTECTION

O. Ilari
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Paris

Introduction

The Nuclear Energy Agency (NEA) of the OECD was initially created as a European organisation (ENEA) in 1957. Japan joined the NEA in 1972, successively followed by Australia, Canada and the United States. The main aim of the NEA is to assist Member countries in the peaceful and safe development of nuclear power. The work of the Agency is carried out under the OECD Council by the Steering Committee for Nuclear Energy, which is assisted by a number of specialised committees including the Committee on Radiation Protection and Public Health (CRPPH) and the Radioactive Waste Management Committee (RWMC).

The CRPPH is responsible for the Agency's activities in radiological protection and related environmental problems. Its work includes the review and discussion of national radiation protection policies and practices, the review of progress of radiation protection philosophy, the interpretation of the ICRP Recommendations, and the study of the ways of their conversion into practical application in Member countries. The Committee also prepares technical studies and state-of-the-art reports on specific topics, as a reference material for national authorities and the radiation protection community.

The current priorities of work of the Agency are focused primarily on the nuclear fuel cycle, nuclear safety and radioactive waste management, including their major radiation protection implications, covering both technical and policy issues within those areas. In this context, the role played by the CRPPH is to define and assess relevant radiation protection and public health issues and to provide technical and policy guidance leading towards their resolution. At the same time, the Committee continues actively to pursue its traditional role as a permanent forum where national radiation protection and public health authorities exchange experience and views and co-ordinate their practices.

Three main areas of work can presently be identified in the Agency's radiation protection programme:

- Interpretation of the ICRP recommendations, and review of problems of application of these recommendations to practical situations and their transfer into national regulations.

- Radiation protection aspects of radioactive waste management.

- Protection of workers in the nuclear industry.

The Secretariat's activities for this work are carried out by the Agency's Division on Radiation Protection and Waste Management, with the assistance of international Expert Groups and technical consultants.

Radiation Protection Recommendations

Following adoption of new basic radiation protection norms in 1981 by NEA, jointly with the International Atomic Energy Agency (IAEA), the World Health Organization (WHO) and the International Labour Organization (ILO), the Agency organised through the CRPPH an international review of the difficulties likely to be encountered by national authorities in interpreting and applying the ICRP recommendations from which the IAEA/WHO/ILO/NEA Revised Basic Safety Standards are derived.

A number of clarifications were sought from the ICRP on the application of the principle of optimisation of protection, and two statements were made public drawing the attention of national regulatory authorities to two specific issues: the application of the ICRP principle of justification of practices and the introduction of a lifetime dose limit for radiation workers. The purpose of these statements was to try to avoid the introduction of discrepancies in national radiation protection regulations, which so far have shown a remarkable degree of consistency.

Other interesting achievements in this field are the recent preparation and issuing of a series of reports aimed at throwing light on controversial matters, such as, for example, the meaning of the average doses in groups of workers and their relationship with the ICRP dose limits, and the different meanings and uses, as well as limitations, of the concept of collective dose.

Radiation Protection in Radioactive Waste Management

Because of the very long times for which some wastes will remain radioactive, and the inevitable uncertainties associated with safety assessments projected far into the future, attention has been given to seeking a common understanding of how radiation protection objectives based on ICRP recommendations could be developed for application to waste disposal, particularly to its long-term aspects. This work, initiated in 1982, was completed in 1984 with emphasis on two main areas: the definition of objectives for the upper limit of radiation risk for individual members of the public, and the clarification of the role and limitations of collective dose assessments. The latter is particularly relevant to the development of methodologies to be applied to ensure that the overall detriment from waste disposal is as low as reasonably achievable.

A substantial effort was also made during the last few years in the examination of long-term aspects of uranium mill tailings management. The aim of this study was primarily to review the long-term radiological protection problems, and to illustrate the application of the ICRP principle of optimisation in the decision-making process concerning the selection of tailings management options.

Another field where the Agency plays a major role is the sea disposal of low-level radioactive waste. In this respect, the Agency ensures the operation of an international consultation mechanism which specifies general and radiological safety conditions for the sea dumping operations carried out by some Member countries under the London Dumping Convention. Under this

mechanism, the NEA is also responsible for an international surveillance over the execution of these operations, and co-ordinates the international assessment of the suitability of the dumping sites from the radiological protection viewpoint. This co-ordination also covers the associated international programme of environmental monitoring and research which is carried out by a concerted effort of a number of Member countries.

Protection of Workers in the Nuclear Industry

In the last few years the concern of experts and regulatory authorities has been progressively focused on the existence of some groups of workers in the nuclear industry exposed to levels of radiation risk which are higher than the general average risk of workers in that industry. These are the uranium miners and some groups of workers such as maintenance, inspection and health physics workers in nuclear facilities, particularly nuclear power plants.

Recognising the importance of these issues, the NEA has been carrying out for several years a vigorous programme in this field. In the area of uranium mining, the Agency has given in the last few years, and is still giving, a significant contribution through the promotion of studies and exchanges of information and the preparation of technical guidance. Expert Groups of the Agency have prepared technical reports on radon, thoron and their daughters dosimetry, metrology and monitoring. A correct dosimetry and monitoring can only be obtained from accurate and reliable instrumentation. To improve practices in this area the NEA, in co-operation with the CEC, is carrying out an important inter-calibration and intercomparison programme for dosimetry and monitoring equipment covering the whole OECD countries area.

As far as the increasing exposure of workers in nuclear plants is concerned, the need for a broadening of the data base available on this issue, as a foundation for improvements in plant design and operation, was felt by the NEA since 1977, and an international enquiry on the levels and trends of occupational exposure in nuclear facilities was carried out between 1978 and 1980. Although the results of this study were judged interesting and useful, the difficulties associated with the collection, analysis and comparison of large quantities of dosimetric data at the international level were recognised by the CRPPH. The Committee, therefore, decided to focus its attention on a more specific issue, namely the problem of the implication of nuclear safety-motivated decisions on the exposure of workers. Nuclear safety-related actions, aimed at reducing the probability and/or consequences of nuclear accidents, imply in fact in many cases an additional exposure of the workers for inspections, special maintenance, etc.

There is a growing feeling among experts that this situation may present an imbalance between the requirements of safety of plants, protection of the public and protection of the workers, and that there is, therefore, a need for a rational approach to find the optimum balance between these different requirements. The NEA is contributing to the solution of this difficult problem through the activities of a Group of Experts entrusted with the

task of reviewing available dosimetric data on occupational exposure, identifying those aspects which involve problems of balancing between plant safety and radiation protection requirements, and developing a conceptual approach for the determination of an optimum balance. The subject of this work involves delicate issues, such as the application of ICRP principles and approaches to probabilistic events as in nuclear accidents, and difficult conceptual decisions, such as the relative weight to be attributed to the protection of the workers and the reduction of the potential risks to the public.

The ICRP recommendations of 1977 have introduced a number of new issues in the approach and methods for the assessment and recording of doses to workers. In practice, radiation protection operators found it difficult to adapt fully to the approach recommended by the ICRP, partly because of problems of interpretation and partly due to technical limitations, such as the sensitivity of monitoring methods for the intake of some long-lived radionuclides. An attempt to reach an international consensus on these issues through the work of an NEA Group of Experts has recently achieved a successful outcome, and a set of practical recommendations addressed to operational health physicists and dosimetric services are being published by the Agency.

Other Activities

Although the main thrust of the Agency's radiation protection work is focused on the areas described above, the CRPPH continues to keep under review more general matters in order to contribute to put the radiological impact of nuclear energy in the proper perspective with other sources of human exposure.

In this context, the studies on radon, thoron and their daughters dosimetry, metrology and monitoring have been extended to include those aspects which are more relevant to the exposure of the public, notably in dwellings. Also, a working group of the Agency is preparing a guide on the radiation protection aspects of the commerce, use and regulation of consumer products containing radioactive materials, which incorporates the most recent developments of the ICRP doctrine and will replace the NEA Guidelines on Consumer Goods, which were published in 1970.

A more exhaustive description of the Agency's organisation and activities as well as a list of the NEA publications relevant to radiation protection are given in the booklet "International Co-operation in Radiological Protection, OECD Nuclear Energy Agency, 1984"* issued by the Agency on the occasion of the IRPA 1984 Congress.

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*Provisional title.

RADIATION PROTECTION ACTIVITIES IN AUSTRALIA
The Australian Radiation Protection Society

In Australia there is a wide range of industries, hospitals, education centres and research establishments, many of which use sources of ionising and non-ionising radiations in their various operations. The radiation protection control of these operations is vested in a two tier governmental regulatory system.

The following sections outline work being undertaken in Australia which involves the provision of radiation protection services.

The Australian Atomic Energy Commission (AAEC)

The AAEC is a Commonwealth Statutory Authority. Its Lucas Heights Research Laboratories (LHRL) are located on the outskirts of Sydney.

At the LHRL the Health and Safety Division is responsible for providing occupational health services. The Division's specialised sections deal with occupational medicine, health research, and hygiene, industrial safety and radiation protection.

A comprehensive range of radiological protection services is provided by a staff of about 30 Health Physicists and Technicians. These services include provision of advice on radiological protection, workplace surveillance and assessment of doses to personnel.

The techniques used include measurement of external radiation dose rates, airborne radioactivity and surface contamination; photographic and thermoluminescent dosimetry; urinalysis and whole body monitoring.

The AAEC also employs Health Physicists in its Regulatory Bureau, and in its Environmental Science Division; and provides a number of services related to radiological protection, including the provision of expert advice to radiation users in industry, medicine, and government; the maintenance of the Australian standard of activity; and the monitoring of the visits of nuclear powered warships.

The Australian Radiation Laboratory (ARL)

In 1929 the Commonwealth Department of Health established a small radium laboratory. That laboratory evolved into the present ARL. Throughout this period, its responsibilities have been revised and broadened so that today it serves as a national laboratory covering the general field of radiation health.

ARL provides the resources and expertise necessary for research, development and advice relating to the occupational and public health implications of exposure to ionising and non-ionising radiation including:

- the development, in conjunction with other organisations, of national standards, codes of practice and guidelines covering all aspects of the health implications of ionising and non-ionising radiations;
- the provision of a nationally available personal dosimetry service for radiation workers (film, TLD);

- the assessment of radiation exposure to the population of Australia arising from the medical and industrial uses of radioactive materials;
- the provision of expert advice to radiation users in industry, medicine, and government;
- the maintenance of the Australian standard of exposure.

Radiation Protection in the Australian States

Under the Australian system of government, health and safety of the public and the workforce is the responsibility of the separate State Governments. There are six of these located in the respective State capitals and a seventh in the Northern Territory which is in transition from the status of a Federal Territory to that of full statehood. The Federal Government is responsible for public health in the remaining Federal Territories.

The States became concerned with radiation protection in the 1950s, and a model Act and Regulations were developed. Licensing and registration programs were gradually introduced and radiation control branches established within the State Health Departments.

In Queensland, the Division of Health and Medical Physics of the Department of Health administers radiation protection, employing two physicists and two technical officers for this section of its activities. A problem which this state shares with Western Australia is the great distances between the State capital and the centres of population and industry.

In Western Australia, the Physics Division of the Health Department's State X-ray Laboratory has a staff of five physicists and seven technical officers to administer radiation protection.

In South Australia, the Radiation Control Section of the Health Commission employs ten scientists and five technical officers. The State is planning for a major uranium mining project.

New South Wales has the largest population and the greatest concentration of industry. The Radiation Branch of the Department of Health is staffed by five scientists and five technical officers.

Tasmania has the smallest area and population of the States. Its radiation protection needs are served by one health physicist.

In Victoria, the Public Health Division of the Health Commission is responsible for radiation protection. In 1983, four scientists were employed, but this number is expected to increase.

The Northern Territory is small in population, but has extensive uranium deposits. The Environmental Health Services branch of the Department of Health administers radiation protection with a staff of two scientists, and a further three are to be added shortly. The Department of Mines and Energy also has two health physicists concerned with uranium mining, milling and transport.

The seven state and territory organisations have a total of 29 scientists and 19 technical officers. These organisations provide a range of administrative, scientific and educational facilities and activities. Three provide film badge services and three have TLD facilities. Two are currently concerned with radiation safety in uranium mining and two with mineral sands.

Clinical Use of Radiation

The clinical use of radiation is well established throughout Australia. The State Health Departments ensure that Clinical departments satisfy the appropriate state legislation for radiation protection.

Radiology Departments are located in both the city and country areas although specialised X-ray equipment is located mainly in the major teaching hospitals in the state capitals. The exception has been the use of CT scanners which have proliferated into the private sector. More than 20 CT scanners are in use in Sydney alone.

The majority of nuclear medicine services are located in university teaching hospitals.

In most Australian states, radiotherapy services have been limited to one centre in each capital city. However, in Sydney there are five radiotherapy departments situated in the university teaching hospitals. As well as cobalt-60 therapy machines, linear accelerators ranging in energy from 4 MeV to 10 MeV are commonly used. There are three electron beam linear accelerators in use in Australia. One of the Sydney hospitals uses a dual purpose 10 MeV X-ray and 6-18 MeV electron beam linear accelerator.

About 60 physicists specialise in the clinical use of ionising radiation. Teaching hospitals in the major cities employ a Radiation Safety Officer (RSO) to ensure that satisfactory standards of radiation safety are maintained. In Sydney there are seven university teaching hospitals whose physicists, together with those of the State Health Department, have formed a Hospital RSO Group.

The Radioactive Ore Mining and Beneficiation Industry

The mining of radioactive ores in Australia has a history stretching back to the 1890s when uranium was first identified in Australia. In the 1930s, ore was mined at Radium Hill in South Australia to recover radium for medical purposes. The first major producer of uranium was the government owned Rum Jungle mine in the Northern Territory which operated between 1954 and 1971. Rum Jungle was followed by smaller mines in other parts of Australia, leading to a second phase of exploration and development which today places Australia as a major potential supplier of uranium. In addition to the exploitation of uranium reserves, thorium concentrates are being produced from sand mining operations in several locations.

About ten health physicists are employed directly by uranium mining companies.

Following the recommendations of the Ranger Uranium Environmental Inquiry, a cautious and considered approach has been adopted in the exploitation of Australian radioactive ore reserves leading to the involvement of several research bodies. The co-operation between mining companies and regulatory authorities has been aided by the active participation of groups at the ARL, the LHRL, and the Office of the Supervising Scientist for the Alligator Rivers Region, in addition to several university groups working under contract.

Education and Research Centres

In Australia, there are 21 universities, several institutes of technology and colleges of advanced education, and a number of other major centres which conduct teaching and research programs. Where these programs involve the use of ionising and non-ionising radiation RSOs are appointed. Their duties and responsibilities vary with the types, the

numbers and the modes of use of the sources employed and the application of full time effort is not always necessary. However, the radiation safety programs generally include administrative, monitoring, assessment, advisory, educational and disposal services.

Lectures and full courses on radiation safety are provided by some of these establishments. University and other tertiary personnel gain access to the specialised laboratories and facilities of the AAEC through the auspices of the Australian Institute of Nuclear Science and Engineering.

A meeting of campus radiation safety officers held in 1974 to exchange information and to discuss mutual problems led to the formation of the Australian Radiation Protection Society.

Australian Radiation Protection Society (ARPS)

ARPS was formed in 1975 and was affiliated to IRPA in 1977. Membership is 195. The next annual national conference, the 9th, will be held in Darwin, Northern Territory on 9-12 July 1984.

International Activities

Australia has contributed to several international studies in the fields of health physics and waste disposal, notably the OECD (NEA) study on the long term management of uranium mill tailings; and is now (and has been) represented on a number of IAEA working groups and panels, and on ICRP committees.

Other Radiation Protection Activities in Australia

In the Federal sphere, there are four items which are not covered elsewhere in this review.

- The Australian Ionising Radiation Advisory Council considers problems referred to it by the Australian Government and reports directly to the Minister for Home Affairs and the Environment.
- The National Health and Medical Research Council has a Standing Committee on Radiation Health which develops codes of safe practice reflecting international recommendations in the areas of both ionising and non-ionising radiations.
- The Standards Association of Australia prepares, inter alia, codes of practice and specifications for equipment relevant to the practice of radiation protection.

Although few manufacturers of health physics equipment currently operate in Australia, there are many distributors who provide technical advice and service for the comprehensive range of imported instruments.

THE AUSTRIAN ASSOCIATION FOR RADIATION PROTECTION (ÖVS) - REVIEW AND OUTLOOK

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The Austrian Association for Radiation Protection was established in 1966 as a national organization. The decision to establish a national society for the representation of the interests of its individual members and of radiation protection with respect to other national bodies and governmental institutions has turned out as a very promising decision in the view of the development of the ÖVS during its seventeen years of history. During these years the ÖVS has stimulated, organized and encouraged education, training and research in radiation protection. Radiation protection and related subjects are included in the curricula of various universities and are concentrated especially at the Atominstitute of the Austrian Universities (Atominstitut der Österreichischen Universitäten) and the Technical University of Wien. Education and training of all users of ionizing radiation are regulated by laws and governmental regulations. The necessary training courses are provided by various institutions in separate courses and curricula for medical applications and technical applications including physical research. Research and development in radiation protection has reached an international level at the universities as well as at various other research institutes.
The follow up of future international regulations for the definition of dose concepts and concepts for the evaluation of risk and its influence on further Austrian legis-

lation, governmental regulations and modes of procedure as well as the consideration of already existing concepts e.g. ICRP 26 are discussed. A national lab for the calibration of dosimeters and the official recognition of TLD badges as equivalent to the established film badges are further remarkable developments. As a consequence of the very sensible approach of the public opinion in Austria towards nuclear energy all aspects of quantitative evaluation of collective and individual risk and public acceptance of enhanced risk from technical and medical activities are discussed in detail. An increased effort to improve facilities and methods for low level measurements is a physical outcome of this trend. A great effort has recently been devoted to the formulation of national standards for the application and utilization of ionizing radiation. Various committees within the national bureau of standardization are active in this field.

The very well developed training courses for all persons who apply ionizing radiation have been further improved. A certificate can be obtained by each successful participant. The annual scientific meetings of the ÖVS show an improved scientific level of the presentations.

To stimulate radiation protection as a research field and to encourage advanced students to enter this field the Zakovsky award was established. Applicants for this award donated by the ÖVS must be younger than 34 years and a special paper on work in radiation protection must be presented by them. This award stimulated the interest of young scientists in the other activities of our organization.

The geographical situation of Austria gives us the opportunity to establish good relations to the Hungarian and Yugoslavian colleagues. A Hungarian-Austrian scientific meeting in Györ (Hungary) in 1981 and an IRPA regional conference in 1983 organized with the Radiation Protection Associations of Hungary and Yugoslavia in Wien are the

dominating results of this special relations.
Besides the ÖVS there exists an Austrian Medical Health Physics Society which has only graduated physicians as ordinary members. The information and training of its members as well as the representation of their interests with respect to governmental bodies and institutions are objectives of some priority for the Medical Health Physics Society. A close cooperation between the two societies in various institutions and committees and a cooperation in the field of education and training has been successfully established.
IRPA's main activities for the next future should contain every effort to avoid any desintegration between the concepts of the operational health physicists who have to do the day by day work and the theoretical concepts for phantoms, standardized measuring procedures and philosophical concepts for the evaluation of risk. A periodical exchange of views between these groups of active persons seems to be necessary and fruitful.

THE CANADIAN RADIATION PROTECTION ASSOCIATION

A BRIEF HISTORY AND DESCRIPTION OF ITS CURRENT ACTIVITIES

by

J.R. Johnson
CRPA President (1984-85)

The Canadian Radiation Protection Association was formed in 1979 and currently has over 300 members. This formation was initiated by the efforts of R.V. Osborne, who earlier had circulated a questionnaire to Canadian members of the Health Physics Society and others asking them whether they saw a need for a Canadian organization for radiation protection, and outlining several choices for such an organization. Among the choices were:

- An independent organization representing radiation protection in Canada
- A Canadian chapter of the Health Physics Society
- A division of an existing Canadian association, such as the Canadian Association of Physics

The majority of respondents saw a need for a Canadian organization, and felt that the membership of such an organization would be served best if it was completely independent. Consequently, an Ad Hoc Committee was formed to organize the proposed Association, and to draft a Constitution and Rules. The work of this Committee resulted in the formation of the Canadian Radiation Protection Association and its Constitution and Rules were unanimously approved in principle at an open meeting in April 1979, attended by over 100 potential members. However, it required a further three years and considerable effort before the Association was legally incorporated in 1982.

Besides working towards the incorporation of the Association the main activities of the CRPA in the first three years of existence were:

- To establish an efficient system of committees and procedures to run the Association. This is particularly important in a large, sparsely populated country such as Canada.

- To organize annual general meetings (AGM). Following the organizational meeting in Toronto in 1979, annual meetings have been held in Montreal, Ottawa, Vancouver and Toronto. Besides the business meeting of the Association, these meetings contain a technical program of invited and contributed papers lasting for about two days and technical exhibits. They have been attended by 100 to 150 members and non-members. Abstracts of papers presented at the meeting are distributed to all members and are published routinely in the Health Physics Journal.

- To produce a quarterly Bulletin. This bulletin is the main instrument for communication within the Association. Besides containing information about the business of the Association, it contains notices of upcoming events, short scientific and technical reviews, reviews of conferences, job openings and other items of interest to members.

Once the committees and procedures of the Association, the AGM and the Bulletin were firmly established, the officers of the Association decided to poll the membership to see what areas they would like to see the Association spending more effort on, and what areas they would like to see the Association take initiatives in. The result of the poll can be summarized as follows:

- CRPA should take a more active role in the public information forum, on its own and by supplying information, visual aids, etc. to members.

- CRPA should establish a program to assist members in keeping abreast with developments in radiation protection. This should include both the technology and the concepts of radiation protection.

- CRPA should initiate work on specific topics in radiation protection which are of current or future concern in Canada. This could be achieved by presenting briefs to the appropriate authorities, by organizing or sponsoring workshops and seminars, by supporting graduate level education in specific areas, or other appropriate means.

The officers of the CRPA have discussed the results of the poll in detail and are currently studying ways by which the recommendations of the membership can be implemented. It is felt that some progress has been made on the last recommendation, in that the CRPA is co-sponsoring an International Conference on Occupational Radiation Safety in Mining in Toronto, Canada, 1984 October 14-18. In addition, the possibility of holding workshops on the Criteria for Radioactive Waste Disposal and on the new (Canadian) Atomic Energy Control Regulations are being discussed.

Currently, the project that has the highest priority is the preparation of a proposal to host the next IRPA Congress in Toronto, Canada in May, 1988.

FACHVERBAND FÜR STRAHLENSCHUTZ E.V.
Mitgliedsgesellschaft der International Radiation Protection Association (IRPA)
Federal Republic of Germany and Switzerland

The FS was founded in 1966 and has now some 680 members and 15 affiliate members of which about 20% are in Switzerland.

1. IMPORTANT RADIATION PROTECTION ISSUES IN THE MEMBER COUNTRIES

1.1 Switzerland

There is good and close contact and cooperation between the radiation protection specialists in all fields, most of which are FS members, and also between teh authorities, the Federal Commission for Radiation Protection and the FS. Thus the FS is involved in most activities without a special need for official representation or status.

Radiation protection is up to now part of the nuclear energy legislation but shall get a separate law. The draft being worked out is based on a proposal of the Federal Comission on Radiation Protection and covers all areas of application. As far as feasible and practicable the latest ICRP recommendations shall be adopted. The excellent experiences with the now 20 years old radiation protection regulations (revised 1976) will influence the new law.

The general situation of radiation protection is good. Some optimization is still possible and some tendencies towards unncecessary perfectionism have to be kept under control. Radiation protection training is compulsory for all professionally exposed persons, and on the lower and medium levels most of them are adequately trained either during their professional formation in schools and courses or in post graduate training, but the academic curricula, especially in medicine, are not yet satisfactory. A coming task will be the formation of the next generation of health physicists.

A major problem is the objective and better information of the public, mass media, teachers and politicians on radiation protection. The public is mainly exposed to wrong and biased informations by the media and by environmental movements such as the WWF or pressure groups of all political shades, mostly minority groups from the extreme wings. Correct information becomes even more important as two national ballots on nuclear energy are coming. The growing concern on radon in houses may help somewhat to put things into perspective. Besides such short-term information problems there is the long range problem of trying to get objective information on radiation into the normal school curricula of all types. Thus part of the training and information programms have to be aimed at teachers.

1.2. Federal Republic of Germany

In the FRG radiation protection in the application of nuclear technology, radionuclides and X-rays is almost perfectly regulated by the

X-ray and the radiation protection regulations, numerous government guides and the comprehensive set of DIN standards. The abundance of rules may lead to the risk of decreasing respect in practice. The authorities intend to follow international recommendations, but quite often the German national regulations are more severe than those of other countries.

At present a partial revision of the regulations is under discussion, caused by the revised EURATOM basic standards which have adopted ICRP 26 and 30. The German draft deviates in some aspects from Euratom and ICRP and remains more conservative.

Final storage of radioactive wastes is a main problem. Large scale investigations are carried out in view of disposal of radwastes in the planned repositories of Gorleben and the iron ore mine Konrad with the goal to estimate the resulting exposure of the population. For low level wastes a limit of specific acitvity is discussed below which such wastes might be disposed of as conventional waste.

A main task in radiation protection is training. The Federal Ministry for Internal Affairs (BMI) has issued a guideline on required knowledge which is based on a detailed catalogue of training topics developed by the FS. Training institutions for radiation protection officers and qualified experts will now need an authority approval.

Like other countries the FRG has in the past years carried out a wide survey of the radiation exposure of the population from natural radionuclides, expecially Rn-222 and Rn-220. Countermeasures for reduction of excessive radon daughter exposures are considered.

Other important topics are emergency planning for nuclear installations, the accident consequence model of phase B of the German Reactor Safety Study, the retention of volatile radionuclides in a planned reprocessing installation and the basic accident calculation scheme for the BMI guideline on the evaluation of the design of PWR power plants according to para. 28(3) of the radiation protection regulation.

In the FRG,too, the situation regarding objective information of the public on the hazards of ionizing radiations is far from satisfactory, because the mass media often are not ready or able to present factual and objective informations on such topics. There is almost no public discussion of radiation in medical applications where there would still exist the best opportunities to reduce collective dose by better information and protection measures.

2. PRINCIPAL CURRENT ACTIVITIES OF THE F S

In the past two years one of the main activities was the preparation of this 6th International Congress of IRPA.
Every year the FS organizes a scientific conference on a specific topic, sometimes in cooperation with other scientific societies or IRPA Associate Societies of neighbouring countries. The proceedings are published in the FS report series and are distributed free to all members, to representatives of IRPA and its societies and to important libraries. The annual general assembly is held during these conferences. A quar-

terly bulletin is sent to all members.

In order to keep the FS active also inbetween these annual meetings, a number of Working Groups were created twelve years ago and new ones followed. Bound by a few general rules these working groups organize themselves and have the following tasks:

- Exchange of experiences among specialists in the same field and information and technology transfer to the FS members
- Elaboration of recommendations and manuals on practical application of regulations, guidelines and rules. The results are published as FS reports and distributed to FS members.
- Preparation of FS statements and position papers on topics of official consultations
- Cooperation with the scientific program committees of annual conferences
- Quality control and improvement of methods in internal and external dosimetry by intercomparison programs etc.
- Contacts with other professional and scientific societies, standardizing organizations, committees and institutions in the relevant fields.

The present Working Groups cover the following topics: Incorporation Monitoring, Training, Working Place Monitoring, Environmental Monitoring, Dosimetry of External Irradiation, Radiation Protection at Power and Research Reactors, Public Relations, Non-ionizing Radiations. Among the special results of the working groups are recommendations on training requirements and programs worked out on behalf of the BMI and a very successful slide collection on radiation protection in nuclear energy which was sold in 200 copies to schools, information officers and health physicists as an aid for better information of the public. The FS reports number over 30 titles, half of them being conference proceedings.

The FS tries to improve contacts and cooperation with authorities, professional and scientific societies and organiziations, mass media and the public, the latter e.g. by press conferences during annual meetings. The nonpolitical basis and the uncomplicated constitution and management of the FS allow flexible, informal and unconventional approaches and, if necessary, quick reactions in a pragmatic and efficient way which also facilitiates penetration of "official barriers". We try a new idea first and see whether it works before we formalize anything.

A major medium range action of the FS has been started in 1983, in a first stage by a working group of the Board of Directors. The goals are a critical evaluation of 20 years of practical experience in radiation protection and with the existing regulations and the development of a basis for an optimized radiation protection concept and for modernized and reasonable radiation protection regulations. Recent ICRP recommendations shall be included as far as practicable. In a first step an action program and basic theses are prepared, then the other working groups and the members will participate or be consulted. The resulting concepts and proposals shall be published not only in scientific reports but in suitable form also in the media. Although this remains an independent action of the FS, close contacts will be kept to the authorities. By this action the FS also intends to establish itself in the view of

the public and of the politicians as the professional organization in radiation protection matters in contrast to political pressure groups with "scientific" disguise.

Coming events of the FS will be the 1985 annual conference on "Radiation exposure of the population, including medicine and radon" and in 1986 the 20th anniversary conference of both the FS and the Austrian society (ÖVS), which probably will become a regional IRPA Congress and may be devoted to experiences and conclusions form 20 years of radiation protection.

3. PROPOSALS TO I R P A

- The IRPA Bulletin was a good idea and must be continued as the only sign of life of IRPA between the Congresses.
- The representation of IRPA on the Editorial Board of the "Health Physics Journal" should be increased.
- The complicated constitution and rules of IRPA should be revised and simplified. They have turned IRPA, the EC and the GA into rigid, slow and almost selfsustaining "organisms" which barely manage to keep IRPA alive, which only with long delays are able to respond to any input or problem and whose useful output is far too small compared to the available potantial.
- After 20 years of build-up and consolidation IRPA should now be turned into a dynamic and active organization which is able to promote the needs of radiation protection in an efficient and timely way, which acts, not only reacts, and which leads the way in creating and promoting new concepts of applied radiation protection and establishes close cooperation with those scientific bodies which deal with the fundamental concepts. Active IRPA participation is also needed in order to counterbalance the growing majority of authority representatives in international bodies dealing with rad. protection.
- For efficient and quick action IRPA should form permanent working and expert groups on important topics in order to work out IRPA positions on drafts of new recommendations, standards ets., in consultation with the Associate Societies. At present there is only one IRPA committee (INIRC) doing work which is really related to radiation protection, all other committees have purely legal or administrative character with the sole task to try to make the complicated rules work.
- IRPA should try to improve coordination of scientific conferences in cooperation with international organizations and associations in order to reduce the number of collisions of topics and dates.

4. RECENT PUBLICATIONS OF THE F S

Proceedings:

FS-78-18-T	"Radioaktivität und Umwelt", 12.Jahrestagung 1978 Norderney, 2 Vol.
FS-79-19-AKD	"Tendenzen in der Personen- und Umgebungsdosimetrie" Seminar München 1978
FS-79-20-T	"Radioaktive Abfälle", 13.Jahrestagung/7.Regional Congress IRPA, Köln 1979 FS/ÖVS/NVS
FS-80-25-T	"Industrielle Störfälle und Strahlenexposition", 14.Jahrestagung Jülich 1980
FS-82-27-T	"Radiologische Auswirkungen von Kernkraftwerken und anderen kerntechnischen Anlagen auf den Menschen und seine Umwelt", 15.Jahrestagung Lausanne 1981, FS, SFRP
FS-83-30-T	"Strahlenschutz-Messtechnik", 16.Jahrestagung München 1982
FS-83-32-T	"Strahlenschutzaspekte bei radioaktiven Kontaminationen", 17.Jahrestagung Aachen 1983

Publications by Working Groups:

FS-78-15-AKU	"Loseblattsammlung Arbeitskreis Umweltüberwachung", 1.Teil 1980, 2.Teil 1982
FS-81-21-AKI	"Inkorporationsüberwachung auf Jod", Loseblattsammlung AKI, 1981
FS-81-26-AKA	"Lernzielkatalog zur Fachkunderichtlinie" 1981
FS-81-28-AKÖ	"Diasammlung Strahlenschutz bei der friedlichen Nutzung der Kernenergie" 1982
FS-82-39-AKD	"Technische Empfehlungen für Festkörperdosimeter zur Umgebungsüberwachung"
FS-83-33-AK NIR	"Loseblattsammlung Nichtionisierende Strahlung: Ultraviolett" (to be published)

INDIAN ASSOCIATION FOR RADIATION PROTECTION

Brief History, Objective and Activities

The Indian Association for Radiation Protection (IARP) was formed in 1968 and was registered under the Public Trusts Act, in 1969.

The objectives of the Association are:

- to bring about proper awareness of the hazards from ionising radiations amongst their users in particular and the public in general;
- to encourage adoption of appropriate means for avoiding or reducing radiation exposures in the applications of ionising radiations and nuclear technology in the country, such as power generation, industry, medicine, agriculture, scientific research etc., thereby maximising the benefits which are derived out of these applications while minimising the risks; and
- to facilitate contacts and exchange of information amongst specialists in radiation protection and related disciplines in the country and with their counterparts in other countries.

The Association has a total membership of nearly 200 specialists including 120 life members from different parts of the country. It is managed by an Executive Committee, headed by Dr. K.G. Vohra, a well-known authority in the field of radiation protection. Dr. A.R. Gopal Ayengar, the internationally renowned radiation biologist and Dr. A.K. Ganguly, an international authority on health physics are Presidents Emeritus of the Association. The Association has on its rolls numerous scientists, medical doctors and engineers who are well-known in India and abroad for their scientific contributions and technical accomplishments.

The Association has successfully organised the following meetings:

First Annual Conference on Radiation Protection, Bombay, April 1973.

First Topical Meeting: Safety Standards for Siting of Nuclear Installations, Bombay, June 1974.

First Asian Regional Congress on Radiation Protection, Bombay, December 1974.

Second Topical Meeting: Accidental Criticality, Bombay, June 1975.

Third Annual Conference on Radiation Protection, Delhi, January 1976.

Third Topical Meeting: Protection from Modern Industrial and Environmental Risk - Lessons from Nuclear Industry (Jointly with Society for Clean Environment and Environmental Mutagen Society of India), Bombay June 1976.

National Seminar on Diagnostic Radiology & Radiotherapy with particular reference to radiation protection in medical institutions and hospitals (Jointly with Department of Atomic Energy and WHO Regional Office for South-East Asia), Bombay, November 1976.

Fourth Annual Conference on Radiation Protection, Madras, March 12-14, 1977.

Fifth Conference on Radiation Protection, Gwalior, February, 14-16, 1978.

Sixth IARP Conference on 'Safety Aspects in Nuclear Fuel Cycle', Bombay, March 7-9, 1979.

First Workshop on "Radiation Protection in Diagnostic Radiology", Trivandrum, February 18-20, 1980.

Seventh IARP Conference on 'Radiation Environment in Life and Earth Sciences', Baroda, February 25-27, 1980.

Second Special Symposium on 'Natural Radiation Environment', Bombay, January 19-23, 1981.

Second Workshop on 'Radiation Protection in Diagnostic Radiology', Agra, November 25-27, 1981.

Ninth IARP Conference on 'Radiation Protection in Nuclear Science and Technology', Kanpur, February 23-26, 1982.

Tenth Annual Conference on 'Radiation Risk Assessment - Present and Future Considerations', Bombay, March 1-4, 1983.

The Association was admitted in May 1970 as an associate society of the International Radiation Protection Association (IRPA). Members of IARP are also members of the international Association. Some members of the Association are actively associated in the work of a number of committees of IRPA. Members of IARP are eligible to obtain at a special concessional subscription rate copies of Health Physics a professional journal published by IRPA's associate society in USA and Canada.

A branch of the Association was formed in New Delhi in January 1976. The Association arranges scientific lectures, educational film shows, etc. at regular intervals.

The Indian Association for Radiation Protection (IARP) publishes a scientific quarterly, "BULLETIN OF RADIATION PROTECTION" since 1978, which is issued free of cost to the members of the Association. The Bulletin carries original articles, technical notes, product information, commentaries, news excerpts etc., in the field of radiation protection and health physics. Periodically special issues devoted to specific topics are brought out. Some of the special issues covered, 'Thermoluminescence; Dosimetry and Applications", "Biological Hazards of Microwaves and Protection Aspects", "Safety Aspects in Fuel Processing Facilities". Selected papers from the Annual Conferences of the IARP are brought as Proceedings Issue of the Bulletin.

The present office bearers of the Association are:

EXECUTIVE COMMITTEE

President	:	Dr. K.G. Vohra
Vice-President	:	Dr. C.M. Sunta
Secretary	:	Dr. P.S. Iyer
Treasurer	:	Mr. L.H. Peshori
Members	:	Dr. T.S. Iyengar
		Mr. R.K. Kher
		Dr. S.K. Mehta
		Dr. A.R. Peddy
		Mr. S.D. Soman

BOARD OF TRUSTEES

Members	:	Shri T. Subbaratnam
		Shri P.H. Patel
		Dr. K.C. Pillai
		Dr. V.R. Shah

THE ISRAEL HEALTH PHYSICS SOCIETY

T. Schlesinger
Soreq Nuclear Research Centre
Yavne 70600, Israel

Objectives

The objectives of the Israel Health Physics Society (IHPS) are to promote and improve radiation protection and health physics as a profession, to improve dissemination of information and knowledge between the public and the professionals engaged in Health Physics and related fields, to promote research and development and educational programs in Health Physics and related scientific disciplines and to cooperate with national and international associations which have similar objectives.

History, and Membership and Activities

The Israel Health Physics Society was founded in 1963, following the initiative of the late Prof. Yehuda Feige, a member of the temporary Executive Council of IRPA and one of IRPA's founding members. A delegation representing the Israel Health Physics Society, then 32 members participated in the first General Assembly of IRPA in Rome in September 1966. In the second International Congress of IRPA in Brighton, England, in May 1970, Dr. Feige, the president of the Israel Health Physics Society, was elected to be a member of IRPA's second Executive Council.

The IHPS developed since then and became an established member of Israel's Scientific Associations. Its membership in 1983 is close to 70 and includes Physicists, Engineers, Physicians and Technicians engaged in various fields of professional activities related to the use of radiation and nuclear technologies in medicine, industry, research and education.

Among the major activities of the IHPS in recent years are:

1. Organization of the regional congress on radiation protection in Jerusalem in 1973 (with 300 participants).
2. Organization of the 5th International Congress of IRPA in Jerusalem in 1980 (with 600 participants from about 30 countries).
3. Publication of a newsletter of the IHPS about 3-4 times a year.
4. Participation in the organization of the Annual Meeting of the Israel Nuclear Societies with a special session on Health Physics and Radiation Protection.
5. Financial help to its members in subscription to the Health Physics Journal.
6. Contacts with the IRPA and sending delegations to IRPA's International Congresses and IRPA's General Assemblies.

NORDIC SOCIETY FOR RADIATION PROTECTION
AIMES AND ACTIVITIES

Per Grande
National Institute of Radiation Hygiene
Copenhagen, Denmark

Nordic Society for Radiation Protection is a society of professionals aiming at improving radiation protection in the five Nordic countries (Denmark, Finland, Iceland, Norway, and Sweden) by exchange and distribution of information. Membership: approx. 400.

Ordinary meetings are normally held every 3rd year. The country in turn for the meeting has the precidency, which accordingly changes every 3rd year.

Additionally "tema meetings" might be arranged, i.e. meetings dealing with a subject of special interest/importance/actuality, e.g.
"Natural Radiation in our Environment", 1980 (mainly Rn), and
"Risk Assessment and Risk Perception", 1983.

President: Per Grande, National Institute of Radiation Hygiene,
Frederikssundsvej 378, DK-2700 Broenshoej. Tel.: 2 94 37 73

Secretary: Bente Lewinsky, same address.

Treasurer: F. Heikel Vinther, Risoe National Laboratory, DK-4000 Roskilde.
Tel.: 2 37 12 12

The 1984 oridnary meeting will be held in Copenhagen October 10.-12.

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RADIATION PROTECTION IN SOUTH AFRICA

J Kruger
NUCOR
Private Bag X256
Pretoria 0001
South Africa

1. THE SA ASSOCIATION FOR PHYSICISTS IN MEDICINE AND BIOLOGY

The South African Association for Physicists in Medicine and Biology (SAAPMB) was formed 24 years ago as a Medical Physics society. The scope of the association was extended in the late 1960s to cover the wider field of Biophysics, and it became the national society for Health Physics. The SAAPMB, with a membership of about 100, today has two groups, one for Medical Physics and one for Health Physics. The Health Physics Group is affiliated to IRPA. The association organises annual congresses dealing with Medical as well as Health Physics topics. The members of the Health Physics Group consist of people occupied on a full-time basis in radiation protection, as well as people occupied only part-time in radiation protection and sharing this with their other responsibilities. The Group has to satisfy the needs not only of professional Health Physicists, but also of Health Physics technicians and non-professionals.

2. THE SCOPE FOR RADIATION PROTECTION IN SOUTH AFRICA

South Africa is well advanced as far as applications of ionising radiation are concerned. The field extends through the mining and processing of source materials, the development of nuclear power and nuclear technology, industrial applications and the use of radiation in medicine. Major nuclear activities are centralised, so that only a few larger organisations employ personnel on a full-time basis for radiation protection. The Atomic Energy Corporation and the Government Department of Health and Welfare as regulatory authorities, the Nuclear Development Corporation, the Uranium Enrichment Corporation, the Council for Scientific and Industrial Research, the SA Bureau of Standards and the Electricity Supply Commission are the most typical examples of such organisations. As for the rest of the field, that is medical applications, educational institutions, research organisations, uranium mining and milling and industry, radiation protection is performed as a part-time function. One typically finds that the most suitable person, e.g. from the Physics department of a university, the industrial radiographer, a radiologist for individual medical practices, or the medical physicist for a hospital complex, has to perform this duty. Such personnel frequently have a minimum of training in radiation protection, either in the form of short courses or training in a related field, or by on-the-job training.

3. MAIN OBJECTIVES OF THE HEALTH PHYSICS GROUP

3.1 Promotion of interest in radiation protection

The first objective would be to interest and attract Health

Physics practitioners into the society. Drawing in the full-time practitioners has, on the whole, been successfully accomplished, although there still is, surprisingly perhaps, a lack of interest. This is borne out by the fact that the activities of the Health Physics Group, papers presented at conferences, and the promotion of radiation protection in South Africa, are carried by a few individuals. To interest the part-time practitioner remains a serious problem. Although these people may not practise health physics to the same depth and scope as those in full-time occupation, the number of people involved and the field they cover, are large. Furthermore, these people need assistance and guidance more than the full-time practitioner. Such support could effectively be provided by the Health Physics Group if active interest could be cultivated amongst the part-time practitioners. It is typical that only 20-30 % of papers at annual SAAPMB meetings are on radiation protection, which illustrates that the objective of promoting interest is a continuing process.

3.2 Training

The next objective would be to educate on the informal as well as the formal level. The limited number of interested people and the size of the country make the task of formal education through approved courses, a formidable one. Educational institutions are not interested in creating regular courses if there are no regular students available. By personal efforts and promotion through the Health Physics Group, courses, to be provided on request, were made possible at the Technikons on the technical level, and at universities on the post-graduate level. In all these courses, members from the SAAPMB are required to be actively involved in the lecturing and examining, thus assisting the educational institutions in a field where expertise is not necessarily available from their own sources. Training on the technical level has been running for a few years with very satisfactory results. Training on the academic level on a more regular basis has just been initiated. Although until now it was considered that a basic degree in Physics is required for academic training, and the training is vested in the Physics faculties, the question does arise whether this is, in fact, the most suitable arrangement.

3.3 Professional guidance

The small number of full-time Health Physics members has to guide and lead the part-time members, because the latter individuals are concerned primarily with the application of nuclear techniques, be it in medicine, science or industry, and radiation protection is only of secondary concern to them. The full-time members are also responsible for formulating radiation protection philosophy and interpreting national and international standards

and practices. This is done firstly through each individual's professional employment, and secondly, through the Health Physics Group as a professional society. Guidance is provided not only to SAAPMB members, but also to national authorities.

4. CURRENT TOPICS

Although the ALARA principle has been applied intuitively in the past, optimisation of dose in quantitative terms in accordance with recent concepts, is receiving considerable attention. It is foreseen that the practice of comparing radiological risk with risks from other sources in this optimisation process, will come to the foreground. A relevant issue is the de minimus dose, on which international guidance is eagerly awaited. The implementation of SI units in a way that will ensure a smooth and safe transition is a matter of urgency. Recommendations from the ICRU in this regard, and the position of the unit of exposure, are urgently awaited. Other issues under review are the methodology of internal dose estimation, the use of radiation risk factors and dose limitation not only with respect to radiation workers but to other exposed individuals (e.g. in the medical field), and the position of the radiation hazard in relation to other occupational hazards. The support from IRPA in the past and the role of IRPA in enhancing international communication and cooperation, are of great significance, especially to a small society as geographically isolated as South Africa. An opportunity to exchange information with other societies, through IRPA, is considered to be of vital importance.

SOME THOUGHTS ON RADIATION PROTECTION ISSUES IN THE UNITED KINGDOM

R.G.C. Grix
Council Member, British Radiological Protection Association,
Chairman, Association of University Radiation Protection Officers.

Important National Issues

A matter of great importance and concern in the U.K. is the imminent introduction of legislation to control the use of ionising radiation. The problems associated with this introduction are not new, and since the implementation of new laws in 1974 dealing with the whole field of occupational hygiene, considerable thought has been given to the difficult process of extending control from relatively simple industrial uses to other fields such as scientific research in medicine. In the 10 years that have elapsed since the introduction of the laws, there have been several attempts to produce comprehensive legislation which have been abandoned because of the appearance of new external factors such as the Directive of the European Communities and the advice given in ICRP 26. The current exercise, which threatens to be brought to fruition in the immediate future, is the result of several years of consultation and is based on a working party convened by the responsible government authority. The group has contained representatives from employers, workers, and individual radiation protection experts and has produced consultative documents which have been published, inviting comments from interested parties. At the same time a National Advisory Body has produced notes for guidance on how different aspects of the legislation should be implemented.

Current Involvement of BRPA

Recognising the danger that even well-intentioned legislation can inhibit productive and scientific activity, BRPA has been fortunate in having some of its members sitting on the working party drafting the regulations. Thus it has been able to keep its membership informed of the trends and implications of the proposed legislation. Additionally there has been feedback to the working party of the need for scientific consistency and flexibility.

The difficulty has been clearly seen that in attempting to legislate for all aspects and all uses for ionising radiations, the proliferation of detail can reduce the law to nonsense. Sanity and flexibility can only be preserved by permitting the application of professional judgement in individual situations, and BRPA has become even more aware of the need to enhance the esteem and practice of the Radiation Protection profession so that this can be available to all.

In extending professional advice outside the area of legislation, the society have published two "Manuals of Good Practice" concerning the Radiation Protection of the patient during medical investigation. The first of these on Diagnostic Radiology was published in 1980 and enjoyed success both in this country and abroad. This year a sequel was published on Nuclear Medicine and a similar success is expected.

Future Objectives

This society is a federation of interests which range across the discipline of operational health physics, medicine and education. In identifying the need for a profession of radiation protection, it has become very aware of the

importance of reconciling sectional interests if a profession of radiation protection is to be recognised externally. There is a widespread fear that so different are the aspects of radiation protection in, say, hospitals compared to Nuclear power stations, that there could never be an interchange between staff and therefore never a corporate identity. Some of the respected professions using radiation in their work have little interchange between different aspects of the work and this has not appeared to be a disability. This society has however an interest in obtaining more recognition from the public for persons able to give professional advice. It has striven to reduce the internal disagreements that exist (often publicly) in its various sections. Just as engineers have separate sectional interests, for instance mechanical, electrical, chemical, etc., similarly radiation protection has operational, medical and educational divisions that are separate but should be regarded as of comparable importance.

Another vital factor is that it is important that professional advice should be given with a mind to financial consequences. It does not need an expert to get 100% safety by closing down production altogether. Our art must be to obtain maximum safety with minimum expenditure and interruption of work, and to show the public that safety can sometimes pay for itself.

The Role for the IRPA

Members of IRPA will have no doubts as to its value in fostering the interchange of information internationally. An important bonus is the human relationships that are generated in the process. A vital role which has emerged and should be developed is the collection and collation data on cause and affect relationships which are so important to those of us who do not have investigatory sources of our own. Finally we have become increasingly aware that the IRPA should be eternally vigilant about legislators. Their raison d'etre is to control everything and they are very easily attracted to new sources of data with the intention of using them for purposes which were not intended scientifically.

IMPORTANT RADIATION PROTECTION ISSUES
HEALTH PHYSICS SOCIETY

Bryce L. Rich, President
Health Physics Society
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, Idaho 83415

The Health Physics Society views the following as the important radiation protection issues facing the nuclear industry and particularly the radiation protection profession at the present time.

1. Public Information. Inadequate public information on radiation effects constitutes a severe handicap related to acquiring the beneficial utilization of nuclear power and other nuclear devices and materials. In the absence of adequate public information, those with special social motives, as well as poorly informed officials, have created an environment in which a much needed technology is being withheld from public use. It is essential that radiation be placed in proper perspective as a potential public and industrial hazard which can be controlled, but which has benefits thus justifying some level of risk. The role of radiation protection societies in educating the public is essential.

2. ALARA. As ALARA programs are pursued and exposure reduction is obtained, costs per sievert continue to rise. Radiation protection societies should be on the forefront in assuring knowledgable judgments related to cost/benefit.

3. Nuclear Safety Concerns. Plant nuclear safety concerns (particularly at nuclear power generating plants) are resulting in an increase in worker exposures as a result of the need to provide for in-plant inspection and preventative maintenance to ensure the functioning of safety-related equipment. This is recognized as a legitimate need for radiation exposure expenditure but can easily be pursued to excess. The judgment process of working compromise between ALARA and nuclear safety inspection programs needs radiation safety input.

4. Radiation Protection Guides. ICRP 26 Radiation Exposure Limitation concepts are the subject of considerable discussions within the US nuclear community -- as governmental agencies attempt to implement the concepts. The Health Physics Society recognizes this as a significant radiation protection issue.

5. Accident Impact. The Three Mile Island accident impact on the nuclear industry (not only within the US but worldwide) carries with it significant radiation protection issues. This item is related to the public information issue discussed in 1. above but is an issue in which the radiation protection community should be intimately involved as the lessons from this accident are learned and applied.

6. Nuclear War Issues. The probability of nuclear war is a major radiation protection issue from both moral and technical standpoints. The issues figure heavily into political decisions, and affect the world society in general. The radiation protection community must make their services available at a proper level.

7. Education and Training. Professional and technology-level education and training programs in the radiation safety areas are experiencing difficulty producing personnel with adequate educational background for the coming nuclear program needs. The radiation protection societies should take a positive roll in providing and/or stimulating support in this area of need.

CURRENT ACTIONS AND RECENT ACHIEVEMENTS

The Health Physics Society is making a significant effort to address these issues, which can be summarized as follows:

Public Information

1. The HPS has a scientific and public issues committee under the chairmanship of the immediate Past-President which responds to public issues on a timely basis.

2. The Society's Public Information Committee is active in performing a public information service. Examples of recent accomplishments has been production of a speaker's kit for each of the 39 Chapters, which includes demonstrations and slides. Local Chapters are active in varying degrees, producing films, news media interface, and speaker services throughout the nation.

3. The HPS State and Federal Legislation Committee is also becoming increasingly active in addressing radiation protection standards as they are produced at every level.

Technical Advances

1. The HPS is active in sponsoring technical programs and symposia, such as the recently completed International Beta Dosimetry Symposium, participation in the AAAS program, participation in the recent World's Fair displays at Knoxville, etc.

2. The HPS is active with other scientific societies, such as the American Nuclear Society in developing standards and community education projects.

3. Through the monthly NEWLSETTER, society members are kept abreast of current events. The HPS JOURNAL also provides significant interchange of technical information and dialogue between professionals in the subject areas mentioned.

4. The HPS JOURNAL has been made available to other IRPA societies to stimulate technical and scientific interchange.

Standards.

The HPS is engaged in a vigorous standard effort through managing the ANSI N13 committee and in generating appropriate standards through a Standards Committee activity in the Society. This activity involves 5-10% of the total professional society membership in generating appropriate standards for use within the industry.

Education and Training.

The HPS currently provides five fellowships for graduate level education at approved universities in the area of Radiation Safety. Continuing Education programs at both the Chapter and National levels provide opportunities for technical upgrade of HPS members.

MAJOR OBJECTIVES AND FUTURE DEVELOPMENT

In addition to the efforts outlined above, the Society is pursuing future developments in the following areas:

1. The HPS has established an ad hoc committee to study long range plans for the Society. At this point in the maturation process of the society, it is appropriate to reevaluate progress and establish long range goals for future effort guidance. This long range planning study will be completed within the next year.

2. Increased public information efforts are being planned.

3. Sponsorship of scientific symposia and similar studies are being planned.

4. A significant proportion of the Society's resources are being directed to funding of special fellowships for professional radiation safety education and training and will be increased as resources allow.

5. The Society recognizes the need for cooperative interrelationships with other scientific societies and communities.

PROPOSALS FOR IRPA IMPROVEMENT

Suggestions for consideration by the IRPA council for significant worldwide radiation protection community emphasis are as follows:

1. Public Information on a worldwide basis is perceived as a significant problem. We suggest that IRPA sponsor means of increasing radiation protection information communication to the public on a worldwide basis.

2. We feel that an increased effort in the area of standards development should be made by the radiation protection community in general and coordinated on an International basis. IRPA should be in a position to stimulate and coordinate some aspects of this standard-making effort.

3. It is felt that IRPA should become more progressive in sponsoring regional topical symposia to address technical needs of the radiation protection community in general.

4. An important role which IRPA could perform would be in the area of maintaining an accident/incident registry in which lessons learned from various parts of the industry could be communicated to the radiation protection community on a shorter time frame through some routine publication.

CURRENT CONCERNS, TRENDS AND ACHIEVEMENTS OF THE YUGOSLAV RADIATION PROTECTION ASSOCIATION IN THE FIELD OF RADIATION PROTECTION

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1. Introduction

One of the major characteristics of the Yugoslav Radiation Protection Association (YRPA) activity is that the part of work being done by the Association is very closely related with the work on many problems in the field of radiation protection which is done by the research institutes and government agencies. Society is very often initiating activities in different branches of radiation protection such as are research, legislature, and almost always takes stands on the current radiation protection issues in the country. Beside that YRPA, as one of the IRPA founding Societies, is very active in promoting international cooperation of the Society's membership.

Here, in this review, we will shortly elaborate on some of the current issues, trends and efforts of our Society, some of which might be, and the most probably are, of interest for the other Societies too.

2. Society's Current Activities and Efforts on the National Level

2.1 Yugoslav Radiation Protection Association has always been in our country an active partner in the field of radiation protection legislature. In recent years attantion in this field was focused on the application of the ICRP Recommendation, Publication 26. This effort has resulted in a new Yugoslav "Law for Protection Against Ionizing Radiation", which opened possibility for more detailed application of the ICRP three basic principles: (a) justification of practice, (b) optimization of protection and (c) limitation of individual risk. These principles will be incorporated into nine Regulations which follow the Radiation Law. Our Society published a book on this subject, which is actually a collection of papers of the competent Society's members. This book will serve as

a some kind of guide for future work on Radiation Protection Regulations. It could be said that the application of the ICRP latest recommendation in the national legislatures is still an international problem. Special difficulties arise when these recommendation have to be applied in practice.

2.2 In Yugoslavia, as well as in many others countries, one of the problems in radiation protection is population exposure due to sources of ionizing radiation in mass application,such as are fire warning devices,radioactive lightning rods etc. Tens, and maybe hundred, of thousand of these sources have been mounted, and still are being mounted in the country,very often in the inhabited areas. Our society took this issue very seriously, and the whole activity was summarized in a booklet which now serves as a some kind of guide for the protection against exposure due to this kind sources. It should be emphasized that this is an international problem too since these sources, produced in few countries, are world wide applied. Very often competition between companies which are making these kind of sources ignores radiation problems.

2.3 Medical exposure to radiation is a concern of our Society,as probably is of all others. Much of the effort has been done by the Society to improve situation in this field. The most important recent step in that direction done by the Society was a try to include into Radiation Protection Regulations few items regulating medical application of radiation on the bases that: "Unnecessary exposures should be avoided; necessary exposures should be justifiable in terms that would not otherwise have been received; and doses actually administred should be limited to the minimum consistent with medical benefit to the individual patient."

2.4 The realibility of the personnel dosimetry,with film bages or TLD, has occupied and is still occupying attention of our Society. One of its committies is studing this problem,esspecially the question of the interelationship between personnel dosimetry results and results obtained by the periodical,usually once in a year,control of the working conditions and radiation fields arround sources. Correct results and conclusions could

probably be obtained only by combining these two approaches of irradiation control.

2.5 Problem of Protection against nonionizing radiations is rather serious in our country. Even though that research in this field is going on in many Institutes, and sources of nonionizing radiation are being rather widely applied, standards and limits for exposure to this radiation do not exist. Society had included problem of the protection against nonionizing radiation into its Program, and we are now about to initiate precedure, at the responsible government agencies, for establishment of the corresponding limits for nonionizing radiation.

2.6 Program and activities on indoor exposure to Natural Radiation and related risk assessment has not yet been defined on a national level. Executive Council of YRPA has decided recently to initiate this program by organizing the first Yugoslav meeting on indoor exposure to be held in September 1984.

3. Society's Current International Activities and Efforts

3.1 Yugoslav Radiation Protection Association is one of the IRPA founding Societies and as a such has been very active in promoting international cooperation of its membership. Just to mention that the members of YRPA, as well as the YRPA itself, have been very active in organizing and participing in such activities as are Yugoslav Summer School on Radiation Protection, Few Meetings sponsored by the IAEA, etc. These efforts will continue in the future.

3.2 In september of the year 1983 in Vienna was held "XI.Regional Congress of IRPA; Austrian-Hungarian-Yugoslav Radiation Protection Meeting", organized by Austrian Association for Radiation Protection in association with Hungarian Radiation Protection Society and Yugoslav Radiation Protection Association. It has been agreed by these three Associations to have such a meeting again, organized in Yugoslavia, the most probably in September 1986.

4. Conclusion

All radiation problems discussed, or just mentioned, in this short review are international in scope, and should be discussed and solved internationaly, appriciating differencies from country to country.Each Society can make considerable contribution to the solution of these problems in its own country, but on international level too. One of the ways to achieve the latest is active cooperation and work of our national Societies through IRPA.

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