Seventh International Congress of the
International Radiation Protection Association

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IRPA 7

Seventh International Congress of the
International Radiation Protection Association

Sydney, 10-17 April 1988

RADIATION PROTECTION PRACTICE

VOLUME I

Organised by the Australian Radiation Protection Society

PERGAMON PRESS
SYDNEY · OXFORD · NEW YORK · TORONTO · FRANKFURT

Pergamon Press (Australia) Pty Ltd
19a Boundary Street, Rushcutters Bay, NSW 2011, Australia

Pergamon Press Ltd
Headington Hill Hall, Oxford OX3 OBW, England

Pergamon Press Inc.
Maxwell House, Fairview Park, Elmsford, NY 10523, USA

Pergamon Press Canada Ltd
Suite 104, 150 Consumers Road, Willowdale, Ontario M2J 1P9, Canada

Pergamon Press GmbH
6242 Kronberg-Taunus, Hammerweg 6, Federal Republic of Germany

Pergamon Press SARL
24 rue des Ecoles, 75240 Paris, Cedex 05, France

Preliminary pages typeset by Authentic Print, Sydney
Printed in Australia by Macarthur Press Pty Ltd, Parramatta

ISBN 008 034440 2

Cover design: Hand Graphics

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ROLE OF PROBABILISTIC EVENTS IN THE APPLICATION OF THE JUSTIFICATION CRITERION

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ABSTRACT

Probabilistic events (potential accidents) at large industrial installations - specifically nuclear power plants - play a major role in the public debate. According to the IAEA (ICRP) criteria an installation is only justified if it results in a net benefit for the society. It is logical and in accord with the importance assigned by public opinion that probabilistic events be included in a justification evaluation.

This also entails an assessment of the probability of occurrence and the consequences of such events. As regards the consequences, however, one has to keep in mind that the impact on a given society not only depends, for example, on an absolute number of casualties but also to a great extent on the fraction of the population involved: a hundred casualties equally distributed over 100 towns is different from 100 casualties in one town.

For the application of the optimization criterion it has been recommended that a beta factor be used to describe the more than linear increase in risk aversion which occurs as the level of risk to a single individual increases.

The application of this basic concept for the evaluation of a societal risk caused by probabilistic events may be even more useful than its application in the optimization process. In addition, the ability of a society to heal a limited amount of harm and inversely the decrease in this ability when an increasing fraction of the society is involved could be included in this factor.

It has to be kept in mind that an individual has a limited life expectancy. Therefore even the worst case causes only a shift in that value. A society, however, usually wishes to be "immortal" and will therefore have a higher aversion to any risk which may threaten this desired immortality.

Such a beta factor will therefore approach large values if a large fraction of a given society would be injured by an accident or even if it would only be forced to leave its homeland. This effect will have to be offset by a corresponding decrease in the probability of occurrence of such events.

A beta factor depending upon the fraction of a society affected by an accident can therefore vary, depending on whether the consequences of such an event are being considered, for example, by the authorities of a town adjacent to a plant or by a central government.

The paper addresses potential applications of this methodology as well as some of the problems which may be encountered.

RADIATION RISKS: THE ETHICS OF HEALTH PROTECTION

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Since the inception of commercial uses of nuclear technology, radiation protection standards established by regulatory agencies have reflected moral concerns based on two assumptions: (1) that the linear, zero-threshold hypothesis derives from scientific data in radiobiology which are virtually conclusive; (2) it is morally "better" for public health protection to assume that any radiation exposure, no matter how small, has some harmful effect which can and ought to be prevented. These assumptions have been reenforced by a popular belief that, since World War II, technological man has introduced into the biosphere enormous quantities of synthetic toxic substances contaminating an otherwise benign natural world. These include "unnatural" radiation sources as well as huge quantities of "sinister" chemicals with no natural equivalents.(1) Moreover, official policy has enshrined a quasi-dogma about synthetic substances, chemical and radioactive: it is "prudent" to assume that "even the most minute dose, even a single molecule, may trigger a lethal change in a cell that will cause it to multiply malignantly."(2)

In the past few years these beliefs and related assumptions have received closer scrutiny, revealing hidden reasons for regulatory selection of radiation risks as objects of paramount ethical concern, with the result that greater risks to health have escaped comparison and mitigation. Based on this scrutiny this brief paper explores two questions:

- Are presupposed assumptions ethically justified on grounds of scientific evidence and ethical consistency?
- Should moral objections claiming to invalidate comparative risk assessments be accepted or rejected?

RADIATION RISK SELECTION: SCIENTIFIC EVIDENCE AND IRONY

Radiation exposures from man-made technologies--whether for medical diagnosis or electricity generation--have been singled out as a unique cause of the dread disease of our age, cancer, as well as of genetic mutations affecting distant future generations. An entire ethical framework for social criticism has been erected on the scaffolding of beliefs which are at odds with the actual status of scientific evidence.

1. Hypothetical Harm. More than 25 years have been spent in developing radiation protection philosophy and standards--years dominated by a conservative assumption: every radiation dose greater than zero entails some possibility of somatic/genetic harm. The linear, zero-threshold hypothesis has led the public to believe that "there is no safe dose of radiation" and "every radiation dose is an overdose." Ethical objections appear to stand or fall on this hypothesis.

Professional ethics compels us to recognize that, despite a vast array of radiobiological data, there is no conclusive evidence to prove the existence or absence of a threshold. Moral objections assume the linear hypothesis to be an unassailable scientific conclusion; but in fact it is only an inconclusive theory, an extrapolated hypothesis, an ultra-conservative and protective rule of prudence. Humans could not exist if the linear hypothesis were applied to and enforced upon personal lifestyle exposures to natural terrestrial and cosmic radiation.

The absence of evidence of harm from low level exposure is not due to incompetence or lack of attempts to find effects. L. Taylor is unambiguous: "No one has been identifiably injured by radiation while working within the first numerical standards set by the NCRP and then the ICRP in 1934. Let us stop arguing about the people who are being injured by exposure to radiation at the levels far below those where any effects can be found. The fact is, the effects are not found despite over forty years of trying to find them. The theories about people being injured have still not led to the demonstration of injury and, though considered as facts by some, must only be looked upon as figments of the imagination."(3)

A profound misunderstanding of the inconclusive scientific status of the linear hypothesis renders moral arguments dependent on it inherently flawed.

2. Hormesis. It has been scientifically established that there are net beneficial effects from exposure to low levels of toxic substances, e.g. copper, selenium, fluoride. Professional ethics should compel competent members of the scientific community to examine radiobiological data through the lens of an hypothesis counter to--yet equally worthy of attention given--a linear, zero-threshold hypothesis. T. D. Luckey (4) has presented persuasive evidence that exposure to low-level radiation might have net beneficial effects. Indeed it may be essential for the continued well-being of living organisms which have evolved in relation to wide variations in exposure to natural radiation.

According to J. N. Stannard (5), the guideline of ALARA in USNRC regulations has been interpreted to mean, "If you can do it technically, you must do it" without any regard to excessive cost or more effective health protection. An ALARA guideline unjustifiably assumes that any degree of reduction in radiation exposure will do some good. However, evidence suggests three possible hormetic outcomes: (1) increased growth and fertility of both plant or animal organisms, (2) increased longevity, and (3) a reduction in cancer frequency.

3. Etiology of Cancer. Thirty years ago, when John Higginson ascribed the incidence of cancer in industrialized societies to "environmental causes," he meant a total environment--i.e. agricultural practices, hygiene, diet and behavior, social mores--not physical chemicals.(6) Through

misinterpretation and distortion, his complex theory has led people to believe that some carcinogen lurks in everything humans eat, drink and breathe. John Totter has shown that mortality from cancer appears independent of industrialization in a country and of its man-made pollution. He maintains that one should look for primary carcinogens--not among man-made agents--but among all-pervasive "normal" environmental components. Totter suggests that the culprit is oxygen: it is a recognized mutagen; experiments have shown that it causes tumors in fruit flies; in the Ames assay test for screening carcinogens, it shows up positive.(7)

To summarize: no scientific nor ethical justification exists for singling out radiation as a unique cause of cancer or genetic mutations when over a thousand other toxic agents in common commercial use are capable of producing the same health effects if exposed to them in sufficient doses. There is both an ethical and scientific basis for recognizing a practical threshold or de minimis dose below which risks of exposure are trivial and ought legally and morally to be ignored. A hypothetical harm can entail only a hypothetical violation of rights, fairness, equal protection and intergenerational equity.

COMPARATIVE RISK ASSESSMENT: FALLACY vs. REALITY

To counteract moral objections that radiation risks pose utterly unique threats to health, at least five methods of comparative risk assessment have evolved: (1) comparisons among risks already widely accepted by the public with "new" risks; (2) comparisons among risks of alternative ways of achieving the same objective; (3) comparing technology-induced radiation exposures with natural background; (4) comparison of risks with benefits to be achieved; (5) comparative measures of the cost-effectiveness of various risk-reduction options.

Critics of comparative risk assessment object that each method "begs the question of why an increase in risk, however small, should be acceptable."(8) These methods are also attacked because they assume that radiation risks are not unique but commensurable with other risks, so that expenditures for risk reduction can achieve "equity" by a more economically efficient use of dollars-per-life-saved.(9)

The first objection commits the fallacy of misplaced concreteness. It is fradulent to isolate one technology in such a way that it only represents incremental risks--as if these were simple additions to a current risk background. To the contrary, any "new" risk in reality reorders an entire system by displacing, offsetting, substituting for, or otherwise restructuring a prior pattern of benefits and harms. Only systemic risk analysis which surveys a total spectrum of threats to health, and endeavors to compare risk-reduction options and associated costs, can be expected to achieve the most equitable health protection for an entire population.

The incommensurability objection claims that "efficient use" of risk-reduction dollars cannot achieve "equity" because risk is not distributed in a population equally, nor is "efficient life saving" the only goal of safety policy. The objection fails on two accounts: it confuses "efficiency" with effectiveness, and it misinterprets "equal protection" as absolute protection. It is an inescapable reality that a public policy cannot possibly "do no harm." Even the most conscientious of policies will entail risks of some unintended harm to some hypothetical individual. When we compare more or less cost-effective methods of reducing risks to people and the biosphere, and compare net benefits, we are not wielding utilitarian tools for placing a callous dollar-value on human life or impairment as a moral judgment of individual worth. Much less are we estimating economic losses to society as a measure of personal expendability sacrificed to achieve technological advancement or "benefits" to an abstract society. In the real world of decisions about setting priorities for allocating public money, we are maximizing the value we place on human life by endeavoring to reduce widespread harm, thereby preventing diminished quality of life and premature loss of life expectancy. These decisions are covenantal expressions of our common humanity.

In conclusion, a fundamental ethical imperative of fairness should prevent moral concerns for health protection from being trivialized by an obsession with hypothetical health effects from but one technology, thus siphoning public concern away from preventable causes of widespread disease, malnutrition, morbidity and death claiming thousands of lives daily, and endangering a basic set of conditions for the well-being of our posterity.

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THREE-DIMENSIONAL DOSE-RESPONSE MODELS OF RISK FOR RADIATION INJURY AND CARCINOGENESIS*

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ABSTRACT

The use of computer graphics in conjunction with three-dimensional models of dose-response relationships for chronic exposure to ionizing radiation dramaticly clarifies the separate and interactive roles of competing risks. The three dimensions are average dose rate, exposure time, and risk. As an example, the functionally injurious and carcinogenic responses after systemic uptake of Ra-226 by beagles, mice and people with consequent alpha particle irradiation of the bone are represented by three-dimensional dose-rate/time/response surfaces that demonstrate the contributions with the passage of time of the competing deleterious responses. These relationships are further evaluated by mathematical stripping with three-dimensional illustrations that graphically show the resultant separate contribution of each effect. Radiation bone injury predominates at high dose rates and bone cancer at intermediate dose rates. Low dose rates result in spontaneous deaths from natural aging, yielding a type of practical threshold for bone cancer induction. Risk assessment is benefited by the insights that become apparent with these three-dimensional models. The improved conceptualization afforded by them contributes to planning and evaluating epidemiological analyses and experimental studies.

METHODS

Mathematical relationships for risk distributions.

The bodily intake of radioactive materials, such as Ra-226, can lead to protracted, chronic irradiation of tissues, such as the bone, with subsequent cancer induction or systemic injury. Similar responses may be observed for repeated external exposure to penetrating radiation. The independent risk probability of fatal induced cancer is obscured by the separate independent competing risks of death, especially those that occur near the end of the normal life span. The observed occurrence of fatal cancer induced by exposure to ionizing radiation is the resultant of the convolution of all causes of death, and is a type of dependent risk since it depends upon both the dose-response relationships for the irradiation and as well upon other effects including especially deaths associated with natural aging.

Probability distribution functions are utilized to describe the various risk distributions as functions of time from beginning of exposure until death and as a function of dose rate at any specific time. These relationsnips have the general form that the independent risk distribution for a given effect has a probability density, $f(t,\bar{D})$, and cumulative risk function, $F(t,\bar{D})$, which are distributed with respect to elapsed time, t, but depend on average dose rate, $\bar{D}$, forming three-dimensional mathematical response surfaces. The cumulative risk for a single effect is the independent probability (values between zero and one) of an individual succumbing to the specified response (e.g., dying of bone cancer) assuming that there are no other possible effects. Likewise, the probability density,

*Research supported in part by the Office of Health and Environmental Research of the U.S. Department of Energy.

f, is the fraction of all individuals originally at risk who succumb per unit of time (e.g., per day) after exposure begins. Thus, if T is the time from initial exposure to death from a specified cause and there are no other causes of death, then the probability of dying at a given average dose rate before or at a specified time, t>T, is given by $F(t,\bar{D})$ with t=A-E and A the age of the individuals at risk and E their age at the beginning of exposure.

When several separate risk distributions, F_i, are superimposed in the time and dose-rate space, the occurrence of one cause of death, i, is a fraction, Ω_i, of individuals who succumb to that cause at a specific average dose rate. For example, effect i=1 could be spontaneous deaths associated with natural lifespan, effect i=2 could be deaths associated with a specific form of radiation-induced cancer, and effect i=3 could be deaths from systemic injury induced by radiation exposure (Raabe, 1987).

Independent distributions, F_i, are mathematically unchanged by the presence of the other risks, but the occurrence of the other risks may reduce the number of individuals that succumb to a specific cause. Thus, the actual occurrence of cancer deaths caused by radiation exposure will be less than if there were no other possible causes of death, and $\Omega_2 \leq F_2$. The occurrence fraction is the dependent risk from exposure to ionizing radiation since it is the resultant of the various causes of death including those associated with natural lifespan. The number of individuals among those exposed who succumb to an effect, i, is predicted by Ω_i, not F_i.

Lognormal independent risk model

The independent risks of cancer induction and other effects have been usefully modeled as a lognormal functions of time to effect (Raabe et al. 1980; Raabe et al., 1981a; Raabe, 1984; Raabe, 1987). The lognormal model involves a basic dose-rate/time/response relationship given by:

$$t = K\bar{D}^{-S} \qquad (1)$$

where $\bar{D}$ is the average dose rate to the tissue at risk, t is the elapsed time to death (or other endpoint) after initial exposure, K is a parameter associated with level of risk and exposure conditions, and S is the negative slope of the logarithmic form of the function. At any given $\bar{D}$, both K and t are lognormally distributed with geometric standard deviation, σ_g. This relationship defines a three-dimensional lognormal dose-rate/time/response surface for a specific effect such that:

$$Z = (\ln t - \ln K_m + S \ln \bar{D})/\ln \sigma_g \qquad (2)$$

where K_m is the fitted median risk value of K, σ_g is the observed geometric standard deviation of K values for the individual cases, and Z is the standardized normal deviate which is equal to zero at the median risk (F=0.5). Hence, the cumulative risk for a lognormal dose-rate/time/response distribution for a specific effect can be calculated for each $\bar{D}$ and t by numerical integration of the standardized normal distribution.

Data for ^{226}Ra-injected beagles.

An example of the use of the three-dimensional models is given in an analysis of bone cancer induction and skeletal injury in lifetime studies of Davis beagles injected with Ra-226 (Raabe et al., 1981b). Briefly, 234 purebred beagles were administered 8 fortnightly intravenous injections of Ra-226 in 0.1 N nitric acid saline solution in five dosage groups beginning at 435 days of age and ending at 540 days of age (midpoint of intake at 485

days of age). Lifetime dosimetry involved whole-body gamma ray spectroscopy of Bi-214 and application of the appropriate radon/radium retention ratios in bone; the irradiation of bone cells was primarily by alpha rays. To compare the Ra-226 dose-response results in beagles to other species including people, available human data (Evans, 1966, Evans, 1974, Argonne National Laboratory, 1985) and female CF_1 mouse data (Finkel et al., 1969) involving bone burdens were also evaluated with the three-dimensional models.

Life span distributions.

The distribution of deaths associated with natural lifespan involves many causes including various forms of both communicable and non-communicable diseases, and is commonly represented utilizing the Gompertz function. The Gompertzian cumulative risk from an age E until a time t=A-E is described by h_o, the hazard rate at birth, and ψ, the exponential coefficient (Raabe, 1987). For beagles, $h_o = 2.037 \times 10^{-6}$/day and $\psi = 1.104 \times 10^{-3}$/day.

RESULTS

In the Davis beagle study, a total of 115 cases of fatal bone cancer (primarily osteogenic sarcoma) were fit to the lognormal model by least squares for dose rates spanning 0.3 to 20 cGy/day. The resulting function had K_m=2500, S=-0.29 and σ_g of 1.17 based upon survival time in days and dose rate in cGy/day (Table 1). The time post intake was calculated from the midpoint of the injection period (A=485 days). An approximate (S=3) lognormal bone injury risk function was fit to the observed cases of deaths from systemic injury at high dose rates.

When the combined cumulative risk, $F_c(\bar{D},t)$, for all causes of death is divided into its component constituents using mathematical stripping as described by Raabe (1987), the separate fatal occurrences show that radiation bone injury predominates at high dose rates and bone cancer at intermediate dose rates. Also, low dose rates result in spontaneous deaths from natural aging, yielding a type of practical threshold for bone cancer induction.

To compare the Ra-226 dose-response results in beagles to other species including people, lognormal dose response relationships with identical S=0.29 were fit to available human data and female CF_1 mouse data involving skeletal burdens of Ra-226 (Raabe et al., 1980; Raabe et al., 1981a; Raabe et al., 1983). These results for the independent risk functions are summarized in Table 1. The three-dimensional occurrence distribution for people of fatal bone cancer induced by Ra-226 can be assumed to includes a concurrent risk of carcinoma of the head not found in beagles.

Table 1. Parameters of the lognormal dose-rate/time/risk distribution for observed fatal cancer in laboratory animals and man where S is the negative slope and σ_g is the geometric standard deviation of K values about the median K_m for n animals with correlation coefficient, r, based upon radiation dose rate in cGy/day and survival time in days.

Response	Radionuclide	Species	K_m	S	σ_g	n
Bone Cancer	^{226}Ra	Beagle	2,464	0.29	1.17	115
Bone Cancer	^{226}Ra	CF_1 Mouse	850	0.29	1.30	249
Bone Cancer	^{226}Ra	Man	9,000	0.29	1.39	32
Bone Injury	^{226}Ra	Beagle	10^7	3.0	Estimate	

The same basic three-dimensional lognormal dose-rate/time/response function with S=0.29 was found to describe the results for the three species but displaced in time by a species dependent response ratio (RR). This response ratio can be interpreted as the ratio of the respective median values, K_m, for the three species which were well correlated to (but not proportional to) life expectancy (Raabe et al., 1981a). The RR for people/beagles=3.6 and people/mice=10. Thus, people are one-tenth as sensitive as mice for osteogenic sarcoma at the same average dose rate.

DISCUSSION

The three-dimensional dose-rate/time/response relationships provide for an improved understanding of the interaction of competing risks and natural life span in combination with chronic exposure to ionizing radiation. In particular, the usual methods which utilize cumulative dose tend to obscure or ignore the effects of time and dose rate upon the risk.

It is clear the the convolution of induced cancer risk distributions from chronic exposures and natural life span leads to a steep, nonlinear occurrence function. A type of quasi-threshold describes the occurrence of radiation-induced cancer at low dose rates. Hence, the common use of linear risk functions to describe life span dependent risk will be expected to overestimate the actual occurrence of cancer at low dose rates.

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LIMITATION OF EXPOSURE TO UV IN COMPARISON WITH IONISING RADIATION: POLICES AND REGULATIONS

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INTRODUCTION

Physical characteristics and biological effects of ultraviolet (UV) and ionising radiation display similarities. The present paper endeavours to give an overview of facts and approaches relevant for protection against UV as compared with ionising radiation (1).

EXPOSURES AND RISKS

Exposure to UV radiation may cause erythema of the skin and inflammation of the eye (photokeratitis and photoconjunctivitis). These effects proceed within a few days after the exposure has exceeded a certain threshold value. There are indications that chronic exposure to UV radiation may induce cataract. Exposure of the skin contributes to the ageing of the skin and the risk of the occurrence of skin carcinoma. There is no indication for a threshold below which the skin cancer risk is not affected.
There is little principle difference with respect to ionising radiation, the exposure to which can cause both acute (non-stochastic) and delayed (mainly stochastic) effects. It should be remarked, however, that the dose-effect relationship for photocarcinogenesis in human populations is basically a quadratic function (1A). Therefore no elegant concept like that of the collective dose for ionising radiation exists.

Also, whereas the evidence for beneficial effects of ionising radiation is still under discussion, the positive effects of UV irradiation (vitamin D production, pigmentation) are more clearly defined.

It is of interest to compare the overall risks of exposure to UV and ionising radiation respectively. For simplicity, one can consider in case of UV the carcinogenic effect of skin exposure, which is for more than 90% due to the natural source, the sun. In reference (1B), the corresponding individual risk has been calculated for the Netherlands, on the basis of statistical data on skin cancer and melanoma mortality, to be in the range $(0,5 - 2,5)\ 10^{-5}$ per year. The range reflects the uncertainty about which fraction of melanoma mortality is due to UV exposure (1C). This situation is not grossly dissimilar from that for ionising radiation, where approximately 80% of exposure is due to natural sources. In a low natural background radiation country like the Netherlands, a corresponding individual risk can be calculated to be $2,5 \times 10^{-5}$ per year on the basis of the linear risk factor recommended by ICRP.

BASIC EXPOSURE SITUATIONS

Basic principles of dose limitation recommended by ICRP - justification, As Low As Reasonably Achievable (ALARA) principle, individual dose limits -

* Member, IRPA International Non-Ionising Radiation Committee

are not meant to be applied uniformly in all exposure situations. The individual dose limits for exposure to ionising radiation e.g. do not apply to medical and natural sources of radiation. Besides the three tradional situations - patients' exposure, exposure in the work-place, exposure of the members of the public - additional categories may be needed. They are summarised for UV and ionising radiation in Table I.

Table I

Basic situations of exposure to radiation		Ionising	UV
Patients exposure	preventive		+
(intentional)	diagnostic	+	+
	curative	+	+
Occupational	enchanced natural	+	+
exposure	planned/within limits	+	+
(unintentional)	unplanned/accidental	+	+
Public exposure:	cosmetic		+
(intentional)	enhanced natural		+
	enhanced natural	+	+
(unintentional)	planned/within limits	+	+
	unplanned/accidental	+	+
	large-scale emergencies	+	

Whereas there exists an overall analogy, some differences between UV and ionising radiation are striking. No preventive medical uses of ionising radiation are common whereas they do exist for UV, such as to achieve desensibilisation or vitamin D production. No sudden large-scale accident emergencies are known for UV sources in contrast to ionising radiation, where risk management, planning and preparedness for nuclear accidents form major ingredients of the policy and regulatory processes. On the other hand, slow changes of natural UV levels with potentially serious global deterioration of human and ecological environment may occur if the ozone layer is depleted due to certain forms of pollution.

Finally, intentional exposure for tanning purposes has become a characteristic application of UV radiation in many industrialised countries. This type of exposure is without analogon in the field of ionising radiation.

LIMITATION OF INTENTIONAL EXPOSURE

In North West European countries, a substantial fraction of the population have become regular users of tanning equipment (Sweden 4% (1D), the Netherlands 7% (1E)).

The effects of UV radiation are wavelength dependent. Using a spectral effectiveness function or action spectrum the radiant exposure (in J/m^2), or irradiance (in W/m^2) can be weighted per wavelength with a factor appropriate for the biological effect considered. Integration over the whole UV spectrum produces an effective radiant exposure, or effective irradiance.

Action spectra for skin erythema and for pigmentation, respectively, can be defined as a function of the skin type. They are similar but not identical. Representatives from two international organisations, IEC (International Electrotechnical Commission) and CIE (Commission Internationale d'Eclairage), have agreed at Amsterdam in 1987 to recommend and use the simplified action spectrum proposed by McKinlay and

Diffey (1F). So far, no such general agreement has been arrived at as far as the pigmentation is concerned. Also there is no agreement on an action spectrum on carcinogenesis although there are strong indications that the latter is well represented by the erythema action spectrum.

Since 1980 several countries (USA, Canada, Sweden, UK, FRG, Australia, the Netherlands) have promulgated regulations or recommendations concerning the exposure to tanning equipment, as summarised in (4). The general aim is to prevent acute, undesirable effects (and in some cases to limit the risk of chronic effects) without undue curtailment of the desired cosmetic effect. In most countries, the requirements concern a limitation of the irradiance (in W/m^2); a limitation of the initial and/or total radiant exposure (in J/m^2) during an irradiation course; the existence of devices such as a timer and an emergency switch; the use of eye protection; the information to be provided to the public, especially concerning the hypersensitive individual, the effects of drugs, cosmetics etc.

From the foregoing, it is clear that the total annual radiant exposure is not limited. It may be interesting to mention that the skin cancer incidence in the Dutch population has been calculated to increase by a few percent due to the estimated exposure from tanning equipment.

LIMITATION OF UNINTENTIONAL EXPOSURE

Unintentional exposure from artificial UV sources can occur both in the work-place (research laboratories, curing of ink and lacquers) and for the members of the public (illumination, amusement industry). American Conference of Governmental Industrial Hygienists (ACGIH) was the first body to formulate corresponding threshold limit values (2). The underlying principles and limiting values have been further developed by the IRPA International Non-Ionising Radiation Committee (INIRC) (3) and by the Health Council of the Netherlands (4). Both latter documents require the ALARA principle to be applied. The purpose of the IRPA/INIRC guidelines is to provide limits of exposure to UV and represent conditions under which it is expected that nearly all individuals may be repeatedly exposed without adverse effects. The Dutch Health Council states more explicitely that the exposure limits recommendations are based on two principles:
- harmful effects for which there is a threshold must be avoided
- the risk of chronic effects for which there is no threshold dose, is to be restricted to a reasonable value.

The exposure limits apply to exposure during a normal working day, and are adequate to protect lightly pigmented individuals, both in the working and general population. They do not apply to lasers; exposure duration less than 0,1 microseconds; some rare, highly photosensitive individuals; individuals concomitantly exposed to photosensitising agents; and to aphakic individuals (persons with a lens removed).

The exposure limit has a minimum value of 30 J/m^2 at 270 nm. In the wavelength region up to 310 nm, it is based upon combined effects on the skin and the eye, and is equal in all three documents. At higher wavelengths the exposure limit of the Health Council of the Netherlands has been derived from the erythema action spectrum. In the 310-400 nm region ACGIH and INIRC/IRPA propose to limit the irradiance to 10 W/m^2 for exposure durations longer than 1000 s. For shorter exposures the radiant exposure is limited to 10 kJ/m^2. These recommendations are related to the possible induction of the cataract. The Dutch Health Council concluded, however, that an exposure limit related to the cataract is only necessary in case of chronic exposures and recommends a value of 1 W/m^2.

It has been estimated (4) that a radiant exposure equal to the proposed exposure limit corresponds, for average exposure conditions, to skin cancer risk increase of 25% for an average indoor worker. The skin cancer risk of an outdoor worker is a factor 5 larger than that of an indoor worker. The risk of skin cancer induction in case of chronic exposure to UV radiation approximating the exposure limit appears to be much less than the extra risk caused by working outdoors.

CONCLUSIONS

Both UV and ionising radiation may cause acute and late effects. The average mortality risk due to overall exposure is similar in both cases. Several basic situations of exposure are comparable; but UV exposure is characterised by (a) intentional use for cosmetic purposes, and (b)lack of sudden large-scale emergencies. The limitation of unintentional exposure to UV and ionising radiation follows similar principles of protection. It would be desirable to deepen further the degree of uniformity and harmonisation in protection against these two types of radiation.

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ESTIMATION OF POPULATION DOSE FROM ALL SOURCES IN JAPAN

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1. INTRODUCTION

The purposes of estimation of population doses are to understand the per-caput doses of the public member from each artificial radiation source and to determine the proportion contributedof the doses from each individual source to the total irradiated population. We divided the population doses into two categories: individual-related and source-related population doses. The individual-related population dose is estimated based on the maximum assumption for use in allocation of the dose limits for members of the public. The source-related population dose is estimated both to justify the sources and practices and to optimize radiation protection. The source-related population dose, therefore, should be estimated as realistically as possible. We investigated all sources that caused exposure to the population in Japan from the above points of view.

2. MATERIALS AND METHODS

(1) Investigated sources

We investigated all sources contributing to exposure to the population in Japan. The sources surveyed in this report are shown in Table 1.

Table 1 Surveyed natural and man-made sources

- natural sources
  - cosmic rays
  - terrestrial radiation
  - internal irradiation
  - radon and thoron and their decay products
- man-made sources
  - nuclear power production
  - nuclear explosions
  - consumer products
  - technologically modified natural radiation
  - medical radiation

(2) Population dose

We estimated two population doses. One was the per-caput dose from each source of both natural and man-made radiations. The other was the maximum individual dose from all artificial sources. The per-caput doses were calculated realistically and the individual dose for each critical group was calculated with maximum assumption. In calculating the doses, the effective dose equivalent proposed by ICRP was used.

(3) Data sources

We used available information reported in some publications by the authors or other researchers.

In the case of no information regarding Japan, particularly indoor radon concentration, we used other countries' data published in UNSCEAR reports and elsewhere.

3. RESULTS

(1) Per-caput effective dose equivalent

An average annual effective dose equivalent for one person from all sources is shown in Fig. 1. Note that the doses from man-made radiation sources amount to about 50% of all population doses. The doses from both man-made radiation and from natural radiation each contribute almost the same to the population doses in Japan. It was also found that the doses from medical radiation constitute 98% of the doses from man-made radiation.

The annual per-caput effective dose equivalents from natural background radiation in Japan were 270 μ Sv from cosmic radiation, 300 μ Sv from terrestrial radiation, 240 μ Sv from internal radiation and 840 μ Sv from the inhalation of radon, thoron and their products.

The per-caput effective dose equivalents from radiological diagnosis and therapies are shown in Fig. 2. The doses from X-ray radiographies performed in the hospital occupy the larger part of the doses from medical radiation. The doses from mass health examination, that is chest and upper abdomen photofluorographies, follow the doses from the X-ray examination. The types and frequencies per person of X-ray examination are shown in Fig. 3. On the average, one person receives one X-ray examination once a year.

Dental radiographies are performed frequently in Japan, but their effective dose equivalent is very small. Consequently, dental X-ray examinations do not contribute at all to population dose in view of the effective dose equivalent. On the other hand, since the effective dose equivalent of upper abdomen X-ray examination is higher, its contribution to the population dose is large.

The average annual effective dose equivalent from man-made sources other than medical uses, that is fall-out radiation, technologically modified natural radiation, and discharge radiation from nuclear facilities, is only about 30 μ Sv and contributes 0.8% of the population dose.

In Japan, 14 nuclear power stations and 35 nuclear reactors are now operated. The operational levels on the discharge of radioactive nuclides from nuclear power stations is set by national authority at 50 μ Sv/y for members of the public. The actually received doses are controlled at below 10 μ Sv/y.

(2) Maximum effective dose equivalent from artificial radiation sources

The maximum effective dose equivalents from man-made sources are shown in Table 2. These doses were estimated to be maximums with intention of use for allocation of dose limits for member of the public, that is 1 mSv/y.

Table 2 Maximum effective dose equivalent from man-made sources

source	dose (μ Sv)
nuclear power plants	50
fuel reprocessing facilities	100
consumer products	10
coal power plants	50
phosphate fertilizers	20
high altitude flights	110

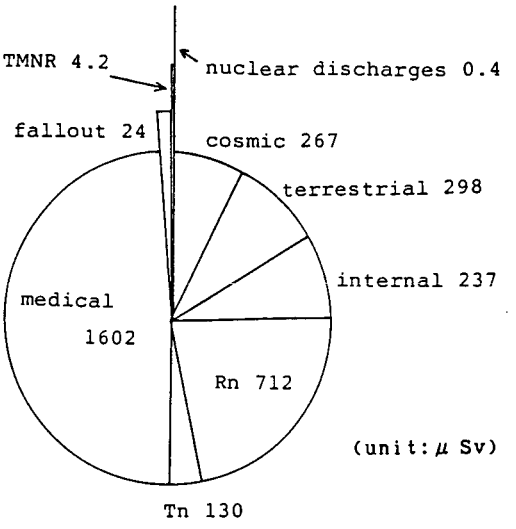

Fig.1 Average annual effective dose equivalent to one person from all sources

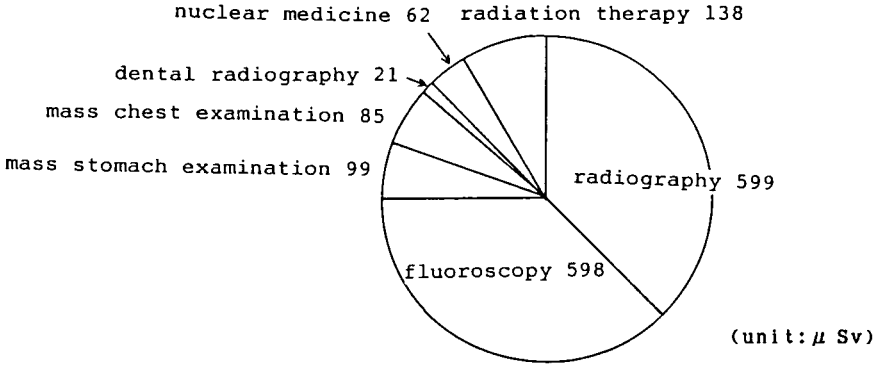

Fig.2 Per-caput effective dose equivalents from radiation diagnosis and therapy

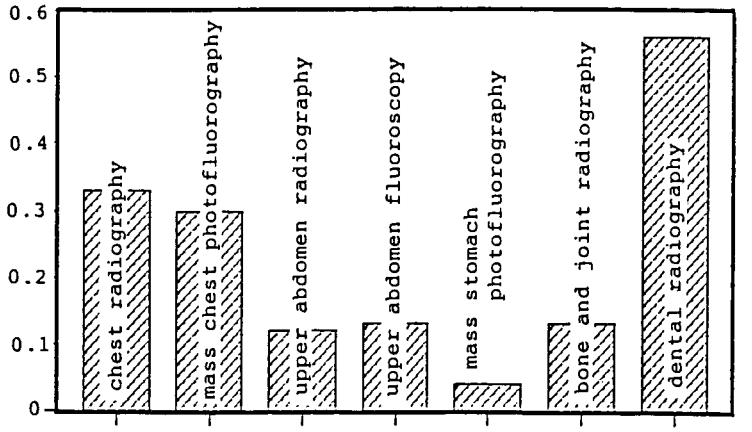

Fig.3 Types and annual frequencies per person of X-ray examination.

4. DISCUSSION

The per-caput effective dose equivalent from natural radiation in Japan, 2 mSv/y, is approximately similar to other countries located at the same latitude and height. The proportion of natural background radiation to all population dose, about 50%, was different from other highly developed countries, in which the dose from man-made radiation sources were lower than in Japan. The doses from medical uses in Japan are higher than those in other highly developed countries. For example, the doses from medical radiation in Japan are six times higher than those in England. Radiological examination such as X-ray diagnosis, mass health examination, dental X-ray and nuclear medicine is carried out frequently in Japan. In Japan, legal periodical chest examination is enforced for all of the population and periodical upper abdomen X-ray examination is carried out frequently in persons more than 40 years of age. The diagnosis X-ray and nuclear medicine are daily used for many patients in hospitals. The practitioners and radiologists should judge the application of radiography, fluoroscopy or radiation therapy for patients from the viewpoints of justification of the practices. Our survey in Japan showed that despite frequent uses of medical radiation, most practitioners and radiologists had little or insufficient knowledge of radiation protection and radiation risk and detriment. Our urgent problem is to promote the spread among practitioners of knowledge of radiation protection for patients. Another problem is to obtain further information on indoor radon concentration. There is little such information in Japan. Further investigation is needed to estimate exposure to indoor radon. In Japan, some systematic survey of indoor radon concentration is being done by the National Institute of Radiological Science (NIRS) and other societies. More information will be expected in a few years.

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FOOD IRRADIATION ISSUES, TECHNICAL AND PUBLIC, IN THE UNITED STATES

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Food and Drug Administration

In the past few years in the United States there have been a few controversies relative to our food supply. A certain segment of our population has always been concerned about food additives and the use of pesticides, the allegation being that those chemicals are "unnatural" and can lead to deleterious effects among the consuming population. The issue of pesticide use reached a peak several years ago with the news that eythylene dibromide (EDB), commonly used to disinfest grain, could have long-term carcinogenic effects. The food and agricultural industries are searching for safer chemicals or other techniques to ensure a safe and varied food supply. One technique being studied is Food Irradiation.

"Food Irradiation" is defined, for the purposes of this presentation, as the use of ionizing radiation in food processing.

Current proposals and uses for food irradiation limit the sources to certain radioactive isotopes (cobalt-60, cesium-137) which are sufficiently long-lived and emit penetrating radiation for practical use, and to machines (x-ray, electron beam) which can produce sufficient penetrating radiation with rather simple technology. There are advantages and disadvantages of each source. Radioactive isotopes need to be shielded when not in use; machines produce the radiation only upon electrical stimulation but are more complex technically. Isotope sources have limited useful lives because of radioactive decay.

In a typical food irradiation facility, containers of food are transferred automatically into the radiation field produced by the source. The food is irradiated for a particular time dependent upon the strength of the source. The irradiated food is then automatically transferred outside the radiation field for shipment. Shielded walls would be necessary around the radiation area to protect the public and workers. If the source is a radioactive isotope, a shielded safe would be necessary for storage of the source while it was not being used for irradiation.

There are quite a few reasons for which food irradiation would be useful in processing. In order of increasing radiation dose, food irradiation would be useful for:

Inhibition of sprout formation, and thus increase the shelf-life of sprouting vegetables at 50-150 Gy;

Insect disinfestation at 200-800 Gy;

Elimination of spoilage organisms at 1,000 to 3,000 Gy;

Elimination of pathogenic and parasitic organisms for which 3,000 to 8,000 Gy is necessary;

Food sterilization at 25,000 to 50,000 Gy.

A problem can occur in that spoilage organisms are eliminated at levels lower than that necessary to eliminate pathogenic organisms. The natural taste-smell test for suitability for food may thus become unreliable. If the spoilage organisms are eliminated but pathogenic organisms are allowed to proliferate, organoleptic tests for freshness would be invalid.

For increase in shelf-life, irradiation is suitable for some foods but not others. Papayas, mushrooms, onions, and shrimp can have their shelf-lives extended because of the retardation of evident aging processes. The amount of radiation to be used for each food must be determined empirically. Some species of cherries, for example, can be shelf-life extended, whereas others would be degraded by the irradiation. Irradiation conditions must be determined for each specific food item.

Food irradiation, like anything else, is not a panacea. There are problems associated with irradiation in that food quality may be affected at higher doses, the hygienic quality of the food must be controlled prior to irradiation, and reirradiation may lead to organoleptic deterioration of the food product. As with any other type of irradiation, the effects of food irradiation are cumulative with dose; therefore, food would need to be labeled that it had been irradiated so that a future processor does not reirradiate the food causing deterioration.

Historically, the idea of food irradiation arose in the 1940's when the U.S. Army experimented with the irradiation of food for field use. At that time, many of the techniques currently in use had not been developed resulting in the food's having the famous "wet dog" taste. By selective irradiation techniques of particular foods many of those early problems have been eliminated. Current techniques can include cryogenic temperatures during irradiation, for example.

In 1963 the Food and Drug Administration approved the use of food irradiation to control insect infestation in wheat.

In 1964 the Food and Agricultural Organization of the United Nations issued recommendations concerning food irradiation that included the following:

(1) Legislation concerning food irradiation must be promulgated;
(2) The safety of the food irradiation must be cleared;

(3) Specific foods must be cleared individually;
(4) Compliance must be accomplished using chemical testing, licensing, biological testing, labeling, dosimetry, and record keeping.

In 1980 an FDA committee concluded that animal feeding experiments are not necessary for foods at less than 1000 Gy. That conclusion resulted from chemical analyses of foods which had been irradiated compared to those which had not been irradiated and consideration of the levels of radiolytic products produced versus the amounts that would be necessary for practical animal experimentation.

In 1983 the FDA approved the use of irradiation to control microorganisms and insects in spices. The consideration here included the fact that spices not only are susceptible to microorganism and insect infestation, but also are a relatively small portion of the diet.

In 1985 the FDA approved the use of up to 1000 Gy to control trichinosis in pork. I understand that we are perhaps the only developed nation which has a pork trichinosis problem, and, therefore, our pork is not suitable for export to most other countries in the world.

In April, 1986, the FDA permitted further use of irradiation to inhibit the growth and maturation of fresh food and to disinfect foods adulterated with insects. All foods that are irradiated must be labeled to show this fact, both at the wholesale and at the retail levels. Thankfully, a previous recommendation to use the term "picowaved" has fallen by the wayside. Labeling needs to contain a statement concerning the radiation treatment and bear a symbol which, as the public becomes accustomed to it, may be all that would be required for labeling in the future.

There have been quite a few myths concerning the food irradiation process, and we should discuss some of those myths. First of all, of course, the use of the sources proposed will not make the food radioactive.

Since irradiation causes chemical changes in foods, concern has been expressed that there would be deleterious nutritional deficiencies caused in the irradiated items. It is true that irradiation does result in chemical changes in the foods, and some of the vitamins can be affected. However, those nutritional deficiencies are very small compared with nutritional deficiencies induced by other methods of food processing, such as cooking, or even by storage of the food. Some foods which showed a large decrease in certain vitamins are really minor sources of those vitamins in the American diet.

Some critics have emphasized that irradiation can produce new chemicals in foods. There are radiolytic products induced in food by irradiation. The major radiolytic products are already present in part per million quantities in foods with no apparent harmful effects to the consumer. Unique radiolytic products, that is, those produced only by irradiation and not by any other food processing techniques, are chemically similar to substances already found in food and are of such very small quantities (much less than parts per billion) that they cannot be considered deleterious to the consumer.

Another criticism focuses on the possibility that reirradiation of the same food at various stages in processing can increase the concentrations of radiolytic products. It is true that as dose increases the concentration of radiolytic products in the food also increases. However, because irradiation can also degrade the radiolytic products, the concentration of those products reaches a plateau at approximately 10 kGy, after which further irradiation eliminates as many of the radiolytic products previously formed as causes further radiolytic product formation.

Others emphasize the hazards to workers and public with the use of such a dangerous modality as intense sources of ionizing radiation. Of course, standard health physics techniques are necessary for safe operation of the facility.

A last myth, which has recently surfaced, is that food irradiation is being promulgated simply as a way to use the radioactive wastes that have been produced as byproducts of nuclear power and weapons production. In accordance with this myth, we would become so dependent upon the radioactive isotopes for food irradiation that we would continue nuclear power and defense weapons production simply to obtain the isotopes for this modality. In the United States, this myth is given credence by the fact that the Department of Energy is funding much of the food irradiation research; the fact that DOE must do so under Congressional edict is not considered. The argument, of course, does not address accelerator sources which are proposed for future facilities.

In summary, food irradiation cannot be a panacea to solve all food processing problems. It can, however, be used to save a large percentage of the world's food supply which is lost to pest infestation; it can increase the shelf-lives of many foods, thus increasing the variety of foods available to a larger population; it can eliminate much of the use of carcinogenic insecticides in our food supply; it can reduce the salmonella and trichinosis problems in meat items.

IMPROVING RADIOLOGICAL SAFETY THROUGH ORGANIZATIONAL DESIGN

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ABSTRACT

A pilot study of the perceptions of radiation hazards among American hospital personnel indicates that members of various occupational categories exhibit diverse behaviors toward X-rays, radiopharmaceuticals, and radioactive waste. These results are summarized in relation to other studies of radiation-hazard perception in Britain and the U.S. The implications for risk management and institutional design are brought together in a systematic model that relates specific characteristics of various medical occupations to different levels of vigilance and concern for potential radiation exposure to personnel and to patients.

The key variables determining cavalier, complacent, paranoid and vigilant responses to potential hazards are found to be the extent of personal control and the degree of integration of the respondent into the institution. A means of determining and adjusting an appropriate balance between individual control and institutional procedures is suggested for each of a range of occupation types that use or encounter ionizing radiation in the workplace.

RISK OF OCCUPATIONAL INJURIES IN THE INDUSTRY OF JAPAN

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Introduction

The ICRP Committee stated in their publications[1),2),3)] that it is important to compare the total harm that may be caused by the radiation with the total harm involved in other occupations, with respect to fatal or minor injury, occupational disease, or the effects of mutagenesis in the working environment. Death has commonly been used as an index of the comparative safety or harm of different industries, and the frequency of death attributable to occupational causes already has a certain validity. In this way, assessment was made on the yearly trend of change in the fatality rates due to occupational work in seven categories of industry in Japan during the period from 1975 to 1984[4),5)]. Furthermore, the frequency of injuries of defined severity in occupational hazards, fatality rates of accidents and diseases due to occupational work, and also commuting accidents to and from work were examined.

Methods

The data from Statistics [6)] and Undertaking[7)] Annual Reports of Workmen's Accident Compensation Insurance Council published by Ministry of Labor, and Population Census[8)] by Statistics Bureau, Prime Minister's Office, were used in this study. Industry is classified into eight categories on the basis of the Standard Industrial Classification for Japan: Forestry, Fisheries, Mining, Construction, manufacturing, Transportation and communication, Electricity, gas, water, steam and hot water supply, and Others. However, in the report , category of "Others" was omitted from the risk estimation, because of its wide-spread coverage of various occupations. The base number is the total number of workers which correspond to the number of workers covered by the Workmen's Accident Compensation Insurance. All risk rates are shown as the number of claims per million person years worked, unless otherwise stated. In order to estimate the risk rate for each industry, the number of claims is divided by the estimated population in each industry.

Results and Remarks

(1) Risks of fatal and non-fatal occupational injuries

Occupational deaths in industrial workers showed a rather wide distribution ranging from 3,000-5,000 per million workers per year in Mining to tens per million workers per year in Electricity etc. (Fig. 1). In Japan, Mining belongs to the highest risk industry, fatal occupational risk being 4,196 per million workers per year, while the lowest one is 88 per million workers per year of Electricity etc.

Figure 1. Trend of fatality rates per million workers per year in seven industries (1975-1984)

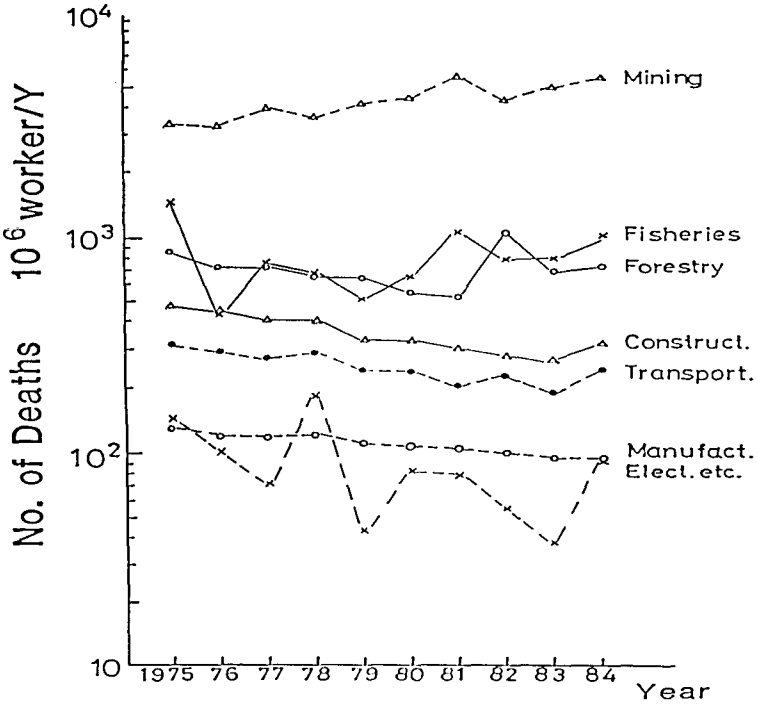

Table 1. Mean rates of fatal and non-fatal injuries per million workers per year in seven industries (1975-1984)

Industry	Total injuries	Fatal injuries	Non-fatal injuries: permanent total disability	Non-fatal injuries: Permanent partial disability	Non-fatal injuries: one or more days leave from work
Forestry	80,209	716	379	4,202	74,912
(%)	100	0.9	0.5	5.2	93.4
Fisheries	80,846	803	126	3,189	76,728
(%)	100	1.0	0.2	3.9	94.9
Minig	166,789	4,196	7,777	10,226	144,590
(%)	100	2.5	4.6	6.1	86.8
Construction	45,000	351	254	3,241	41,154
(%)	100	0.8	0.5	7.2	91.5
Manufacturing	44,776	111	82	2,601	41,982
(%)	100	0.2	0.1	5.8	93.8
Transportation	41,247	250	97	2,150	38,750
(%)	100	0.6	0.2	5.2	94.0
Electricity etc.	5,591	88	22	221	5,260
(%)	100	1.6	0.4	4.0	94.0
Mean of seven industries	45,289	222	179	2,766	42,122
(%)	100	0.5	0.4	6.1	93.0

As pointed out in ICRP Publications, it is also important to study the frequency with which harmful effects of different kind and severity occur in different occupational contexts. Risks expressed as the word "occupational injury" can be classified into two groups, one representing fatal risks and the other non-fatal risks, including permanent as well as temporary disability. Compensation or pension will be paid on the basis of the Workmen's Accident Compensation Insurance Law to people according to the categories of severity. Table 1 shows the risks of injured person per million workers per year in each industry. It is shown in the Table that the fatal injuries were less than one per cent comparing with various types of non-fatal injuries in all industries except Mining, and that more than 90 per cent of injuries was one or more days' leave from work. The total risks with fatal and non-fatal injuries are distributed from 1.7×10^{-1} in Mining to 5.6×10^{-3} in Electricity etc., the average risk of seven industries being 4.5×10^{-2}. It was also shown in the Table that percentage of permanent total disability in each category of industry is close to that of fatal injury, respectively.

(2) The ratio of accident and disease in occupational deaths

Above data on the death number of workers in each industry include the deaths caused by accidents and occupational diseases. Table 2 indicates the deaths of accidents and of diseases separately in the total occupational deaths. The numbers of accidental deaths in occupational in all seven industries were greater than those of occupational diseases. That is, these ratios were below one tenth except for Mining and Electricity etc. where it is about one fifth. The main causations of death in occupational diseases are the injuries and the pneumoconiosis due to inhalation of metallic or mineral particles.

Table 2. Occupational deaths (1974-1984)

	Deaths(%)		10^6workers/Y		
	Accident	Disease	Total	Accident	Disease
Forestory	95.7	4.3	522	500	22
Fisheries	90.1	9.9	614	553	61
Mining	79.5	20.5	1345	1069	276
Construction	94.1	5.9	215	202	13
Manufacturing	88.7	11.3	49	43	6
Transportation	94.2	5.8	132	124	8
Electricity etc.	81.5	18.5	29	24	5
Average	91.8	8.2	118	108	10

(3) Risk at work and risk in transit to and from work

Usually, the statistical data on Workmen's Accident Compensation Insurance Council in Japan do not include the commuting accidents. In ICRP Publication 45, the ratio of death in transit to and from work in various countries was compared. As shown in Table 3, the ratio of death in transit was only around ten per cent of occupational death. Comparing with other industrial countries, this ratio is very low. It might be due to the utilization of low-risk public transportations such as train or bus instead of motor car.

Table 3. Ratio of traveling and occupational deaths (1975-1984)

Industry	10^6workers/Y		T/W
	Work	Transit	
Forestory	594	20	0.03
Fisheries	709	23	0.03
Mining	1842	28	0.02
Construction	290	13	0.04
Manufacturing	67	14	0.21
Transportation	185	15	0.08
Electricity etc.	55	11	0.20
Average	156	14	0.09

Comparison was made on these results with those of other countries which were shown in ICRP Publications 27 and 45 in terms of number of occupational deaths per million workers per year, specifically, with the values in U.S.A. for the years of 1969 and 1972 as given in ICRP Publication 27 and with those in Canada for the years from 1975 to 1981 as given in ICRP Publication 45, respectively. Fatality rates from occupational injuries in various industries in Japan are within the same order of levels which were observed in other countries as cited in ICRP reports.

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FOOD IRRADIATION - A FRESH CASE OF RADIATION PHOBIA ?

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INTRODUCTION

From the mid-1950's until the present day any activity with words such as atomic, nuclear or radiation has caused great concern in many members of the public. These concerns arose, quite properly, during the 1950's because of the levels of radioactive contamination produced by atmospheric nuclear weapons testing. The concerns have continued to be fueled by events such as the 1957 Windscale Reactor Fire, the three Mile Island accident of 1979, Chernobyl, as well as various other issues, of which the most potent is the potential impact of a nuclear war. Because of these public fears , often encouraged by various groups for their own normally laudable, ends, the potentially beneficial uses of radiation are often confused with those that are distinctly dangerous.

Such is the case with the proposal to irradiate various foodstuff to either extend their shelf-life or to control or kill pathogens and insects after harvesting.

Two arguments are being used by the opponents of food irradiation. One is that the process is hazardous to both plant operators and members of the public who live nearby. The second is that the irradiation process harms the eventual consumers of the food from either induced radiolytic products or substantially reduced nutritional loss and vitamin loss.

This paper argues that whether or not the second point is valid, the process itself is inherently safe and does not present any untoward radiological hazard.

FOOD IRRADIATION PROCEDURES

Preserving food by ionising radiation - either gamma rays from radionuclides, usually cobalt-60 or caesium-137) or machine generated Xrays or electron beams - works by killing microorganisms and insects in or on the food. Radiation can also delay the ripening of produce by modifying the metabolic processes of maturing fresh fruits and vegtables.

The most common form of irradiation plant is the static sterlizer housing the irradiation source in a pool of water from which it can be raised to irradiate products passing on some type of conveyor system. The lot being contained within a suitably thick concrete cell. One such plant using cobalt-60 operates in the State of Victoria and is used for the irradiation of a variety of non-food items such as syringes, gloves, bee hives etc.

The Victorian plant consists essentially of a 7 meter deep pool of water in which sits a metal frame containing 6 modules each capable of holding up to 42 Co-60 pencils. The frame is raised out of the water into the irradiation position. The controls for raising and lowering the sources and operating the product conveyor are all located outside the 1.6m thick concrete cell which houses the irradiation facility. The walls provide adequate shielding when the source is in the exposed position. The various holes for access etc. were monitored and the highest radiation levels detected were:

(a) at the main exit point - 5 μGyh^{-1}

(b) at the product exit point of the conveyor range - 9-10 μGyh^{-1}

This area is fenced off, the dose rate at the fence gate was about background.

The total amount of radioactive material present in the Dandenong plant at full loading is approximately 37 petabequerel (i.e. 10^6 Ci)).

The irradiation room is ventilated at the rate of 20 air changes per hour to avoid excessive ozone build up. The pool water is circulated continuously through a mixed resin bed demineraliser and activated charcoal filter.

RADIATION SAFETY

Radiation safety was considered under two headings - the safety of plant personnel, and the safety of the general public, particularly those living near the plant. Both routine operations and potential emergencies were assessed.

For plant personnel it was thought that there were two possible sources of exposure: external irradiation through inadequate shielding, emergence of a pencil from within the shielded area, inadverent exposure within the cell, possible exposures during loading and unloading procedures and possible high gamma dose rates at the ion-exchange column caused by a leaking capsule contaminating the water and internal irradiation from leaking cobalt contaminating the pond, then drinking water leading to unsuspected ingestion.

EXTERNAL IRRADIATION

As noted above there are no radiation leaks around the concrete cell and the maximum dose rate during routine work is 9-10 μGyh^{-1} at the exit point of the conveyor range - which is fenced off.

An even better indicator of radiation control procedures are the monthly and yearly exposure records of staff working in the area. Monthly doses are generally zero with occasional exposures of between 10 and 70 μSv.

During loading and unloading, plant personnel assist AECL staff, and the maximum dose received during such operations has been 180 μSv. The highest yearly dose recorded has been about 400 μSv.

Emergency situations considered were;

(a) Inadvertent Exposure Within the Cell

The start up procedure requires that a safety key switch is activated in the irradiation room using the same key that activates the control consule. the key is attached to a radiation monitor. The procedures are that the operator has to enter the irradiation room, check that no-one is there, activate the safety key switch, leave the irradiation room at walking pace, lock the access door, and within 45 seconds activate the control console switch. It is considered that the procedure is completely adequate for preventing anybody inadvertently being left in the cell.

In the reverse procedure as noted above the access door cannot be unlocked except by the control console key by which can only be removed when the source is in the shielded position. The door itself has further safeguards, there is also the cable stretched across the access to the maze, and a radiation monitor is attached to the key.

Because of the various interlocks it was considered that the possibilities of anybody being in the cell with the sources in the exposed position to be nil, even though at least two such events have apparently occured overseas.

(b) Inadvertent Exposure Outside the Cell

The daily monitoring of the ion-exchange column, now changed to continuous monitoring, provides an early warning of any cobalt leakage and exposure to staff should be negligable.

Which leaves the possibility of a pencil emerging from the shielded area. There are three defence mechanisms. Firstly, the source pencils are slotted into channels at the top and bottom of one of the six modules. When full, the hinged end of the module is closed holding the pencils firmly in place. These modules in turn are held in the rack by sliding them into vertical channels at each end of the module. A cover is fitted to the conveyor structure such that should a pencil be dislodged from a frame it would fall to the bottom of the pond and not onto a product box. Thirdly, a gamma monitor, now duplicated, is installed in the product exit maze. This monitor sounds an alarm and shuts the plant down should the dose rate rise above a preset level.

INTERNAL IRRADIATION

The combination of the possibility of the cobalt-60 metal, the ion-exchange column monitor, and the fact that the rods are wipe-tested when being installed makes the likelihood of any contamination of drinking water negligable.

MEMBERS OF THE PUBLIC

It was thought that there were four possible sources of exposure to members of the public, other than visitors.

They are: external radiation during routine operation of the plant; loss of a pencil from the plant environs;

external exposure during transportation of sources to and from the plant;

contamination of water supplies leading to ingestion of cobalt-60.

The radiation levels at the perimeter of the plant are indistinguishable from background and do not change whether the source is in the exposed position or not. Thus members of the public do not receive a radiation exposure during normal operation of the plant.

All transportation is carried out using flasks designed to IAEA standards. These flasks are designed to withstand accidents of much greater magnitude than any that can be conceived as happening between the arrival port and the plant. The dose rates on the outside of the flask range from 50 to 400 μGyh^{-1}, well below the allowable limit. Consignments are accompanied by a radiation safety officer who travels in a separate vehicle. Thus in the event of accident causing the truck to be stopped members of the public could and would be kept away from any potential exposure zone.

ACCIDENTS

A comprehensive review of accidents at sterilization plants has not been undertaken. However the data that have been found suggests that the few fatalities that have occurred appear to have resulted from a failure to follow set procedures coupled with a component failure.

Typical was the fatality at the Norwegian Institute of Engergy Technology. A microswitch failed giving a source shielded signal releasing the barring of the door lock even though the positional display showed the source in an elevated portion. Comparison of the two signals would have shown the discrepency. There was not however a positive failure signal.

In addition the radiation monitor in the interlock system had been taken out of service for maintenance and the radiation dose/interlock system was out of action. Thirdly, the technician failed to use a monitor to check the radiation level before entering the irradiation room.

The prevention of any of these three mistakes would have avoided the fatality.

CONCLUSION

Radiation phobia, although not necessarily an identifiable disease, does exists in many countries. Professional radiation protection personnel have a responsibility to continue the process of public education so that there is a better understanding of the hazards, real and otherwise, of the uses of radiation.

One such current public issue is whether the irradiation of food is hazardous or not.

It is not appropriate to argue from the particular to the general but what can be stated is that the gamma sterilization in Victoria plant does not pose a hazard to plant personnel or public. Neither should plants like it, operated in a similar manner under a similar restrictive and policed regulatory regime, cause radiation problems.

Wherever debates and inquiries are conducted on the suitability or otherwise of radiation preservation of food the safety of the process should not be a significant factor.

12 January, 1988
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DIFFICULTIES IN USING THE OBJECTIVE HEALTH DETRIMENT AS AN INDICATOR OF THE RADIATION HARM IN A POPULATION

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SUMMARY

Owing to the probabilistic relationship between dose and stochastic effects some problems could arise in using the concept of objective health detriment for low probability exposures. The usefulness of other statistical indicators of occurrence of health effects in the exposed population is also discussed.

1. The detriment is a source-related concept, useful to describe by a single quantity the total impact of a radiation source over a group of **N** individuals: it has been defined as the mathematical expectation of harm in the exposed group. It is proportional to the number of exposed individuals, to the probability of each individual suffering the effects and to the severity of the effects.

2. In evaluating the detriment, which in principle accounts for all deleterius effects due to exposure to ionizing radiation, the stochastic effects can be sorted out and grouped togheter in a quantity G_H, the Objective Health Detriment. For planning the protection, it is usually assumed, more on practical and administrative grounds that on radiobiological evidence, that when individual doses are well below dose limits the incremental probability of suffering stochastic effects is directely proportional to the dose increment, with a proportionality factor $R^o = 0.0165/Sv$ for the effective dose equivalent. The sum of the increments of individual doses in the exposed group is thus proportional to the increment in the Objective Health Detriment therein, i.e. $G_H = R.N.H$, where **H** is the per-caput incremental dose. Introducing the incremental collective dose **S** in the exposed group, we can write $G_H = R.S$.

3. The increment in the collective dose could be considered a "measure" of the increment of G_H only if there were a sufficient knowledge of the risk factors: all the uncertaintes in selecting **R** are linearly reflected on G_H. Also the uncertaintes in evaluating **S** are linearly reflected in G_H: the quality of the models used for estimating exposures is crucial. That shall be taken into account in all decisional procedures for optimizing the protection by ranking the different protective options on the basis of the associated collective dose or objective health detriment. In Cost-Benefit-Analysis

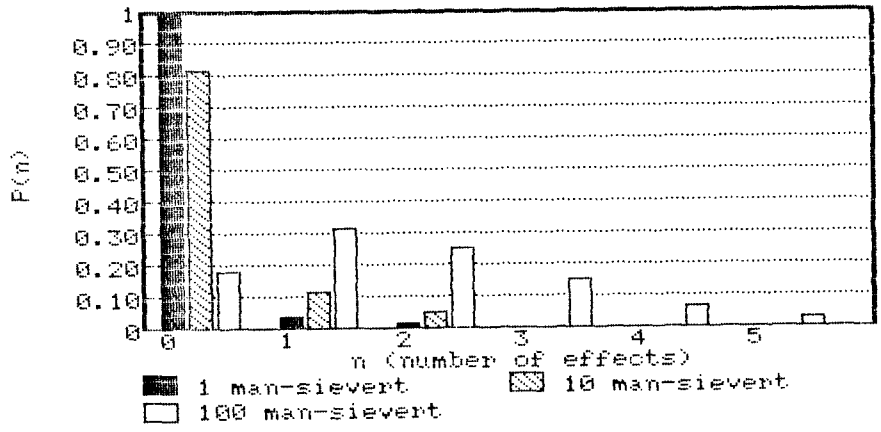
uncertaintes in **R** are also reflected in the numerical value of alpha, if obtained by methods not related to radiological protection (earning power of people, national income,...). The use of these methods shall be not encouraged.

4. Owing to the probabilistic nature of the relationship between dose and occurrence of stochastic effects, if **N** individuals are equally exposed at **H**, the individual probability **r = RH** of suffering deleterious effects does not depend on the "risk" incurred by the other **(N-1)** individuals in the group. The probability of -n- effects among the **N** exposed individuals is given by the binomial distribution (Fig. 1)

$$P(n) = \binom{N}{n} r^{n} (1-r)^{N-n}$$

with an expected value **E(n) = N.r** and a variance **V(n) = N.r(1-r)**.

Fig.1-P(n) for individual dose H=1 mSv



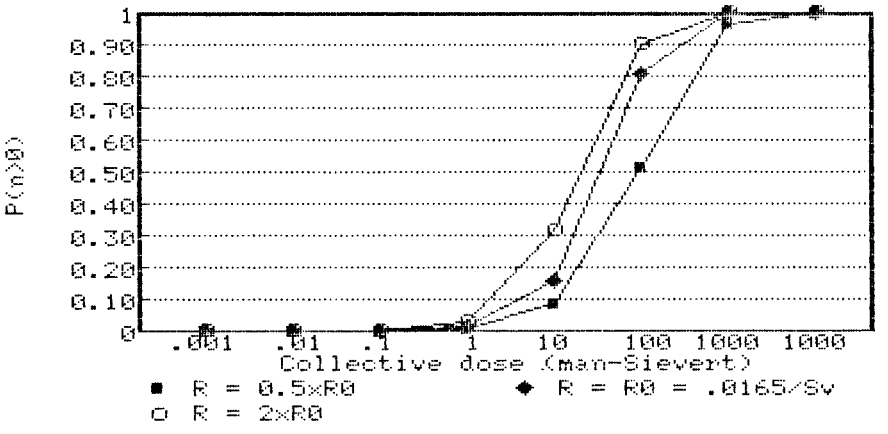
5. With reference to the collective dose **S**, we have **E(n) = R.S = G_H** and **V(n) = R.S (1 - R.S/N) ~ R.S = G_H** for **N** large. The objective health detriment **G_H** shall so be considered as the expected value from a binomial distribution, whose variance, for **N** large equals the expected value itself. The usefulness of such a statistical indicator could seem very questionable for low probability exposure: the expected value of the collective dose cannot be used for getting an indicator of the expected value of the detriment.

6. A more useful indicator could be the integral probability **P(n > 0)** of having effects in the exposed group,

$$P(n>0) = 1 - P(0) = 1 - (1 - R.H)^{N} \sim R.H.N. = G_H$$

a good approximation for $R.S \ll 1$. The additivity property holds also for this indicator, when the impacts of several radiation sources on a group of individuals can be considered, now or in the future, mutually independent: $P(n > 0)$ for each source shall be evaluated and the results summed. As an example, in balancing occupational and public exposures, the values of $P(n>0)$ for the workers and for the population shall be evaluated and summed to obtain $P(n > 0)$ for the different protective options to be selected.

Fig.2-P(n>0) for 1000000 individuals

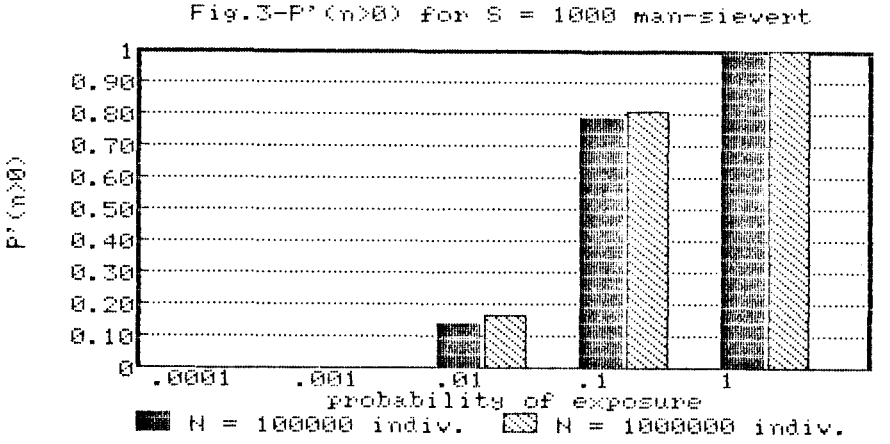


7. As shown in Fig. 2, the value of $P(n > 0)$ is always small for values of S up to 1 man-Sv, irrespective of the value actually assumed by the risk factor R. Using this indicator in a decisional process for selecting protective options could be very useful, because options leading to $P(n > 0)$ lesser than an assumed "indifference value" (1% - 5%) could be considered equivalent: values around 1 man-Sv could be considered as a "threshold" for introducing collective doses into decisional processes. A similar conceptual approach should be adopted for defining exemption levels of sources or practices from the regulatory control.

8. In the range 1 - 100 man-Sv $P(n > 0)$ depends on S and R, but more slowly as G_H does. Above 100 man-Sv $P(n > 0)$ becomes unitary for each reasonable value of R. It should however be clear in mind that such large values of collective dose can be considered only where a large number of individuals is exposed, because the risk factors adopted by the ICRP - and also the same concept of Effective Dose Equivalent - hold only below individual dose limits, i.e. at individual doses below 50 mSv. For high values of collective doses $E(n)$ becomes an useful decisional factor, because its variation coefficient is very small, at least for planned (unitary probability) exposures.

9. $P(n > 0)$ seems to be a very useful statistical indicator of the harm in an exposed group in the field of probabilistic (accidental) exposures. Expected values are very difficult to be handled in the case of rare events, because uncertainty is very large when the probability is low. The integral probability $P'(n > 0)$ of having effects in a group of **N** individuals equally exposed is given by the product of the probability -p- of the event and of the probability $P(n > 0)$

$$P'(n > 0) = p(1-(1-R.H)^{N})$$



Planned and probabilistic exposures could thus be treated in a unified decisional process (Fig. 3) by ranking the protective options on the values of $P'(n > 0)$, to be evaluated and summed for the various conditions of exposure before the options are compared.

10. In conclusion, some practical and conceptual difficulties could arise in using the concept of objective health detriment and, as a consequence, in optimizing radiation protection by means of the cost-benefit analysis. Its use shall be restricted to very simple situations, within a wide multicriteria analysis. The statistical indicator proposed in this paper, that is the integral probability of having effects, could be used in this multicriteria analysis for accounting in the decisional process for the health effects in the population exposed to radiation source or practice. It is worthwhile to note that the proposed approach does not require a formal definition of the quantity collective dose: only the number **N** of exposed individuals and per-caput dose **H** are conceptually required for its applications.

THE WEIGHTING FACTOR FOR THE FEMALE BREASTS; IMPLICATIONS FOR THE EFFECTIVE DOSE EQUIVALENT

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The results of several surveys indicate that the incidence of radiation-induced breast cancer for the worker would be 200 * $10^{-4}Gy^{-1}$ in the absolute risk model (UN77, BE80, LA84, UN86). The mortality to incidence ratio for breast cancer for The Netherlands is 0.4 (cf. GSF85, BE80). Therefore, the absolute risk model would yield a factor of 0.8 * $10^{-2}Gy^{-1}$ for fatal breast cancer although a value of 0.2 * $10^{-2}Gy^{-1}$ has also been reported (Si84). The relative risk model of BEIR (BE80) would suggest a risk factor for the female breast of over 1.2 * $10^{-2}Gy^{-1}$. The natural incidence of female breast cancer in The Netherlands is, with that in the U.K., the U.S.A. and Switzerland among the highest in the world. In this connection the Dutch National Health Council (GR85) considers a risk factor of 1.2 * $10^{-2}Sv^{-1}$ with a range of 0.8 * $10^{-2}Sv^{-1}$ to 1.6 * $10^{-2}Sv^{-1}$ for irradiation-induced fatal breast cancer a more representative value than the 0.5 * $10^{-2}Sv^{-1}$ given in ICRP publication 26 (ICRP77). In the case of uniform irradiation the total female risk - including the genetic contribution - would become 2.6 * $10^{-2}Sv^{-1}$ as compared with 1.4 * $10^{-2}Sv^{-1}$ for the male, as illustrated in figure 1. Hence the female risk, in case of uniform exposure, would approach twice the male risk. If the somatic effective dose equivalent (SED) is considered, the ratio becomes even larger. The SED is the uniform whole-body dose that will cause the same somatic risk as the actual (non-uniform) dose, the genetic risk not being considered. It has been pointed out (Be84) that the same SED in a woman carries with it 1.5 times as high a risk as in a man, on the basis of the "real" risk factors from ICRP 26 (i.e. 0.5 * $10^{-2}Sv^{-1}$ for the female breast). In the remainder of this paper we will adhere to the factor 0.8 * $10^{-2}Sv^{-1}$ for the mortality risk of breast cancer for the (female) worker, as the available evidence supports this value if the absolute excess cancer rate, from which the relative increase of natural incidence is obtained, decreases with age (GSF85, Pr87). However a risk factor of 1.2 * 10^{-2} or more per sievert should not be excluded in case of a single relative risk value for adults (BE80, Pr87).

The introduction of the effective dose equivalent brought equalisation between men and women. The somatic risk for the worker according to ICRP 26 is 1.25 * $10^{-2}Sv^{-1}$. This figure is an average over the sexes, as the somatic risk as used in ICRP 26 is in fact 1 * $10^{-2}Sv^{-1}$ for the male worker and 1.5 * $10^{-2}Sv^{-1}$ for the female worker. For the amount of harm or hurt originating from genetical changes in the first two generations, an average of 0.4 * $10^{-2}Sv^{-1}$ is used for the 18-65 worker age group. Therefore the risk factor for the "mono-mamma" worker becomes 1.65 * $10^{-2}Sv^{-1}$.

To demonstrate the effect of an increase of the risk factor for the female breast from $0.5 * 10^{-2}Sv^{-1}$ to $0.8 * 10^{-2}Sv^{-1}$, let us consider the effective dose equivalent H_E in the (extreme) situation in which solely the female breast is irradiated. As is also usual in SED determinations, we split the risk coefficients into one set for males and one for females. In the split ICRP-model an increase of the risk factor for the female breast from $0.5 * 10^{-2}Sv^{-1}$ to $0.8 * 10^{-2}Sv^{-1}$ results in an increase of the female H_E by a factor of 1.4.

Table 1. The weighting factors in the split ICRP model with fatal breast cancer risks of $0.5 * 10^{-2}Sv^{-1}$ and $0.8 * 10^{-2}Sv^{-1}$, respectively.

	Mortality (genetical) risk per unit dose, R, (10^{-4} Gy^{-1}) and weighting factors (w_T)					
Tissue	male		female			
	R	w_T	R	w_T	R	w_T
Gonads	40	0.29	40	0.21	40	0.18
Red Bone Marrow	20	0.14	20	0.11	20	0.09
Thyroid	5	0.04	5	0.03	5	0.02
Bone surface	5	0.04	5	0.03	5	0.02
Lung	20	0.14	20	0.11	20	0.09
Breast (♀)			50	0.26	80	0.36
Remainder	50	0.36	50	0.26	50	0.23
Total	140	1.0	190	1.0	220	1.0

From Table 1 the increase factor is obvious from the last two w_T columns : 0.36/0.26 = 1.4.

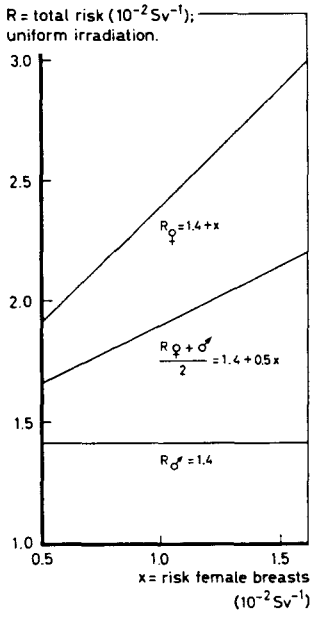

Fig. 1

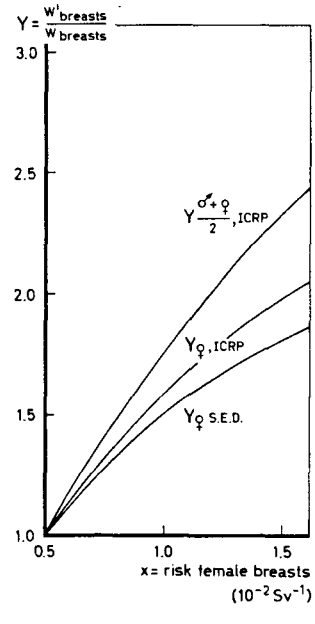

Fig. 2

In the same way the increase of H_E can be estimated for a variety of risk factors for the female breasts. The changing weighting factor for the female breast, $w'_{breasts}$, relative to the standard weighting factor inherent in the chosen model, w_{breast}, can be calculated as a function of the risk per unit dose.

Figure 2 presents this relative change $Y = \frac{w'_{breast}}{w_{breast}}$ as a function of the mortality risk per unit dose equivalent for the female breasts, $x(10^{-2}Sv^{-1})$. This relative change is given for three models:

- The ICRP model for the mono-mamma worker corresponding to the equation: $Y\ (ICRP) = \frac{1}{2}x * 1.65/\{0.25 * (1.4 + \frac{1}{2}x)\}$ with $w_{breast} = 0.15$.
- The "split" ICRP model: $Y_{♀}\ (ICRP) = 1.9x/\{0.5 * (1.4 + x)\}$ with $w_{breast} = 0.26$.
- The SED model: $Y_{♀}\ (SED) = 1.5x/\{0.5 * (1 + x)\}$ with $w_{breast} = 0.33$.

The above mentioned ratio of 1.4 can be readily found from figure 2 at $x = 0.8$ using the curve for the "split" ICRP model. If for the mortality risk of the female breast $0.8 * 10^{-2}Sv^{-1}$ is taken, the excess fatal cancer rate of females in case of uniform irradiation becomes 1.8 times that of males.

In the ICRP model of the mono-mamma worker an increase of fatal female breast cancer enhances the risk of the worker, irrespective of the worker's sex. Especially if the radiation-induced excess breast cancer risk becomes very large, a reconsideration of the equalisation of man and woman may become desirable. Even more so as the application of w_T is not limited to the radiation worker.

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BASIS FOR THE DESIGN OF RADIOLOGICAL PROTECTION INSTRUMENTATION FOR NUCLEAR FACILITIES

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INTRODUCTION

Radiological protection (RP) is vital to the design of all nuclear plant and the RP instruments to be installed or used in the plant must be of a high standard of reliability to ensure safe working and minimum risk. Plant size, type and inventory of radioactive materials will influence the complexity of the instruments ranging from simple monitoring for tritium in air in a luminizing facility to the full range of monitoring around nuclear reactors and in fuel reprocessing plants with: external gamma-ray and neutron detectors; radioactive gas monitors; and beta/gamma-ray and alpha particle detectors for radioactive aerosols. Installed instruments should have a standard of reliability comparable to reactor instrumentation and portable and personal dosemeters for delineating fields and measuring personnel dose for record purposes must be robust and designed for the purpose for which they are used.

RADIOLOGICAL BASIS FOR PLANT DESIGN

In the UK nuclear industry there are many codes of practice for the safe design of plant and its operation. There is a recent code of practice[1] (with a guidance note) specifically to give radiological guidelines. Radiological principles in the UK are based upon ICRP recommendations[2] which are converted to regulations[3] and a code of practice[4] by the UK Health & Safety Executive (HSE), with advice from the UK National Radiological Protection Board (NRPB).

The design dose targets[1], prior to optimisation of the design using cost benefit analysis, are, for radiation workers: 10 mSv/yr averaged over radiation workers on a particular plant; 15 mSv for any individual radiation worker; 150 mSv/yr extremity dose equivalent to an individual. The design target for non-radiation workers is 5 mSv/yr and 0.5 mSv for a typical member of the most exposed group of the public. These targets are in line with the latest guidance from NRPB[5] which anticipates ICRP recommendations but which has not yet been accepted by the HSE. It should be noted that the HSE regulations[3] require a management investigation if an individual's dose equivalent exceeds 15 mSv in a year, or if a woman of reproductive capacity exceeds 13 mSv to the abdomen in any consecutive 3 months period (10 mSv during a declared term of pregnancy). Also the operational controls for non-radiation workers and typical members of the public will be the same as the design targets[1].

INSTALLED RADIOLOGICAL PROTECTION INSTRUMENTS

The RP instrument engineer has to translate the operational requirements into instruments to demonstrate compliance and indicate levels above 10% of established limits. There is a code of practice for installed RP instruments[6] which is designed to assist the health physicist in specifying the requirement and then to provide guidance on satisfying the request and installing the instruments. RP instruments required in a plant would include:

criticality monitors; gamma-ray & neutron monitors; radioactive gas monitors in working areas & stacks, sampling and monitoring equipment for airborne alpha & beta-particle aerosols in working areas & stacks; hand & clothing monitors; liquid waste monitors. RP instruments will normally have alarms which should not be falsely triggered and need a high standard of reliability and availability (low maintenance down time) which should be comparable with that demanded of nuclear reactor instrumentation[7,8].

Specification by the Health Physicist. The specification starts with a definition of the operational characteristics, viz: sources & radiations, hazards, permitted doses, alarm levels, type of measurement, accuracy and degree of sophistication required. The radiation environment must be specified; for example radon and its daughters can interfere with measurements of alpha radioactivity in air and ventilation, weather and building design can all influence the radon activity level. Background radiation from gamma-rays or neutrons can be variable and interfere with measurements as can shielding and backscattering from structures. All of these effects need to be considered in the specification. The non-radiation environment is equally important, viz: temperature, pressure, humidity, vibration, electrical interference, corrosion, contamination & cleaning and accessibility; all need to be considered and ranges of operation specified. This specification exercise needs to be interactive to avoid over-stringent requirements which could be expensive and to ensure that the sensitivity of any equipment is identified before it is provided.

Specification by the Instrument Engineer. A target technical specification should be produced for preliminary consideration and discussion with the Health Physicist. This will provide an overall design specifying the measurement required, sampling techniques (if applicable), the alarm levels and failure indicators. Availability, ie 'the sum of the times for which the system is in the inoperative failed state divided by the total time during which it should be operative', and reliability ie 'the characteristic of an item expressed by the probability that it will perform a required function under stated conditions for a stated period of time', need to be agreed with instrument user. Information display has become a major part of the specification as plants come under computer control and RP instruments are required to provide their information centrally. Alarms (visible and audible) are still required close to the instrument and autonomous operation in the event of a computer failure is highly desireable. Information handling and centralised instrument require increasing amounts of computer software which is itself subject to the same availability and reliability audits that are applied to the hardware. This provision of software may reduce the reliance on human influences but they must still be considered. Hardware is completed by specification of power supplies (with battery back-up?), cables (reduce interference) and the physical supports for the instruments. Detailed manufacturing & test procedures, and quality assurance are needed before supply costs can be estimated.

Supply, Maintenance and Documentation. Finally comes installation, maintenance and documentation: it is usual for RP installed instruments to have a life of many years and so complete documentation at the start of installation is vital to reliability, maintainability and availability.

PORTABLE AND TRANSPORTABLE RADIOLOGICAL PROTECTION INSTRUMENTS

The specification procedures for portable and transportable equipment[9] is very similar to that for installed equipment except that it is much more

difficult to define the environment in which it will be used. Equipment will certainly be subject to more physical shocks if it is carried or worn and, whilst it should be robust enough for the purpose, it should clearly indicate when it has failed as the instrument, for example, a gamma-ray or neutron survey meter or contamination monitor is acting as the radiation sense organ of the user. Thus the instrument designer should maximise the identification of revealed faults and minimise the number of unrevealed faults. Other instruments will include personal alarm monitors for external radiation or for sampling and counting aerosols of alpha and beta particles. Separate specifications for instruments using passive detectors, eg film dosemeters and personal air samplers, need a combination of the requirements for portable and installed equipment for reading or counting the passive element.

COMPUTER SOFTWARE

As indicated above, the need to provide remote control and reading of installed and even portable RP instruments is producing an increasing requirement to provide software with at least the same reliability and availability as the hardware. Also it is unlikely that the software designer will have a completely free hand in choosing either his hardware or programming language which may be fixed by the overall plant control system. It is possible to provide, at some expense, direct software control within the instrument by preprogramming but once the output reaches the central control this flexibility of approach is lost.

Firstly it is necessary for the user to define an overall requirement in terms of immediate information, manipulation of data, recording of historical data, printouts, displays, failure indicators (including pump failures on samplers) & alarms and interaction with other programs. There may be a variety of installed instruments, eg measuring external gamma radiation, neutrons, gases, tritium, particulates, etc; and uses, eg workplace, stack, natural environment. Each type of instrument will require its own software although there may be much that is in common. Pocket dosemeters can be used for access control to restrict and record doses received: this information on individual access and dose may be recorded for comparison with recording dosemeters (film or thermoluminescent dosemeters). The results of all measurements and dose assessments for an individual will need to be stored to provide the legal dose records[10].

The development of expert systems to handle emergencies and provide rapid dose assessments and evaluations is an area of increasing importance in order to provide information quickly and prompt for further measurements[11].

TYPICAL ASSESSMENT PROCEDURE FOR RP INSTRUMENTATION

Ideally RP instrumentation should form part of the overall plant design from its inception so that sampling points can be fixed, instrument locations decided and cable runs organised. Fitting RP instruments into existing plant is more common but the principles are the same. The stages are as follows.

A Examine the plant and its potential hazards to workers and the public.
B Consider the RP design limits for within the plant and for releases to the environment under normal and accident conditions.
C Decide upon the operation of the RP service, its expertise, facilities for emergencies, services which can be bought in, eg bioassay.
D List all the possible RP instruments that might be required with monitoring levels (minimum & maximum) and possible methods of detection.

E Prepare simple specification giving: range of measurements; special requirements (eg separation of plutonium activity from radon background); uncertainties over the range; electrical requirements (eg power supplies, interference, alarms, reliability, availability); computer imput; test facilities; and environment (radiation and non-radiation).

F Look for connected requirements, where with a small modification an instrument can be used for 2 purposes, and interaction between instruments.

G Consider available instruments and decide where R&D is required to satisfy the specification.

H Define the computer software requirements for installed monitors (on-line) and off-line personnel dosimetry.

I Propose locations for RP instruments, cables, connectors (interfaces), racks, power supplies.

J Provide a provisional cost estimate for R&D, procurement, software and operation of the RP instruments and the RP service.

It is important that an instrument engineer is involved in the last two stages and before procurement in providing complete specifications, final costings, supply, installation, documentation, warranty and maintenance schedules. None of this procedure can be completed without reference to the customer (plant designer) and to the Health Physicist who will interpret the results from the RP instruments and software.

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STANDARDISATION AND CALIBRATION FOR RADIATION PROTECTION PURPOSES IN THE UNITED KINGDOM

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PRIMARY STANDARDS AND CALIBRATION RADIATIONS, X-, GAMMA AND BETA RADIATIONS

Primary standards for the realisation of the quantity exposure were established at the NPL several decades ago. Free air chambers are used for X-rays generated at 300 kV and below and graphite cavity chambers for X-rays generated at 1 MV and 2 MV as bremsstrahlung spectra from a Van de Graaff accelerator. About 10 years ago a set of X-ray qualities was developed providing a series of X-ray spectra specifically to be used for the calibration of protection-level secondary standard ionisation chambers. These spectra generated between 10 kV and 280 kV have approximately the same spectral resolution and for a given tube current provide approximately the same exposure rate at the same distance from the X-ray tube foci. With the exception that the 1 MV and 2 MV qualities have been replaced by collimated caesium-137 and cobalt-60 beams, this range of radiation qualities has continued in use up to the present time. In the near future the ISO narrow spectrum series of X-ray radiations [1] will replace the NPL qualities over much of the energy range.

For beta radiations a parallel plate ionisation chamber was established as a primary standard, enabling the beta ray absorbed dose rate to air to be determined at fixed distances from radionuclide sources.

DISSEMINATION OF CALIBRATIONS

Since the introduction of the series of X-ray protection-level qualities, the aim has been to provide calibrated secondary standards to outside laboratories which would undertake the routine checking of area monitors. Eighteen centres in the UK now hold such secondary standards and, including recalibration work and calibrations for laboratories overseas (25 countries), a total of more than 100 secondary standard protection-level systems have been calibrated against primary standards. Thus within the UK the means has been supplied for the traceable calibration of all area monitors at the 14-monthly interval stipulated in our new Ionising Radiations Regulations 1985 [2] and for the calibration aspect of instrument type-testing.

Secondary standards for beta-radiations have been supplied in the form of radionuclide sources calibrated in terms of absorbed dose rate to air or to tissue at fixed distances using the primary standard described earlier. These sources correspond to the Series 1 sources listed in ISO 6980 [3]. Table I indicates the approximate dose rates available for the sources of highest activity supplied for each nuclide.

TABLE 1. NPL SECONDARY STANDARD BETA-RAY SOURCES

Nuclide	Calibration distance mm	Absorbed dose rate to tissue at 0.07 mm depth mGy h^{-1}
Sr-90+Y-90	300	14
Tl-204	300	2
Pm-147	200	1

PERSONAL DOSIMETRY

All the UK personal dosimetry services are subject to approval by the Health and Safety Executive (HSE), and for the purpose of ensuring that 'doses are assessed on the basis of accepted national standards', periodic intercomparisons are organised. In the case of services for the monitoring of external radiations, dosemeters are irradiated at a major secondary laboratory equipped with NPL standards and sent out to the services for measurement. The service run from the centre housing this secondary laboratory is directly audited by NPL, and other services may also be directly audited by NPL at the request of HSE. The calibration of dosemeters in some services may be based on the use of NPL calibrated secondary standards but the majority make use of radionuclide sources of certified activity. In either case the procedures ensure that all personal dosimetry relates back to national standards. A similar scheme, based on the assessment of tritium in water samples, has been initiated for the auditing of internal dosimetry services.

QUANTITIES AND UNITS

Following the change to SI units in ionising radiation metrology, the British Committee on Radiation Units and Measurements (BCRU) recommended the use of the quantity air kerma K_a for photon dosimetry in air [4]. According to the quantity in which the secondary standard is scaled, NPL provides calibrations in terms of exposure in röntgens or coulombs per kilogram, absorbed dose rate to air in rad h^{-1}, and air kerma (rate) in mGy(h^{-1}).

The BCRU gave interim recommendations [4] for the approximate conversion to 'dose-equivalent' of readings on protection instruments scaled in röntgens, viz. 1 mR = 10 μSv, pending expected guidance from ICRU on new dose equivalent quantities. Following the publication of ICRU 39 [5] and the recommendation that the new quantities ambient dose equivalent, H*(d), and directional dose equivalent, H'(d), be adopted for area monitoring within the UK, recommended conversion factors have been published by BCRU [6] as given by Wagner et al [7]. For X-ray calibration qualities in general use in secondary standardising laboratories, factors are becoming available for converting the air kerma calibrations supplied by NPL to the new dose equivalent quantities. Table II lists factors for the ISO narrow spectrum series of X-ray qualities based on the use of measured spectra folded in with the conversion relationships given for monoenergetic photons. Similar factors are available for use with the NPL protection-level X-ray qualities as long as they continue to be employed.

TABLE II. FACTORS FOR CONVERTING AIR KERMA TO AMBIENT AND DIRECTIONAL DOSE EQUIVALENT - ISO NARROW SPECTRUM FILTERED X-RADIATION SERIES

Generating voltage kV	Mean energy keV	Conversion factors	
		$H^*(10)/K_a$ (Sv/Gy)	$H'(0.07)/K_a$ (Sv/Gy)
40	33	1.18	1.27
60	48	1.59	1.49
80	65	1.73	1.60
100	83	1.71	1.60
120	100	1.65	1.55
150	118	1.58	1.50
200	161	1.46	1.39
250	205	1.39	1.34
300	248	1.35	1.32

For NPL beta-ray secondary standards calibrated in terms of absorbed dose to tissue at a depth of 0.07 mm in units of mGy h^{-1}, the corresponding directional dose equivalent rate has the same numerical value with units mSv h^{-1}.

STANDARDS FOR SURFACE CONTAMINATION AND ENVIRONMENTAL RADIOACTIVITY MEASUREMENTS, AND FOR NEUTRON DOSIMETRY

To allow radionuclide surface contamination monitors to be calibrated regularly with reference to national standards, as required by regulations, NPL has developed a programme for the calibration of large area sources of α- and β-emitting radionuclides. The β-emitting nuclides include C-14, Pm-147, Tl-204, Cl-36 and Sr-90+Y-90 covering an energy range above 0.15 MeV, and the α-emitting nuclide is Am-241. These correspond to the Class 1 reference sources of ISO draft standard 8769 [8]: the calibration certificate gives the measured emission rate. The general requirements for Class 2 reference sources of the same radionuclides are the same as for Class 1 but these sources are calibrated against Class 1 sources using a reference transfer instrument. The requirement for such calibrated sources and for general agreement on methods for improving surface contamination monitoring capabilities has received support from a recent intercomparison of such measurements in UK hospitals [9].

Another requirement which has become evident recently is that for radionuclide solution standards in the 10-100 Bq/g range, in connection with studies of environmental activity. Such standards are now issued for an expanding range of pure and mixed radionuclides. A programme has started for the development of reference materials with agreed low levels of radionuclide content.

Traceability in the UK for radiation protection purposes for neutron radiations is achieved by providing access to standardised neutron fields at NPL, either source- or accelerator-based. These cover the energy range 1 keV to 20 MeV, the accelerator fields resulting from the use of proton and deuteron beams from a 3.5 MV Van de Graaff accelerator with a variety of targets. Neutron-sensitive devices of all types may be irradiated and precision long counters compared with the NPL standard precision long counter. Neutron dose

equivalent values are calculated from the standardised neutron fluence rates by the use of internationally agreed conversion factors.

NATIONAL MEASUREMENT ACCREDITATION SERVICE (NAMAS)

NAMAS, which has its headquarters at NPL, is a Government Service under which about 600 laboratories to date are accredited for certain measurement activities, either of testing or calibration. Calibration laboratories fall within the British Calibration Service (BCS) section of NAMAS and a current aim is to have secondary level radiation protection laboratories accredited within this system and added to the list of about 200 laboratories accredited for calibration work in a variety of fields. The assessment of laboratories is performed against published criteria and a list of the technical criteria for the ionising radiation field is given in Table III (* undergoing revision, + in preparation). The revised documents draw heavily on the ISO standards referred to in earlier sections.

TABLE III. BCS RADIOLOGICAL CRITERIA - INSTRUMENT AND SOURCE CALIBRATION

Doc. No.	Supplementary Criteria for Laboratory Accreditation -
B0811*	Calibration of Radiological Protection Level Instruments: X-, Gamma and Beta- Rays
B0813*	Calibration of Radiological Protection Level Instruments: Neutrons
B0814	Calibration of Radionuclide Sources: Activity, Particle or Photon Emission Rate, Exposure Rate or Air Kerma Rate
B0815+	Calibration of Surface Contamination Monitors

In addition to the supplementary criteria, general criteria are set for laboratory organisation and staffing; importance is laid on written measurement procedures and substantiation of claimed uncertainties of measurement. The use of NPL calibrated secondary standards is a central requirement and we believe that accreditation in this field will be an important contribution towards improved quality assurance in radiation measurements of all types.

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INSTRUMENTATION RESEARCH AND DEVELOPMENT IN U.S. DEPARTMENT OF ENERGY HEALTH PHYSICS PROGRAMS

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INTRODUCTION

The goal of applied research in instrumentation sponsored by U.S. Department of Energy's (DOE) Radiological Controls Division is to maintain the protection of radiation workers at the highest level commensurate with the current level of technology. Current research developments in the Division will be described in this paper.

Planning and controlling exposures to ionizing radiation require accurate, reliable instrumentation to establish and measure dose rates, to indicate high-exposure rate areas, and to control the spread of contamination. Control of the radiation environment, established with sophisticated portable and installed instruments, is verified by bioassay and dosimetry programs that also rely on sophisticated instrumentation.

The DOE Radiological Controls Division conducts several programs under the technical direction of the Pacific Northwest Laboratory (PNL), DOE's lead laboratory in health physics. These programs make use of DOE contractors, universities, and private companies to evaluate and upgrade measurement systems. Applied research is conducted in the areas of beta measurements, neutron measurements, internal dosimetry, air monitoring, and instrument evaluations.

INSTRUMENTATION PERFORMANCE

As a result of the observed poor performance of health physics instruments, a draft performance standard was written and evaluated experimentally at PNL. The evaluation [1] led to changes in the standard, so that all of the tests are practical; it is possible to design instruments that meet the various performance criteria. However, specific instrument types met certain selected criteria only with difficulty. An analysis of the errors for gamma survey instruments that passed the standard [2] showed that for Geiger-Muller (GM) detector-based instruments the expected accuracy would be ±30%, whereas for ionization detector-based instruments the expected accuracy would be ±26%. These are the quadrature sum of errors found during testing of the instruments and do not include errors in calibration, in reading the instrument (precision, recording errors, etc.), or in degradation of performance in the field. The dominant sources of error are angular dependence, energy dependence, and temperature dependence.

In order to meet recommended accuracies for field measurements [±30%, Ref. 3], instruments must be able to meet criteria similar to those found in draft American National Standards Institute (ANSI) Standard N42.17A. The DOE is planning a testing program for instruments used in their facilities.

NEUTRON INSTRUMENTATION

In the neutron program, a new type of radiation-detection instrument has been developed that uses the tissue equivalent proportional counter (TEPC) as the detector [4]. Based on developments in algorithms, counter stability, and electronics, a prototype instrument [5] was developed at PNL to measure the total dose equivalent in mixed radiation fields. This "Total Dose Meter" is designed as an alarming personnel monitor and complements existing dosimeters used for workers exposed to neutrons.

A block diagram of the electronic circuitry of the unit is shown in Figure 1. To measure the very small pulses from gamma rays, it was necessary to develop an ultra-low noise preamplifier, which has a root mean square (RMS) noise of only 130 electrons. To cover the wide range of pulse sizes, it was necessary to use two separate linear amplifiers, one for neutron events and one for gamma events. An analog to digital converter (ADC) was designed using a pulse-height-to-pulse-width circuit that is used to gate an oscillator. Pulses from the ADC are summed and stored in buffers until interrogated by a microprocessor.

BETA INSTRUMENTATION

Tests at PNL [6] have shown that many instruments have severe angular and energy dependence. Some instruments do have adequate characteristics; these instruments generally have a thin window (5 to 10 mg/cm^2) and a thin sensitive volume, both in terms of density thickness (1 to 5 mg/cm^2) and physical thickness (<2 cm). One instrument with suitable energy and angular response is based on a thin plastic scintillator (5 mg/cm^2) and uses pulse-shape discrimination to eliminate noise [7]. Because of this discrimination, complex power-consuming circuitry is necessary. This circuitry makes the instrument cumbersome and affects its long-term stability. Work is underway to improve the instrument and produce one that will operate reliably in the field.

Research has also resulted in the development of a coincidence system using a proportional counter and a thick scintillator to collect beta spectra in the field. This results in a scintillator that can collect either beta or gamma spectra with a rejection ratio of 1000:1. Spectra from a ^{207}Bi source are shown in Figure 2. Such a capability is important in studying beta fields present in a facility to implement effective protective measures.

Laser heating of thermoluminescent dosimeters (TLDs) [8] provides a system that enhances the signal-to-noise ratio by rapid heating of the TLDs and also provides high throughput. Commercial CO_2 lasers are not designed to produce the ultra-stable light

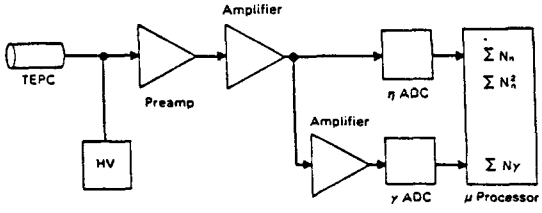

FIGURE 1. Block Diagram of Total Dose Meter Electronics. A neutron (n) and a gamma (γ) channel are used with the microprocessor summing the neutron (Nn) and gamma events (N_γ) and calculating dose.

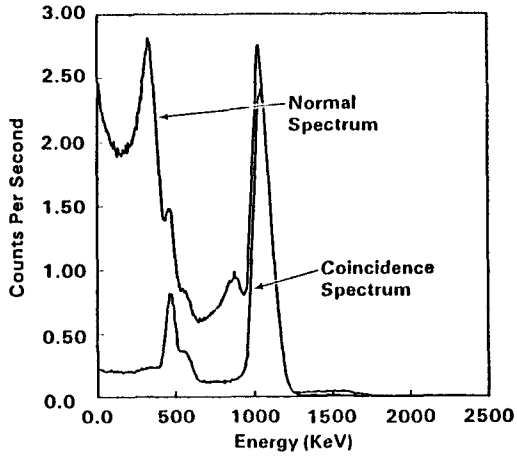

FIGURE 2. Spectra from ^{207}Bi Source in a Plastic Scintillator Operated Normally and in Coincidence with Proportional Counter in Front of the Scintillator

out-put needed to reproducibly read dosimeters. Applied research resulted in adding temperature stabilization and beam monitoring to the laser system to obtain reproducible heating of the TLDs. The beam profile from a laser system is gaussian (nonuniform) and research resulted in methods to produce uniform heating over a selected area (2 to 3 mm^2). The system can now reproducibly pre-anneal, read out, and post-anneal TLDs in 1 second.

INTERNAL DOSIMETRY

Accurate measurement of internal depositions of radionuclides is essential to determine radiation risk to workers who acquire an internal deposition of radioactive materials through inhalation, ingestion, or penetration of the skin. Bioassay techniques involve measuring trace amounts of nuclides in excreta, primarily urine. In 1982, PNL developed a new technique for precisely measuring uranium in aqueous solutions (urine). The technique, kinetic phosphorimetry [9], automatically compensates for quenching effects and therefore produces reliable measurements with less sample preparation than required previously. The instrument is extremely sensitive and can measure uranium concentrations down to a few parts per trillion. The DOE was awarded a patent for the unique capabilities of this technique, which has been commercialized.

The PNL is currently developing a new technique for measuring the concentrations of radionuclides [10]. This technique uses two pulsed dye lasers to resonantly excite only the target isotope to an electronic state just below the ionization threshold. Collisions with an inert gas ionize the excited atoms, and a thermionic diode detector measures the ion production. Recent results with

calcium as an analog for the transuranic elements indicate that detection limits of less than one femtogram should be feasible. This unique technique offers the extreme sensitivity of laser measurements in a relatively simple and inexpensive apparatus that will be useful in measuring nuclides in excreta. The technique may also be useful for measuring nuclides in aerosols.

CONCLUSIONS

Efforts have also included development of a transuranic air monitor [11], specialized studies of instrument performance, and prototype evaluation of several detector systems including fiber optics, organic semiconductors, and others. Recent developments, together with the health physics instrumentation testing, have resulted in improved measurement capabilities. Health physics instrumentation of the future [12] should encompass the latest technological advances in detectors and electronics to improve basic measurement capabilities and to provide greater reliability of existing instrumentation. Applied research will continue to provide the means by which to improve accuracy, ease of operation, reliability, and versatility of health physics instrumentation.

ACKNOWLEDGMENT

This work was performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

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RECHERCHE ET DEVELOPPEMENT POUR L'INSTRUMENTATION DE RADIO-PROTECTION A L'INSTITUT DE PROTECTION ET DE SURETE NUCLEAIRE

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I. INTRODUCTION

Les missions principales de l'Institut de Protection et de Sûreté Nucléaire (IPSN), créé en 1976 sous la tutelle du Ministère de l'Industrie, comprennent en Radioprotection :

- Les **études et les recherches** qui couvrent les domaines des effets des rayonnements sur l'homme et l'environnement, de l'intervention sanitaire et médicale, des activités techniques telles que l'instrumentation de mesure des rayonnements, la robotique, la filtration et la ventilation dans les installations nucléaires, etc.,

- L'**assistance aux services ministériels** en matière de réglementation, de plans d'intervention, ainsi qu'en matière d'information du public et de relations internationales.

En matière de R & D pour l'instrumentation de radioprotection, l'IPSN est chargé de la planification des programmes effectés dans les différentes unités du CEA et dans ses propres laboratoires, de répartir les ressources et d'assurer une coordination et un suivi de l'exécution des travaux.

Pour assumer efficacement ses responsabilités l'IPSN a créé deux comités et un centre technique dont les membres appartiennent à des unités qui participent à l'exécution des programmes et aux services centraux chargés des relations industrielles ou commerciales. Il s'agit :

- du **C.I.R.P.**, Comité de l'Instrumentation de Radio-Protection,

- du **C.C.P.I.R.**, Comité Consultatif en matière de Politique Industrielle pour l'Instrumentation de Radioprotection,

- du **C.T.H.I.R.**, Centre Technique pour l'Homologation de l'Instrumentation de Radioprotection,

lesquels ont pour mission :

- de prendre en considération les problèmes et les besoins des utilisateurs en matière d'instrumentation de radioprotection,

- de définir les caractéristiques techniques des appareils et d'initier les études en laboratoire,

- de rassembler les points de vue des représentants des autorités du CEA pour définir et conduire une politique industrielle pour la réalisation et la commercialisation de ces appareils,

- de maintenir le dialogue avec les industriels des firmes françaises qui produisent ces appareils et, si possible, lancer des développements industriels en étroite coopération,

- d'effectuer l'homologation des appareils dont l'utilisation est recommandée dans les installations du CEA, et de coopérer sur le plan national et international à la normalisation de l'instrumentation de radioprotection.

II. DOMAINES D'INTERET

Un des objectifs de l'IPSN est de développer une instrumentation et des systèmes de traitement et d'exploitation des données qui garantissent que les mesures, effectuées en routine ou en situation d'accident, répondent aux critères de radioprotection, aux prescriptions réglementaires et aux normes techniques publiées par la CEI et l'ISO.

Les programmes actuels recouvrent les domaines ci-après :
- Surveillance du personnel
- Surveillance des zones de travail
- Surveillance de l'environnement
- Surveillance en situation d'accident ou post-accidentelle
- Qualification et standardisation de l'instrumentation de radioprotection.

Dans chacun de ces domaines des études sont menées pour résoudre les problèmes techniques, améliorer la qualité des appareillages et des mesures, et définir les tests des perforrmances en vue de l'homologation. Toutefois la limitation des moyens et des ressources entraîne une concentration des efforts sur les **programmes prioritaires** suivants :

- **Dosimétrie des neutrons** pour les personnels et les zones de travail,

- **Surveillance de la contamination atmosphérique** dans l'environnement, les zones de travail, et au niveau de la sphère de respiration des travailleurs

- **Systèmes de traitement et d'exploitation des données** pour les Tableaux de Contrôle des Rayonnements (T.C.R.) des installations nucléaires en vue de la surveillance des zones de travail.

III. TRAVAUX EN COURS

3.1. DOSIMETRIE DES NEUTRONS

Les travaux menés dans ce domaine recouvrent quatre secteurs qui sont : les détecteurs solides de traces, les compteurs proportionnels équivalents tissus, les compteurs à hélium 3, et les procédures d'étalonnage ;

Les recherches en **dosimétrie à l'état solide** sont orientées vers la réalisation d'un dosimètre multi-éléments (**DINEM**) susceptible de couvrir toute la gamme des énergies de neutrons rencontrées dans les installations nucléaires. Dans l'état actuel d'avancement du projet ce dosimètre est constitué de deux éléments (PGPDIN, CN 85). La figure 1 montre la réponse du **DINEM** comparée à celle de l'émulsion Kodak NTA.

Deux **compteurs proportionnel équivalents tissus** de tailles différentes sont en cours de développement. Le grand compteur (Ø 5 cm, h 5 cm), dont la réponse en énergie est donnée à la figure 2, a été réalisé pour un appareil de mesure des champs de neutrons (**CIRCEG**). La gamme de mesure prévue s'étend de 10^{-6} à 10^{-1} $Sv.h^{-1}$. Le petit compteur (Ø 1,5 cm, h 5 cm), encore à l'étude, doit équiper un dosimètre électronique de poche qui fournira simultanément les équivalents de dose gamma et neutron.

Les études portant sur les **compteurs à hélium 3** ont conduit à la réalisation et à l'industrialisation d'un appareil portable (2,5 kg) pour la surveillance des zones de travail (**DINEUTRON**). Cet appareil, dont le débit d'équivalent de dose s'étend de $2\ 10^{-1}\ \mu Sv.h^{-1}$ à 99 $mSv.h^{-1}$, donne en outre la valeur moyenne du facteur de qualité des neutrons. La figure 3 donne la réponse en énergie du **DINEUTRON.**

Une nouvelle approche a été définie pour les **procédures d'étalonnage en neutrons,** basée sur les spectres réels rencontrés dans les installations nucléaires (réacteurs nucléaires, laboratoires plutonium, usines ...). A cette fin un catalogue des spectres réels les plus fréquents a été établi et des techniques sont mises au point pour reproduire en laboratoire certains de ces spectres qui serviront de référence pour les étalonnages. L'avantage de cette méthode est de simplifier les procédures à mettre en oeuvre et d'étalonner les appareils dans les conditions de leur utilisation.

3.2. SURVEILLANCE DES AEROSOLS

Depuis plusieurs années a été ressentie la nécessité de lancer des études sur le comportement et la mesure des aérosols atmosphériques, notamment des actinides. Les objectifs qui ont été retenus couvrent la réalisation de moniteurs pour :

- la surveillance individuelle,

- la surveillance des postes de travail et de l'environnement, et la mise au point de techniques d'étalonnage de ces moniteurs.

Les travaux sur le **comportement des aérosols** ont permis de concevoir des systèmes de prélèvement et de collection en vue de minimiser la perte des particules par effets mécaniques ou électriques. La figure 4 montre la structure optimale qui a été calculée. Celle-ci a été adaptée et ajustée pour chacun des appareillages réalisés : le moniteur individuel **MONICA** et les balises pour les installations et l'environnement (**Balise Pu, PAUM et Balise Environnement**) (voir tableau).

Le banc **ICARE** permet, grâce à une génération d'aérosols de granulométrie et d'activité connues, en présence ou non de radon, d'étalonner les moniteurs d'aérosols radioactifs et de vérifier, le cas échéant, l'efficacité des systèmes de compensation de la radioactivité naturelle.

3.3. TABLEAUX DE CONTROLE DES RAYONNEMENTS

Grâce au développement des microprocesseurs, des systèmes d'acquisition et de traitement des données ont été développés et associés aux électroniques de mesure et aux dispositifs de signalisation (alarmes visuelles et sonores) pour la surveillance continue des zones de travail dans les installations nucléaires. Le nombre important de voies de mesure à implanter dans les installations nucléaires du groupe CEA, plus de 3000 voies de mesure pour les années 80, nous a conduit au lancement d'un programme de R & D, en étroite coopération avec des industriels du secteur. Ce programme porte sur les unités de traitement des données et de signalisation (**SMOR**) associées aux divers

moniteurs placés dans les zones de travail et les calculateurs de gestion (CSMOR) assurant le stockage, l'exploitation et l'affichage des données.

La figure 5 montre la structure générale du système tel qu'il a été conçu et adopté pour les TCR des installations du CEA.

IV. CONCLUSIONS

Pour la plupart, ces travaux prioritaires doivent complètement aboutir en 1988 et d'autres activités actuellement menées à l'IPSN prendront une plus grande importance dans un proche futur. Il s'agit, pour l'instrumentation de radioprotection, de la mise en application des nouvelles grandeurs opérationnelles ICRU et des nouvelles valeurs du facteur de qualité des neutrons. Il en est de même pour certains programmes lancés à la suite de l'accident de Tchernobyl qui concernent notamment un réseau de surveillance de l'environnement pour le territoire national et la mise au point d'un système de dosimétrie individuelle à distance pour les interventions.

	Débit l/mn	Diamètre filtre mm	Activité minimale détectable	Compensation Radon
Monica	3	18	$1{,}5\ 10^{-2}$ LAI	Mécanique
Balise Pu / PAUM	100	46,5	Pu $5\ 10^{-4}$ LAI ; β 10^{-5} LAI	Microprocesseur
Balise Environnement	1000	135	α $2\ 10^{-2}$ à 10^{-3} Bq/m^3 ; β 10^{-1} à 10^{-3} Bq/m^3	Microprocesseur

LAI = Limite Annuelle d'Incorporation

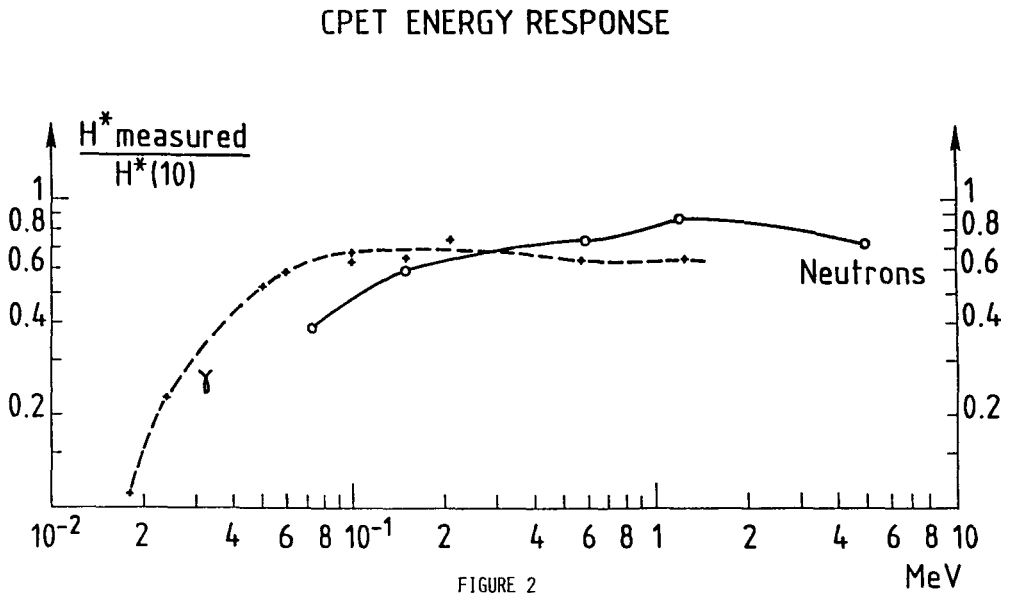

FIGURE 2

DINEM ENERGY RESPONSE

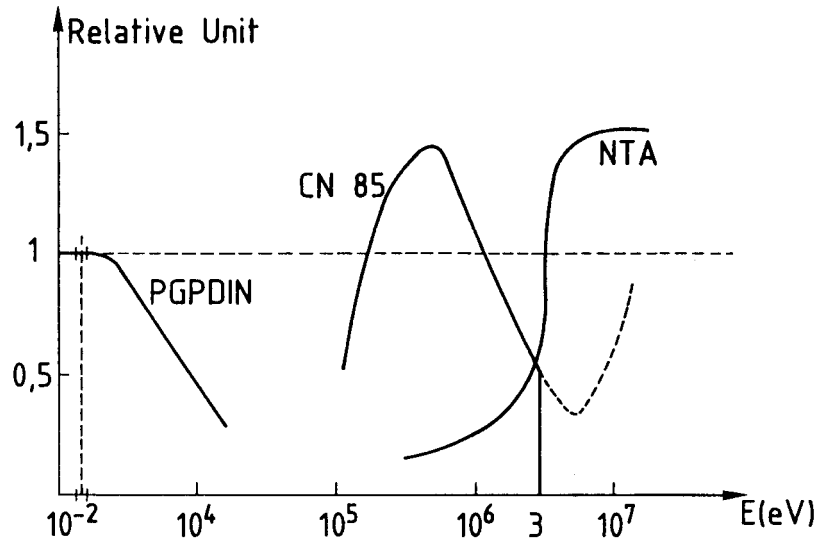

FIGURE 1

DINEUTRON ENERGY RESPONSE VERSUS INDEX

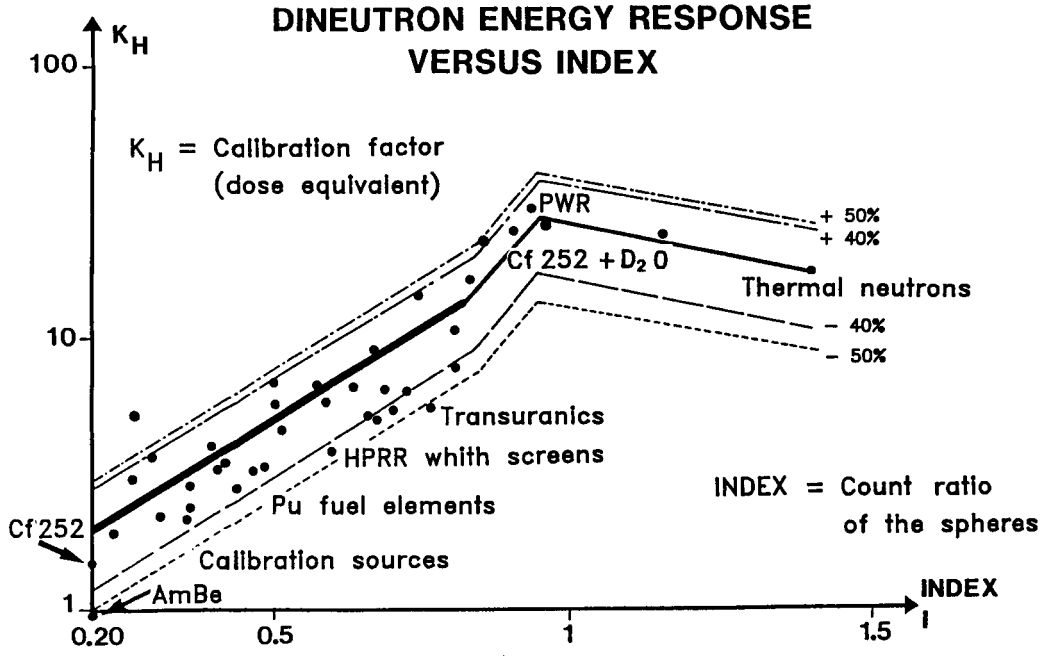

FIGURE 3

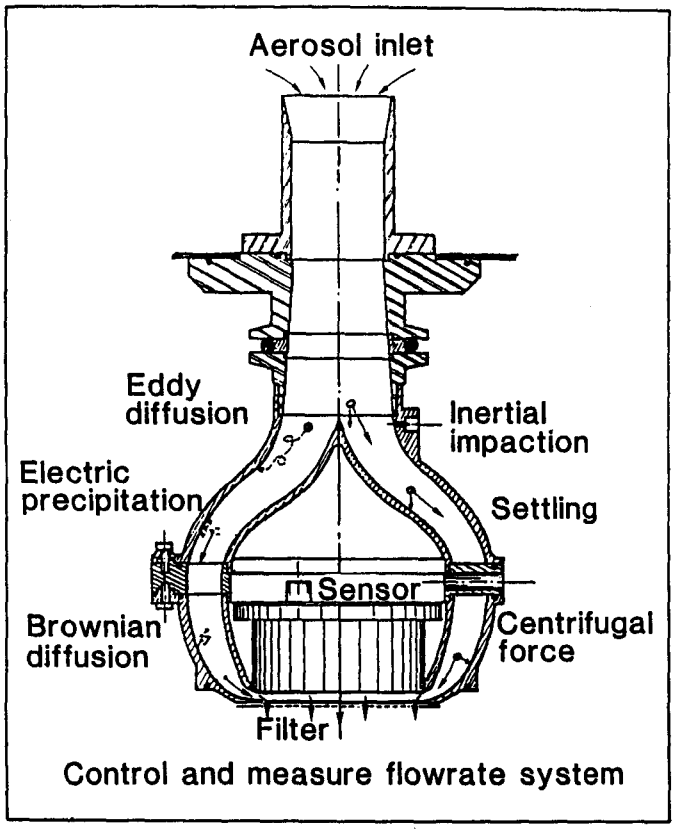

FIGURE 4

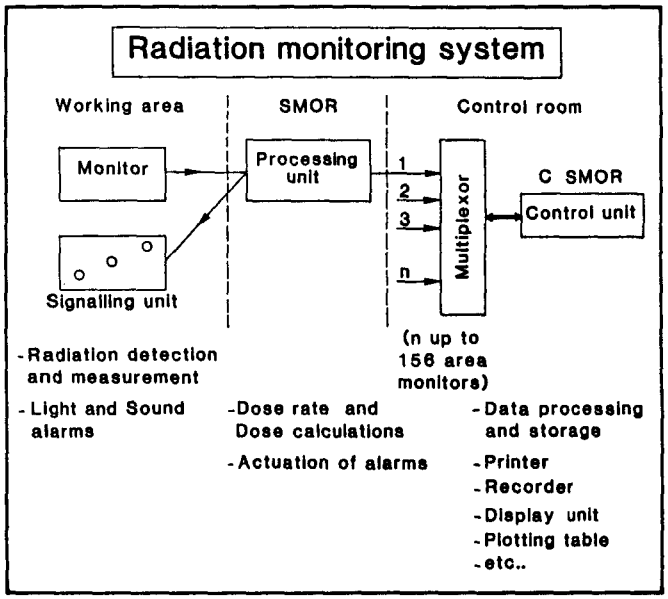

FIGURE 5

A PROGRAM TO MONITOR WORKER FACTORS AND EXTERNAL EXTREMITY AND WHOLE BODY OCCUPATIONAL RADIATION EXPOSURES IN A ACADEMIC SETTING

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INTRODUCTION

Since 1982 a formal ALARA program has been in place at the University of Iowa. ALARA is the recognized acronym for "As Low As Reasonably Achievable." The University of Iowa is required, by its Radioactive Materials License, to conduct all activities involving licensed radioactive materials and radiation producing devices in a manner designed to maintain personnel exposure to radiation at ALARA levels. (1) The initial goal of this program was to reduce all whole body quarterly personnel exposures to less than 3.75mSv (375mrem). The primary means of evaluating the effectiveness of the ALARA program, up through early 1986 was the careful review of printed radiation dosimetry reports. To accomplish this more than 2500 personnel dosimetry records printed on approximately 100 pages were reviewed individually on a monthly basis. This did not allow for easy comparison of personnel radiation exposures from one individual to another.

During the latter part of 1985 it was determined that a computerized system would be helpful in the management of the ALARA program, and aid in the distribution of personnel radiation monitoring devices. In order to be useful, a computer based data management system needs to meet several criteria:

(i) Allow for identification of workers by job classification, and work location.
(ii) Identification of all types and levels of radiation exposure.
(iii) The ability to quickly contact a worker should the need arise.
(iv) The use of a small personal computer that could keep track of one year of dosimetry data, (30,000 records and occuping 3.75 Mbytes of hard disk storage.)
(v) Locating a datatbase management system that could handle 30,000 records and run on an IBM-PC.

The commercial badge vendor (2) was contacted, and it was determined that such accounts would have to be set up in series codes. Each series code would specify a unique department within the teaching hospital, or a unique research group in the medical school or bio-science area. The supplier indicated that dosimetery data would become available on 5.25 inch diskettes for use with the IBM PC-DOS type personal computers. Metafile (3) an integrated information management system was selected for use with the IBM PC-DOS series computers.

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(1) Iowa State Department of Health Radioactive Radioactive Materials License, (1986)

(2) R.S. Landauer, Jr. & Co., 2 Science Road, Glenwood, IL 60425 , USA

Metafile is a free-form, relational database management system combined with text editing and a fourth generation programming language that allows for development of user-specified applications. The system requires 256k bytes of internal RAM when running under PC-DOS or MS-DOS version 2.1 or later; 2 disk drives, one of which may be a hard disk or a RAM disk device; a monochrome or color monitor, and a serial (asynchronous) RS-232C communications port. Initial development required 2 months for the program in distributing and accounting of personnel radiation-monitoring devices (PMDs). During this time each person wearing a PMD was assinged a series code. A series code is a three letter code for identifying the department and investigator. For example series code RRL, represents Radiation Research Laboratory. The first goal of the new system would be relating MBq of P-32 used per month vs. millirems per month exposure to the hands.

MATERIALS AND METHODS

In order to be included in the study a investigator must meet three criteria:

(i) The wearing of a finger ring TLD (4) with the detector facing in towards the palm of the hand. Also each participant needed to wear a whole body dosimeter (5).

(ii) Using P-32 on a regular basis, a minimum of 9.25 MBq (0.25 mCi) used per calendar month.

(iii) Within the laboratory there could be no other source of radiation, i.e. gamma-emitters or machine produced radiation.

In the initial study 55 P-32 users were identified. Of the 55 only 31 of the users ordered 9.25 MBq (0.25 mCi) or more per month. Only 15 investigators met the criteria for inclusion in the study. The study took place from March 1986, through May of 1987. Each investigator was placed into one of four groups based on the amount of P-32 used per month. During the course of the study the correct use of the PMDs was verified by observation. The volume of data generated by this report is presented in table format.

RESULTS

The minimum measurable exposure was 100uSv (10mrem) whole body, and 300uSv (30mrem) for beta exposure to the ring TLD. The linearity of the system was found to be excellent. The correlation coefficient, r , between MBq P-32 used per month and exposure was 0.94.

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(3) Version 9.2 Metafile, Metafile Information Systems, Inc., 15 East Second Street, Chatfield, MN 55923, USA

(4) Model U3 dosimeter, R.S. Landauer, Jr. & Co.., 2 Science Road, Glenwood, IL 60425, USA

(5) Model G1 dosimeter, R.S. Landauer, Jr. & Co.., 2 Science Road, Glenwood, IL 60425, USA

Table 1. Monthly exposure to personnel using P-32

Participant Id Number	P-32 used per month	Average exposure per month in uSv Whole body Deep	Whole body Shallow	Right Hand Shallow
1.	0 to 37 MBq	6.7 ± 26 *	33.3 ± 62 *	160.0 ± 155 *
2.	0 to 37 MBq	221.4 ± 185	300.0 ± 180	285.7 ± 280
3.	0 to 37 MBq	14.3 ± 36	35.7 ± 50	321.4 ± 272
4.	37 to 148 MBq	33.3 ± 51	33.3 ± 51	460.0 ± 318
5.	37 to 148 MBq	0.0 ± 0	66.7 ± 72	500.0 ± 290
6.	37 to 148 MBq	0.0 ± 0	6.7 ± 26	613.3 ± 245
7.	37 to 148 MBq	13.3 ± 35	140.0 ± 99	673.3 ± 284
8.	148 to 370 MBq	21.4 ± 43	85.7 ± 66	792.9 ± 144
9.	148 to 370 MBq	26.7 ± 59	80.0 ± 86	840.0 ± 235
10.	148 to 370 MBq	415.4 ± 99	515.4 ± 31	1061.5 ± 480
11.	148 to 370 MBq	20.0 ± 41	93.3 ± 88	1113.3 ± 479
12.	148 to 370 MBq	14.3 ± 38	42.9 ± 54	1171.4 ± 1382
13.	148 to 370 MBq	0.0 ± 0	20.0 ± 41	1186.7 ± 536
14.	370 to 925 MBq	26.7 ± 46	126.7 ± 134	1520.0 ± 1187
15.	370 to 925 MBq	80.0 ± 41	146.7 ± 64	2026.7 ± 406

* One standard deviation. Minimum exposures were set equal to zero, therefore average whole body exposures less than 100 uSv are minimal, and added as zero. The deep dose should be considered equivalent to the whole body dose. The deep dose is the dose equivalent from all radiation at a depth of 1.0 cm (1000 mg/cm sq.) in soft tissue. The effects of buildup and attenuation of radiation in the body are considered in accord with the International Commission on Radiation Units and Measurements Specifications. The shallow dose should be considered as the dose to the skin of the whole body. The shallow dose is the dose equivalent from all radiations at approximately 0.007 cm (7 mg/cm sq.) in soft tissue. The shallow dose equivalent takes into consideration scattered radiation within body.

SUMMARY

Several factors must be taken into account before drawing any conclusions from this data. The field gradient from a pure beta source is more dependent on distance, than the shape or size. (Th87) The data indicates that special attention should be given to situations where the chance of beta exposure exist, because significant exposure to the hands may occur. The use of a computerized isotope inventory system along with the ability to effectivley identify, as well as corelate dosimetry, has the potential to reduce unwarranted exposure to the hand and other extremities. This program has the potential to be applied to other isotope manipulations.

REFERENCES

Th87 Thind K. S., 1987 "Extremity Dose: Its Definition, Standards and Regulatory Limits, Radiobiological Significance, Measurement and Practical Considerations" Health Phys. 52, 695-705.

U.S. DEPARTMENT OF ENERGY'S RADIOLOGICAL CALIBRATION INTERCOMPARISON PROGRAM

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The United States Department of Energy's (DOE) Radiological Calibration Intercomparison Program is funded by the DOE Office of Nuclear Safety and carried out at the Pacific Northwest Laboratory (PNL) by Battelle. The program operates in the following manner: An invitation is sent annually to DOE personnel responsible for managing or performing radiological calibrations. Prospective participants return a form requesting either the instrument set or the secondary standard beta source set. The requested set is scheduled for use based on availability. Prior to sending the instrument set, the instruments are calibrated using a ^{137}Cs source. The beta sources are calibrated annually. When the results of the irradiations are returned to PNL, they are compared to PNL calibration values, summarized, and reported to the participant. At year end, results are summarized in an annual report to DOE. Participation in the program is voluntary, and the results are confidential and included in the annual report with the permission of each participant.

The instrument set contains three ionization chambers and a Geiger-Mueller counter (GM) as the detectors, a dosimetry-grade electrometer, a scaler/ratemeter, a battery-pack high-voltage power source, and the cabling necessary to perform the measurements. A 30-cc thin-walled air-equivalent plastic ionization chamber is used for measurements in photon fields. The response of the chamber for measuring air kerma is nearly independent of the energy of incident photons for energies between 20 keV and a few MeV. A build-up cap is provided to extend the useful energy range above 200 keV.

An 80-cc tissue equivalent (A-150) plastic ionization chamber (TE chamber) is used in conjunction with the GM counter for measurements of neutron kerma rate in mixed neutron and gamma fields. For convenience, the chamber is usually operated with ambient air as the fill gas instead of tissue equivalent counting gas.

The GM counter consists of a 0.48-cm-diameter tube with an active volume of 0.12 cc. The fill gas is neon, quenched with a halogen quenching agent. The detector is enclosed in an energy-flattening shield to eliminate the normal low-energy photon response characteristic.

The extrapolation ionization chamber is a 2-cm-diameter parallel plate ionization chamber with a continuously variable volume between 0.1 and 1.4 cc. The entrance window has a density thickness of 6.9 mg/cm^2. The collecting electrode and guard ring are constructed of A-150 tissue equivalent plastic and the chamber is open to ambient air.

guard ring are constructed of A-150 tissue equivalent plastic and the chamber is open to ambient air.

The electrometer is a multifunction meter capable of measuring integral charge between 10^{-8} to 10^{-14} coulombs. The scaler/ratemeter used with the GM counter has a built-in high-voltage power supply and a seven-digit display. The electrometer and scaler/ratemeter both operate on 110 VAC or internally contained battery supplies.

The secondary standard beta source set contains three beta-emitting sources, including 74-mg ^{90}Sr, 18.5-MBq ^{204}Tl, and 518-MBq ^{147}Pm. For irradiations, a source is removed from the storage jig using the source handling tool and secured in the irradiation jig. The shutter on the jig is controlled using a microprocessor-based controller/timer. Also included in the set are beam-flattening filters, reference distance rods, the jig stand, and a set of operating procedures.

The calculations used to determine the reference quantities (absorbed dose rate, tissue kerma rate, or air kerma rate) are described below. For photon irradiations, the average integrated current is multiplied by the air density correction (Ctp), the electrometer calibration factor (Ce), and the chamber calibration factor (Cg). The calibration factor converts collected current to air kerma rate.

The calculations of dose rate from the extrapolation chamber measurements follow the method in the ISO Beta Standard. The average slope of current versus plate separation is multiplied by the air-to-tissue stopping power ratio, the W value of air, and divided by the electrode area and air density to yield a value of absorbed dose rate at 7 mg/cm^2. The air density corrections pertaining to the chamber interior and the electrometer calibration factor are applied to individual collected currents prior to the least-squares fit of data, giving the desired slope mentioned above.

The neutron tissue kerma rate is determined by first determining an equivalent "air kerma rate" using the TE ion chamber and the GM counter and the procedures enumerated for photon irradiations above (substituting counts from the GM for corrected charge). The neutron air kerma rate is the difference of the quantities determined by the TE ion chamber, which measures total kerma rate, and the GM counter, which is used to determine photon kerma rate. The neutron air kerma rate is converted to neutron dose rate in tissue following the method of AAPM Report 7 and ICRU 30. The conversion factor contains the ratios of wall-to-gas stopping powers for neutrons and the calibration source, the ratio of W values of neutrons and the calibration source, the ratio of A-150-to-tissue kerma factors for neutrons and the calibration source, the attenuation and scatter factor, and the air-to-tissue kerma factor conversion.

The absorbed dose rates from the secondary standard beta sources are calculated following the method of Bohm. The dose rate at a specific date of calibration is corrected for

radioactive decay and air absorption. The dose rate from the ^{147}Pm source is further corrected for humidity effects.

The results of measurements are divided by reference results determined at PNL to yield ratios of intercomparison. During fiscal year 1986, seventeen measurements were performed on a variety of isotopic photon sources (^{137}Cs, ^{60}Co, and ^{226}Ra). Seven of those measurements utilized the photon transfer chamber. The average of ratios between field measurements and PNL reference values was 1.23 ± 0.26 for the seven measurements. During fiscal year 1987, twenty measurements were performed on ^{137}Cs and ^{60}Co sources using the photon transfer chamber; the average of the ratios was 1.01 ± 0.04.

During fiscal year 1986, one measurement was performed on a $^{239}PuBe$ neutron source. The ratio of the measurement to reference was 4.55. Six measurements were performed on PuBe sources; the average of the ratios was 1.8 ± 0.9. Problems with measurements of neutron sources stem from low kerma rates and irradiation geometries (e.g., increased neutron scatter in neutron wells).

No measurements have been performed on x-ray or beta sources belonging to individual participants using the intercomparison instrument set.

During fiscal year 1986, results were reported from two irradiations by participants using the intercomparison secondary standard beta set. The average of ratios for the ^{90}Sr source was 1.02 ± 0.05. Only one set of measurements yielded useful ratios for the ^{204}Tl and ^{147}Pm sources and those ratios were 1.11 and 0.63, respectively. During fiscal year 1987, three measurements were performed using the beta set. The ratios from the set of available data were 0.98 for ^{90}Sr, 1.00 for ^{204}Tl, and 1.09 for ^{147}Pm.

In conclusion, the accuracy and precision of intercomparison results has improved since fiscal year 1986. The most important benefit of the program continues to be the dissemination of radiological calibration information through the operation of the intercomparison program and the biannual radiological calibration workshop.

ALARA PRACTICES DURING NEUTRON SPECTRAL MEASUREMENTS INSIDE REACTOR CONTAINMENT

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The accurate assessment of radiation dose to personnel who enter reactor containment is a difficult problem compounded by the presence of mixed radiation fields of betas, neutrons, and high- and low-energy photons. Many present dosimeters do not adequately assess the true dose from neutrons, betas, or high-energy photons. In 1980, the National Council on Radiation Protection and Measurements (NCRP) announced that it is considering lowering the maximum permissible dose for neutrons, perhaps by a factor of 3 to 10 less than existing limits. These changes could have serious consequences for the operation of present commercial nuclear power plants and the design of new plants. Present personnel dosimeters will not be adequate if these proposed changes are adopted; in fact, many dosimeters are not sufficiently accurate to be adequate with existing limits.

Knowledge of the neutron radiation fields inside light water reactor (LWR) power plants is required for adequate interpretation of neutron monitoring instruments and personnel dosimetry. The needed information includes: neutron dose rates, neutron dose equivalent rates, and the spectra of neutron energies present. The measurements to acquire this information should be conducted inside reactor containment during power. These measurements will supply data needed to accurately assess personnel neutron dose and to determine the adequacy of the bioshield design.

Battelle Northwest Laboratories (BNW) has conducted neutron dose and energy spectral measurements inside reactor containment at many LWRs over the past five years. Preliminary studies have shown a degraded neutron spectrum inside reactor containment at LWRs with few neutrons with energies greater than 1 MeV. Neutron dose rates were found to vary significantly from location to location inside containment. Measurements have indicated that present neutron monitoring instruments may respond from 50 percent to 600 percent high, depending on the instrument and location inside reactor containment. Personnel neutron dosimeters have responded high by up to factors of 25 depending on the dosimeter type and the calibration method. Although measured spectra at the different locations are somewhat similar, differences related to shielding design rather than reactor and pressure-vessel design are apparent. The BNW staff have primarily used the following neutron measurement systems: 1) the multisphere spectrometer system, 2) the ^{3}He spectrometer system and 3) the Tissue Equivalent Proportional Counter (TEPC).

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* Pacific Northwest Laboratory (PNL), also known as Battelle Northwest (BNW), is owned by Battelle Memorial Institute and operated for the Department of Energy.

The multisphere spectrometer system is a commercially available system for measuring a wide range of neutron energies from thermal (0.025 eV) to 20 MeV. The spectrometer consists of a ^{6}LiI scintillator detector and a set of polyethylene spheres of different diameters. The ^{6}Li scintillator is inserted into a hole drilled into each sphere and is used to measure the slow neutron flux at the center of each sphere. The scintillator detector is connected to a multichannel analyzer (MCA), which is used to determine the count rate from neutron-induced events. Measurements are made with the bare scintillator, the detector covered with cadmium, and the detector at the center of 3-inch to 12-inch-diameter spheres. The count rate from each configuration is used as input to a computer code which determines the neutron fluence as a function of energy and the neutron dose equivalent.

The ^{3}He spectrometer consists of a ^{3}He proportional counter and associated electronics. Neutrons interact with the ^{3}He to produce a proton and a triton. If the reaction products are absorbed in the gas of the proportional counter, the resultant pulse is proportional to the original neutron energy plus 764 keV. A simple computer code corrects for the energy dependence of the ^{3}He detector and calculates the relative flux. The ^{3}He neutron spectrometer has much better resolution and accuracy than the multisphere spectrometer in the higher neutron energy regions above 20 keV, where neutrons are more effective in producing dose. However, the He-3 is only used at neutron energies below 5 MeV due to the limitations in the present analysis method.

The TEPC is a hollow sphere of tissue-equivalent plastic with the 3.2 mm-thick walls filled with tissue-equivalent gas. Details of plastic and gas composition and methods of construction can be found in Report 26 of the International Commission on Radiation Units and Measurements (ICRU 1977) [1]. This form of TEPC, called a Rossi counter, has a helical grid around the central anode wire. The helical grid establishes uniform electric field strength along the entire length of the anode. This produces the needed uniformity in gas amplification at all points along the anode for proper pulse-height analysis. The plastic sphere is contained inside a metal pressure vessel with a valve for admitting tissue-equivalent gas. The gas pressure is maintained at a pressure so that charged particles crossing the cavity lose only a small amount of energy as they traverse the counter. Energy deposited in the cavity is then equal to the linear energy transfer (LET) of the particle times the path length. At these low pressures, the gas-filled cavity has the same mass-stopping power as a sphere of tissue (ρ = 1 gm/cm^{3}) with a diameter of about 1 μm and is said to have an "equivalent diameter" of 1 μm. The TEPC is a device which measures neutron absorbed dose directly and is now under development by the United States Department of Energy (DOE) as a next-generation neutron monitoring instrument. It is anticipated that within the next 2-3 years, it will replace the instruments now in use. The TEPC can also be used directly to determine quality factors.

The above detectors are used with a multichannel analyzer and supporting electronics (e.g., high-voltage power supplier, linear amplifier, portable computer, etc.). Including the detectors and auxiliary equipment, up to 500 pounds of detection systems are taken into containment during the measurements. Because of the high temperature and humidity in reactor containment and the high radiation areas in some locations, it is necessary to limit the time of the personnel in containment to a minimum. The BNW staff follow ALARA principles to minimize the non-radiological hazards as well as the radiological hazards while performing neutron measurement inside reactor containment during power.

Probably the most important method of observing ALARA during these measurement conditions is planning. Prior to the measurements a planning meeting is held with the staff at the reactor site and the measurement procedure is thoroughly reviewed. Topics at this meeting include: 1) identification of detector and MCA setup locations, 2) personnel responsibilities and task assignments, 3) dressing procedures, 4) emergency escape hatch locations, 5) emergency signals and appropriate responses, 5) identification of dose levels in the radiation zones to be entered, and 6) any special radiation work procedures.

Each neutron detection system is thoroughly checked over at BNW before shipping offsite to the power plant. This reduces the time spent trouble-shooting the equipment in containment. All cables are sheathed in plastic at BNW prior to shipping. The equipment is wrapped in plastic at the plant. This, of course, reduced the chance of contamination to the instruments inside containment.

It is beneficial to select measurement locations inside containment that have high neutron dose rates because sufficient data can be collected in shorter periods of time. Thus, we have developed methods to limit the amount of time the equipment operator actually needs to spend in these higher-dose areas. With the use of 50-ft to 100-ft signal and pre-amp power cables, the neutron detector/counter can be placed at the measurement location and the associated electronics can be separated at a significant distance. The equipment operator needs to spend just a few seconds at the high-dose area setting up or removing the detector. The remainder of the time, the operator can be at the controls for the detector, located in a lower-dose rate area. If the collection time is long enough and/or the dose rates high enough, the operator can keep the instruments running and leave containment to further reduce his exposure. Generally, each system requires just one person to operate. However, the buddy system is always employed so that the people are always in sight of each other and easily accessible. Two people are used to set up and remove the equipment. Carts on wheels are used to move the system faster and more safely.

Because of the high temperature and humidity in reactor containment, precautions are also taken to reduce the effects of heat stress to the personnel. Ice vests are worn to allow operators to work more comfortably and efficiently. Stress

monitors are used to determine the maximum stay time for an individual in a certain area or in containment in general. The analyzer is usually set up near a blower since this is normally a low-dose area. It also cools the equipment and the operator of the equipment is more comfortable.

The BNW staff have conducted many measurements inside reactor containment that have had temperatures in excess of 120°F and greater than 90% relative humidity. Yet the staff have been able to stay in containment long enough to conduct the measurements in all the desired locations (generally 8 to 12 locations are measured). To date, no one has been incapacitated due to heat stress. Although some of the dose rates measured were as high as several hundred R/h no one has exceeded 1500 mrem exposure for a complete set of measurements. The average is typically 200-800 mrem per person for four people, or about 2 man rem per set of measurements. No one has exceeded an internal administrative limit of 2800 mrem annually.

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EVALUATION OF THE INSTRUMENT CORRECTION FACTORS NEEDED IN BETA DOSIMETRY

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INTRODUCTION

A wide variety of portable survey instruments using Geiger-Mueller (GM) detectors, ionization chambers, or scintillation detectors exists for the measurement of gamma dose rates. Generally, the same instruments are used for beta monitoring, but the beta response of these instruments has been secondary to their development, calibration, and use; information on the beta response is difficult to obtain and seldom provided by the manufacturer.

Survey meters are calibrated using fields that uniformly irradiate the detector volume with the center of the volume used as a reference point. However, under field survey conditions, frequently encountered sources have irregular shapes and large or small dimensions compared to detector volume. When the distance from the source to the detector volume is small, the radiation fields in the detector are non-uniform. Similarly, low-energy beta sources cause non-uniform irradiation of the volume because of attenuation of the radiation. The readings from such fields are an average of the energy deposition within the detector. This reading would be significantly less than the actual dose existing at the surface of the entrance window, which is an appropriate reference point when working close to the source. To compensate for this discrepancy, correction factors are applied to the readings to give the actual dose. These factors can be in excess of a factor of 100. This paper summarizes observations on energy, angular, and source-geometry response of survey instruments.

INSTRUMENT ANGULAR RESPONSE

The variation in response of portable survey instruments with changes in angular incidence of the radiation is more pronounced for beta radiations than for photon radiations because of attenuation of the betas in the side walls and construction materials of the instruments. Figure 1 shows the angular response of four instruments exposed to a ^{90}Sr-^{90}Y beta source conforming to International Organization for Standardization (ISO) Standard 6980-1984 Series 1 criteria [1]. The instruments were rotated around a vertical axis intercepting the reference point (center of the sensitive volume) with the reference orientation (0°) being defined when the source-to-detector axis was perpendicular to the front window. Instrument A (cylindrical

ionization chamber) has a thin entrance window, but its metal case causes a pronounced angular dependence. Instrument B has the same configuration with a shallow cylindrical chamber. The decrease in depth (1 cm versus 4.8 cm) results in a flat response, out to 45°. Instruments C and D are ion chamber instruments that have Styrofoam end and side walls resulting in a nearly flat response. Tests on five additional instruments produced a response at 45°, which was 33% to 70% of that at 0° [2]. At 90°, the response was 1% to 12% of the reference response.

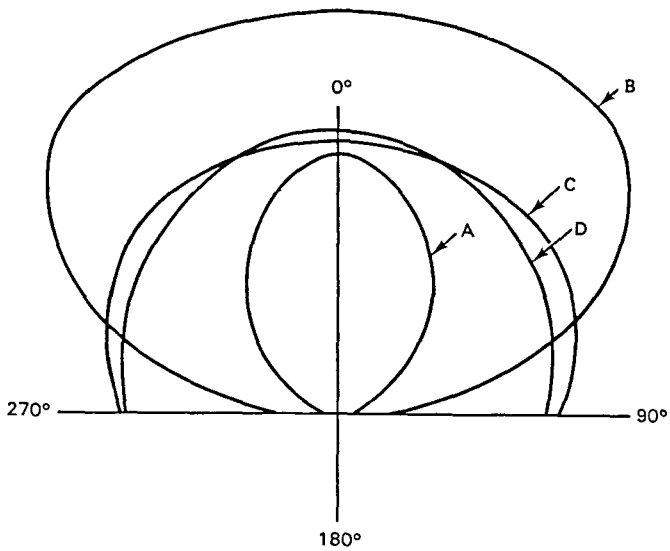

FIGURE 1. Angular Response of Four Instruments to a ^{90}Sr-^{90}Y Beta Source. The Scale is Relative.

ENERGY RESPONSE

Attenuation of betas within the volume of the detector and in the window will lead to an energy-dependent response of the detector [2]. The flattest energy response was obtained with instruments designed for beta measurements. One was an ion chamber modified to decrease the depth of its sensitive volume to approximately 1 cm, whereas the second was a dual ion chamber system with a "thin" chamber for measurement of the beta dose rate. These same instruments have a flat response to low-energy photons with an overresponse of about 10% around 200 keV.

SOURCE GEOMETRY RESPONSE

A technique was developed by PNL to determine source geometry response of instruments by measuring the response to a matrix of point sources at 0.5-cm increments [3]. Source geometries of interest are created from the matrix by selecting and normalizing the response for an appropriate collection of points. The measured data is used to calculate correction factors (ratio of actual dose rate at the surface of detector to response of the instrument) for the selected geometries. These factors are applied as a multiplicative correction to instrument

readings to determine actual dose rates at the detector window. The actual dose rate was based on measurements using calibrated thermoluminescent dosimeters (TLDs).

The correction factors obtained for various distances and dimensions for disk sources are shown in Tables 1 and 2 for three instruments. Instruments A and C are commercial ion chamber survey instruments, and D is a prototype ion chamber of 1-cm depth and 6.0-cm diameter. In addition to the noticeable differences among the instruments, variations are also evident among the correction factors for the ^{90}Sr-^{90}Y and ^{204}Tl sources. Studies also showed that the correction factors will vary with the materials in the vicinity of the source because of backscatter. Changing the source stage from Styrofoam to lead changed the correction factor from 4.7 to 6.3 for ^{204}Tl with instrument B. Scans were also performed with ^{57}Co and ^{137}Cs sources and resulted in similar correction factors.

TABLE 1. Measured Correction Factors for ^{90}Sr Disk Sources

Source Diameter, cm	0.5 cm Instrument			1.0 cm Instrument			2.0 cm Instrument			5.0 cm Instrument		
	A	C	D	A	C	D	A	C	D	A	C	D
0.3	38.3	21.0	9.3	13.9	7.2	3.5	5.7	2.9	1.5	2.5	1.1	0.79
5.0	6.2	3.0	1.4	5.9	2.5	1.3	4.0	2.0	1.1	2.3	1.1	0.77
10.0	3.6	1.9	1.1	3.6	1.8	1.0	3.0	1.6	0.9	2.1	1.1	0.75
15.0	2.8	2.0	1.3	2.7	1.9	1.1	2.3	1.6	0.96	1.9	1.1	0.77
21.0	2.2	2.1	1.4	2.2	2.0	1.2	1.9	1.8	1.1	1.6	1.1	0.82
27.0	2.0	2.2	1.5	2.0	2.2	1.3	1.7	1.9	1.2	1.5	1.2	0.89

TABLE 2. Measured Correction Factors for ^{204}Tl Disk Sources

Source Diameter, cm	0.5 cm Instrument			1.0 cm Instrument			2.0 cm Instrument			5.0 cm Instrument		
	A	C	D	A	C	D	A	C	D	A	C	D
0.3	79	39	17	30	15	6.7	11	5.5	2.8	5.6	2.5	1.6
5.0	12.5	5.4	2.5	11.9	5.1	2.5	8.4	3.6	2.0	5.2	2.3	1.6
10.0	8.6	3.6	2.1	8.6	3.6	2.0	6.7	3.0	1.8	4.5	2.1	1.6
15.0	7.2	4.2	2.4	6.9	4.3	2.3	5.5	3.4	2.0	3.8	2.0	1.6
21.0	--	4.8	2.7	--	5.0	2.6	4.6	4.1	2.3	3.2	2.0	1.8
27.0	--	5.2	2.9	--	5.6	2.9	4.2	4.7	2.6	2.8	2.2	2.1

Measurements were made with large area sources (15 cm x 15 cm) of uniformly distributed activity to compare with study results. These sources were composed of insoluble metallic salts of the isotopes spread over a thin epoxy matrix. The active medium was backed with stainless steel and covered with 9.5 mg/cm^2 Kapton. For instrument A, the correction factor at 1 cm for ^{90}Sr-^{90}Y determined by the measurement system was 2.6, compared to 2.7 from the slab source when based on an extrapolation chamber calibration and 1.8 when based on a TLD calibration. For instrument C, the respective numbers were 2.1, 2.9, and 2.0.

CONCLUSIONS

Most currently available commercial instrumentation does not have the beta measurement characteristics desirable in terms of energy, angular, and source-geometry response. The response of instruments with thinner windows and thinner sensitive volumes represent significant improvements. When beta fields are a problem, instruments that incorporate such features should be considered. Hopefully, current instruments will be modified to improve the beta response using data developed from this study and similar studies [4].

ACKNOWLEDGMENTS

This work was performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

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HIGH SENSITIVE BETA- AND GAMMA-RAY SPECTROSCOPY AND ITS IMPLICATION ON RADIATION PROTECTION

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ABSTRACT

The application of extreme low level detection methods in radiation protection helps not only to reduce counting times but offers also a large variety of investigations which otherwise would not be possible.

In the course of feasibility studies for double beta decay- and solar neutrino experiments the background of Germanium-, NaI(Tl)- detectors, and proportional counters has been systematically studied. It can be characterized by the following components:
A - incompletely shielded external gamma rays
B - intrinsic contamination of detector and shield materials
C - Radon in cavities of the shield and surface deposited Radon daughters
D - muonic events not vetoed by the anticoincidence system
E - cosmic ray-induced radiation (also activation) in detector and shield.

For Germanium detectors and proportional counters the internal contamination could be reduced to such an extent that component D and E became dominant. Since no anticoincidence system is able to completely suppress this two components, the full potential of such detectors can only be exploited by placing them deeply underground.

Measurements with a Germanium spectrometer in a deep salt mine yielded sensitivities of a few μBq/kg for larger samples (up to 10 kg). Miniature proportional counters operated in the Gran Sasso Underground Laboratory are able to detect gaseous activities of less than 1 decay per day.

The advantages of extreme low level counting in health physics are demonstrated by some examples of environmental-, incorporation- and accidental neutron dose monitoring.

A CLASSIFICATION SYSTEM FOR CONTAMINATION LIMITS CONSIDERING THE RADIOLOGICAL RELEVANCY FOR ALL RADIOACTIVE NUCLIDES

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INTRODUCTION

It is a common practice to examine persons and materials with regard to radioactive contamination when they leave restricted access areas.

Starting from a surface contamination limit of 0.037 Bq/cm^2 for persons leaving the restricted access area we have analysed the radiation exposure that may result.

In this connection the following categories of persons are being considered:

- occupationally exposed persons themselves
- persons in the non-monitored areas of the age groups of
  -- 1 year
  -- 10 years
  -- adults

PATHWAYS LEADING TO EXPOSURE

The starting point for the radioactive nuclides are occupationally exposed persons, who leave their place of work daily, carrying a maximum permissible surface contamination of 0.037 Bq/cm^2. For all nuclides this is taken as a basis for calculating the radiation exposure, i.e. we do not differentiate between alpha emitters, beta emitters and gamma emitters.

According to our experience the following pathways have to be taken into account, which lead to exposure of

- occupationally exposed persons: external exposure of their own skin
- occupationally exposed persons: external exposure of their whole body
- occupationally exposed persons: internal exposure through inhalation
- other persons: external exposure to the radiation of a contaminated occupationally-exposed person
- other persons: internal exposure through ingestion

- other persons: internal exposure through inhalation

- other persons: internal exposure via water-soil-plant

The ISH reports Nr. 63, 78, 80 (L1, L2, L3) give dose factors for about 800 nuclides. Nuclides with a half life smaller than 1 hour were omitted. 611 nuclides remained. When calculating the pathway of ingestion we applied the dose factors of the largest resorption.

The radiation exposures calculated for the different pathways were compared with the dose limitations given in the German Radiological Protection Ordinance (L4), where the following limits apply:

50/150/300/600 mSv	for category A occupationally exposed persons and different parts of the body,
1.5 mSv	for other persons outside monitored plant areas according to section 44 (1) of the above Ordinance

SUMMARY

In the vast majority of cases (497 nuclides), exposition through "ingestion by the age group of 1-year-olds" proves to be the leading pathway.

27 nuclides furnish the highest dose values by exposure through "ingestion via water-soil-plant".

In the case of 65 nuclides the pathway of "external irradiation of the skin" of the occupationally exposed person is predominant.

When calculating the ingestion we used the dose factors for the effective dose; the respective dose factors for parts of the body are often ten times higher. With I-129, for example, the dose factor for the thyroid gland is 30 times as high as the one for the effective dose.

As a result of our research the 589 nuclides we investigated into (another 22 isotopes of the inert gases Ar, Kr, Xe, Rn are irrelevant for our consideration) can be divided into categories according to their possible radiation exposure; in doing so, we always refer to the most disadvantageous pathway of exposure for each nuclide in relation to its dose limit. Category I means that the figures reach between 10 % and 100 % of the dose limit. In category II they attain between 1% and 10 %, in cat. III between 0.1 % and 1 %, cat. IV between 0.1 % and 0.01 % and in cat. V they reach less than 0.01 % of the dose limit. Only Cm 250 (spontaneous fission) could exceed the dose limits.

In the following table the nuclides are listed according to

their radiological relevancy in 5 categories as mentioned above.

Category I 10 % - 100 % :

Po 210, Pb 210, Ra 226, Ac 227, Ra 228, Th 229, Pa 231, Np 237, Am 241, Am 243, Cm 245, Cm 246, Bk 247, Cm 248, Cf 249, Cf 251, Cf 254, Am 242 m

Category II 1 % - 10 % :

Cl 36, Fe 60, Sr 90, Nb 95, Tc 98, Cd 113, In 115, I 129, Sm 146, Sm 147, Gd 148, Hg 194, Ra 223, Ra 224, Ac 225, Ra 225, Th 228, Th 230, U 230, Th 232, U 232, U 233, U 234, U 235, Pu 236, Np 236, U 236, U 238, Pu 238, Pu 239, Pu 240, Pu 242, Cm 243, Pu 244, Cm 244, Cm 247, Cf 248, Cf 250, Cf 252, Es 254, Fm 257, Cd 113m, Ir 192m, Bi 210m

Category III 0.1 % - 1 % :

C 14, Na 22, Na 24, Al 26, P 32, K 40, Ti 44, Co 56, Mn 56, Fe 59, Co 60, Ni 65, Zn 65, Ni 66, Se 75, Se 79, Sr 82, Rb 83, Rb 84, Rb 86, Zr 89, Sr 89, Y 90, Y 91, Sr 91, Y 92, Sr 92, Y 93, Zr 95, Zr 97, Mo 99, Tc 99, Rh 101, Rh 102, Ru 105, Rh 105, Ru 106, Cd 109, Ag 111, Sn 123, Sb 124, I 124, I 125, Sn 125, Sn 126, Sb 126, I 126, Te 127, Sb 127, Ba 128, Sb 129, Te 129, I 130, I 131, Te 132, I 132, I 133, Ce 134, Cs 134, I 135, Cs 136, Cs 137, Ba 139, La 140, Ba 140, Ce 141, La 141, La 142, Ce 143, Ce 144, Pr 145, Nd 147, Pm 148, Pm 149, Pm 151, Gd 152, Eu 154, Eu 156, W 188, Os 194, Pb 202, At 211, Pb 212, Ac 226, Th 227, Th 234, Np 238, Np 239, Cm 240, Pu 241, Am 242, Cm 242, Cf 246, Pu 246, Fm 252, Es 253, Fm 255,
Sc 44m, Rh 106m, In 114m, Cd 115m, Te 125m, Te 127m, Te 129m, Te 131m, Cs 134m, Pa 148m, Eu 152m, Lu 177m, Hf 178m, Ir 194m, Es 254m,

Category IV 0.01 % - 0.1 % :

Be 10, Mg 28, Si 32, P 33, Ca 41, K 42, K 43, Sc 44, Ca 45, Sc 46, Sc 47, Ca 47, Sc 48, V 48, Fe 52, Mn 52, Mn 54, Co 55, Ni 56, Ni 57, Co 57, Co 58, Zn 62, Ga 66, Cu 67, Ge 68, As 71, Zn 72, Ga 72, As 72, Ga 73, Se 73, As 74, As 76, Br 76, As 77, Br 82, Sr 83, Sr 85, Zr 86, Y 86, Y 87, Rb 87, Zr 88, Y 88, Nb 89, Mo 90, Nb 90, Mo 93, Nb 94, Nb 96, Tc 96, Tc 97, Rh 99, Pd 100, Rh 100, Ru 103, Pd 103, Ag 105, Pd 107, Pd 109, Sn 110, In 110, In 111, Ag 112, Sn 113, Cd 115, Cd 117, Sb 119, Sb 120, Te 121, Sn 121, Sb 122, Te 123, Sb 125, Ba 126, Ba 131, Cs 132, La 132, Ba 133, Cs 135, Ce 135, Nd 138, La 138, Ce 139, Pr 142, Pr 143, Pm 144, Sm 145, Eu 145, Gd 146, Eu 146, Pm 146, Gd 147, Pm 147, Eu 147, Eu 148, Tb 149, Gd 149, Eu 155s,Eu 150, Pm 150, Tb 150, Tb 151, Eu 152, Sm 153, Tb 153, Gd 153, Tb 154, Eu 155, Tb 156, Sm 156,

Eu 157, Tb 158, Gd 159, Tb 160, Tb 161, Dy 166, Yb 166, Ho 166,
Tm 166, Tm 167, Er 169, Lu 169, Yb 169, Tm 170, Lu 170, Hf 170,
Er 171, Lu 171, Lu 172, Hf 172, Er 172, Tm 172, Lu 173, Tm 173,
Lu 174, Hf 175, Yb 175, Lu 176, Ta 176, Lu 177, W 178, Lu 179,
Ta 180, Hf 181, Re 181, Hf 182, Os 182, Re 182, Ta 182, Ta 183,
Hf 184, Re 184, Ta 184, Ir 185, Os 185, W 185, Ir 186s, Ir 186,
Re 186, W 187, Ir 188, Pt 188, Re 188, Ir 189, Pt 189, Re 189,
Ir 190, Os 191, Pt 191, Ir 192, Os 193, Pt 193, Au 194, Ir 194,
Au 195, Hg 197, Pt 197, Au 198, Au 199, Pb 200, Pt 200, Tl 200,
Tl 202, Bi 203, Hg 203, Pb 203, Tl 204, Bi 205, Pb 205, Bi 206,
At 207, Bi 207, Bi 210, Bi 212, Ac 224, Ac 228, Pa 228, Pa 230,
Th 231, U 231, Pa 232, Pa 233, Np 234, Pa 234, Np 236, U 237,
Am 239, U 240, Am 240, Cm 241, Am 244, Pu 245, Bk 245, Bk 246,
Bk 249, Fm 253, Cf 253, Fm 254,
Zn 69m, Tc 95m, Nb 95m, Tc 97m, Tc 99m, Rh 102m, Ag 106m,
Ag 108m, Ag 110m, In 115m, Cd 117m, Sn 117m, Sb 118m, Sn 119m,
Sn 121m, Te 121m, Te 123m, Ba 133m, Ba 135m, Ce 137m, Nd 139m,
Ho 166m, Lu 174m, Hf 179m, Re 184m, Re 186m, Pt 193m, Hg 193m,
Pt 195m, Hg 197m, Au 198m, Au 200m

Category V < 0,01 % :

H 3, Be 7, F 18, Si 31, S 35, Sc 43, Ti 45, Cr 48,
V 49, Cr 51, Mn 53, Fe 55, Ni 59, Co 61, Cu 61, Ni 63,
Cu 64, Ge 66, Ga 67, Ga 68, Ge 69, Ge 71, As 73, Br 75,
Ge 75, Ge 77, Br 77, As 78, Ge 78, Sr 80, Rb 81, Br 83,
Nb 89, Tc 93, Zr 93, Tc 94, Tc 95, Ru 97, Nb 97, Pd 101,
Ag 103, Ag 104, Cd 107, In 109, In 110, Te 116, Sb 117, I 120,
I 121, I 123, Sn 127, Cs 127, Sb 128, Cs 129, Cs 131, La 135,
Ce 137, Pr 137, La 137, Pr 139, Nd 141, Sm 142, Pm 143, Pm 145,
Tb 147, Nd 149, Eu 149, Sm 151, Gd 151, Tb 155, Dy 155, Tb 157,
Dy 157, Dy 159, Er 161, Ho 161, Dy 165, Er 165, Ho 167, Tm 171,
Ta 173, Hf 173, Ta 174, Ta 175, W 176, Yb 177, Ta 177, W 177,
Yb 178, Ta 178, Ta 179, W 181, Os 181, Re 182, Hf 183, Ir 184,
Pt 186, Ir 187, Re 187, Au 193, Hg 193, Ir 195, Hg 195, Tl 195,
Tl 197, Pb 198, Tl 198, Pb 199, Tl 199, Tl 201, Bi 201, Pb 201,
Bi 202, Po 205, Po 207, Pb 209, Pu 234, Np 235, Pu 237, Am 237,
Cm 238, Am 238, Np 240, Pu 243, Am 245, Cm 249, Bk 250, Es 250,
Es 251, Md 257
Co 58m, Zn 71m, Br 80m, Rb 82m, Sr 85m, Sr 87m, Y 90m,
Nb 93m, Mo 93m, Rh 99m, Rh 101m, In 113m, Sb 116m, In 117m,
I 132m, Pr 138m, Tb 156ms, Tb 156m, Ho 162m, Lu 176m, Ta 180m,
Hf 180m, Hf 182m, Os 189m, Ir 190ms, Ir 190m, Os 191m, Ir 195m,
Hg 195m, Pt 197m, Tl 198m, Pb 202m

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THE WORK OF A PROVINCIAL RADIATION CONTROL UNIT IN CANADA.

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Background.

The Statute of Westminister which established the Dominion of Canada, also identified the areas of responsibility of the federal and provincial governments. Public health and occupational health come in the latter category so that constitutionally the provinces are responsible for radiation safety. This responsibility is however no longer absolute, following the development of the atomic bomb a Federal Atomic Energy Control Act was passed and this gives the federal government control of developments relating to atomic weapons or nuclear energy. In practice this means that all uses of radioactive materials have to be licenced by the Atomic Energy Control Board (AECB) set up under the Act; and that the licence conditions are worded in a way which usually incorporates a considerable component of occupational health related requirements. Today regulatory responsibility is shared between the federal and provincial governments and to date there has been no supreme court decision which clearly identifies the limits of federal and provincial responsibilities. Nevertheless the courts have decided that even where there are valid federal controls, provinces may constitutionally impose more stringent (but not less stringent) controls enforceable within the territory of the province concerned.

For many years after the Atomic Energy Control Act was passed there was little attempt on the part of either federal or provincial governments to control and inspect users of ionizing radiations, and no provincial government enacted specific radiation control legislation. The need for such legislation first became evident in connection with medical uses of radiation particularly diagnostic radioiology. In Saskatchewan an advisory committee was set up in 1959 under Dr Norman Williams to advise the Minister of Public Health on the need for regulations to control hazards arising from the use of ionizing radiations and on the methods which should be used to enforce such regulations. As a result of the report of this committee the first provincial Radiation Health and Safety Act was passed in 1962, and set an important precedent for public controls on the medical uses of ionizing radiations in Canada. The Act required the appointment of the first provincial radiation safety officer, empowered the introduction of radiation control regulations and established an ongoing Radiation Health and Safety Committee to advise the provincial government on all matters relating to radiation hazards and their control. Over the years this committee has done a great deal to shape the radiation safety policies adopted by the province. The provincial radiation safety unit

reports to the committee which is also responsible for periodically reviewing the regulations issued under the act and making recommendations for them to be updated. The Act itself requires all types of radiation emitting equipment to be registered with the provincial government in an analogous way to the federal requirement for the licencing of radioisotope users, sets formal requirements for regular maintenance of all medical radiation equipment and introduces formal training requirements for the operators of this equipment. It has been regularly updated since it was first brought in, the last major revision took place in 1985 and considerably expanded the provisions relating to uses of non-ionizing radiations. Regulations under the Act were first introduced in 1970, primarily to enact the radiation dose limits recommended by the ICRP. Subsequent regulations covered many other areas of concern and last year a comprehensive package of revised regulations, covering all aspects of the use of both ionizing and non-ionizing radiations, was released for public comment.

Medical.

The original responsibilities of the provincial radiation safety unit were primarily directed towards radiation safety inspections of hospital x-ray equipment, and this still remains one of its primary responsibilities as the exposure of patients during medical procedures continues to make by far the biggest contribution to population radiation exposure from man made sources. From the early 1950's onwards such inspections had gradually became general practice in most countries where diagnostic radiology was widely carried out, but initially the function of these inspections was very limited; primarily to look for radiation leakage through the tube housing, check collimation and filtration, and test the adequacy of shielding. Gradually however it became appreciated that protection of patients as well as operators was desirable, and that this involved such considerations as the elimination of unnecessary retake examinations. This has resulted in radiation safety inspections being extended to include checks on many other factors such as the resolution of the system, the speed of the film-screen combination and the optimisation of the processing. Eventually these developments led to recognition that the physicists employed as radiation health inspectors have to provide full medical physics support to all the hospitals which do not have a staff medical physicist on call. Today one of the principal duties of provincial radiation health officers is to help diagnostic radiology departments in small hospitals and clinics to develop mandatory in-house quality assurance manuals that clearly lay down not only the duties of the hospital but also their own role as visiting physicist.

Uranium Mining.

Saskatchewan is a major world producer of uranium from mines with uniquely high grade ores (some localised deposits consist of more than 50% uranium and average ore concentrations of well over 10% are common).

Another major responsibility of the radiation safety unit is therefore to support provincial mines inspectors by visiting uranium mines to enforce both provincial regulations and good working practices. Saskatchewan is the only part of Canada where the internal dose equivalent received from radon daughters and the inhalation of radioactive dust has to be added to the external dose equivalent received from gamma radiation, and the sum has to be less than the permissible occupational dose limit. Provincial radiation health officers are involved in enforcing these dose limits, setting up agreed monitoring schedules for mines and mills, approving proposed work practices, establishing adequate worker training programs and carrying out on-site radiation safety inspections.

Industrial and Educational.

The unit is also involved to a lesser extent with many other industries which employ ionizing radiation techniques. Industrial radiography; density, level and flow gauging; neutron moisture content measurements; tracer studies; oilwell logging and laboratory analysis procedures are well known examples of these techniques. Some of them involve the use of radioisotopes and therefore require Atomic Energy Control Board licences and conformity with A.E.C.B. regulations, others are entirely a provincial responsibility. Staff attached to the radiation safety unit have the responsibility of providing an emergency response service following transport or other radiation emergencies and also participate in education and worker training programs, particularly those directed towards members of the occupational health committees which are mandatory under Provincial legislation in all large workplaces.

Radon.

Today there is increasing public concern about levels of natural radon in buildings and the regulations recently introduced include a requirement that remedial action to correct high radon levels must be carried out in buildings to which the public have access if any person may be exposed to a radon dose exceeding one tenth of the provincial limit for occupationally exposed workers (4 WLM per annum).

Non-ionizing Radiations.

During the last few years the most important change in the work of the unit has been an increasing involvement with problems associated with sources of non-ionizing radiation. The uses of lasers for medicine, industry, education and entertainment is increasing each year and, although there have been few serious accidents, most of these uses involve significant risk of retinal damage. With the proliferation of tanning salons excessive exposures to ultra-violet are becoming increasingly common. Ultrasound and NMR imaging procedures now take place in many hospitals and hearing losses arising from airborne

ultrasound are becoming more frequent. Video terminal operators and persons living near electrical transmission lines are concerned about low frequency R.F. emissions, and catering workers about the leakage of microwaves. Regulations to address all these concerns, and to limit possible exposures of the public, were introduced last year and now have to be enforced by the unit.

Laboratory Facilities.

Until recently the radiation physicists attached to the unit were only equipped with field monitoring instruments but with the increasing scope and complexity of the program for which it is responsible the unit has for a long time felt a need for comprenensive laboratory facilities. The first step in this direction was the introduction of thermoluminescence dosimetry 15 years ago, and the development of new quality assurance procedures for diagnostic radiology necessitated the acquisition of a diagnostic X-ray unit six years ago. However the role of the unit as an enforcement agency for provincial regulations remained severely crippled by lack of instrument calibration facilities until 1985 when the unit moved into a new building in which its laboratory requirements could be accommodated. The facilities now available to the unit occupy about 200 sq. m. and include four principal laboratories. The first contains medical and dental X-ray units and a darkroom with automatic film processors, as well as test benches for the calibration of X-ray, gamma ray and microwave oven monitors (all commercial microwave ovens are checked for leakage by public health inspectors, and the monitors used for this purpose are all calibrated by the radiation safety unit at yearly intervals). The second is basically a radioisotope and wet chemistry laboratory, although it also includes a radon chamber in which the radon monitoring instruments are calibrated. A small room off this laboratory is used for alpha and gamma spectrometry and provides a less intimidating setting for such occasional duties as monitoring the thyroids of travellers returning from Europe immediately after the Chernobyl accident. The third laboratory is an open area containing the units computer system and a thermoluminescece dosimeter reader with annealing ovens. The final laboratory is essentially a service area, this can house any special projects but is set up primarily as a maintenance workshop.

Conclusion.

This paper marks the 25th anniversary of the establishment of the provincial radiation safety unit in Saskatchewan, Canada; and it outlines the way in which the activities and responsibilities of the unit have evolved during this quarter century. Stepping back to review the past in this way is valuable when planning to meet likely future committments during a decade in which it seems likely that public concerns about radiation risks will continue to increase.

EXPERIENCE IN ESTABLISHING AND OPERATING A RADIATION DOSIMETRY AND RADIOACTIVITY CALIBRATION LABORATORY IN THE REPUBLIC OF CHINA

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INTRODUCTION

Increasing public concern and more demanding legislation about radiation safety have made reliable measurement of ionizing radiation a necessity. Standardized calibration and procedures for the measurement of radiation doseage become one of the most important parts to assure measurement accuracy and hence radiation safety. The wareness of this need has led to the establishment of the Radiation Dosimetry and Radioactivity Calibration Laboratory (RDRCL) at the Institute of Nuclear Energy Research (INER). Since July 1976 the RDRCL has been operated by the Health Physics (HP) Division of INER. In setting up and operating the laboratory, several guides(1,2,3) have been followed. The experience we have gained may be of general interest to those whose concern is with the use or calibration of dosimeter.

ORGANIZATION AND RESPONSIBILITY OF THE RDRCL

The RDRCL includs two main parts namely Photon Dosimetry Calibration Laboratory (PDCL) and Radioactivity Calibration System (RCS). The operation of the RDRCL is under overall supervision of the HP Division Director of INER. For rountine calibrating services, the PDCL has two full-time senior physicists well experienced in radiation dosimetry and four full-time technicians with adequate qualification and experience to provide calibrating services. More over, for special consulation or discussion, other senior physicists or engineerers in the HP Division can readily provide help. Since most Radiotherapy centers in Taiwan are located in the northern part and are close to INER. The calibration services for them are convenient. In the past years, the PDCL in addition to discharge served the instrument calibration work for INER, has offered technical service to other public or private companies in Taiwan, e.g. Taiwan Power Company, University of Tsing-Hua, and a good many hospitals.(4)

The main responsibilities to be implemented by PRDL in the laboratory itself could include:

(1). The maintenance of a set of secondary standard (intercomparison) ion chamber and radiation sources.

(2). Calibrating radiation measuring instruments used for clinical dosimetry and radiation protection purposes, and issuing certificates.

(3). The minor repair of instruments when needed.

(4). Coordinating efforts with other HP staffs on personnel and environmental dosimetry services.

(5). Training of health physicists and medical physicists, and paritcipating in training programs sponsored by government.

(6). Providing help to radiation protection services with regard to dosimetric problems.

(7). Joining in dosimetric intercomparisons with other laboratories.

(8). Keeping up-to-date dosimetric measurement methods, and carrying out research on radiation dosimetry.

EQUIPMENT AND FACILITIES OF THE PDCL

(A) X-ray Beams

The PDCL is equipped with two X-ray generating systems. A Baltgraph CE 50/30 type X-ray machine with 50 kVcp X-ray tube incorporating a Beryllium window was installed for low energy work, and a high quality Pantak HF420c calibration type X-ray machine was used for both medium and low energy work. The later can be operated with voltage from 420 to 10 kVcp at tube current down to 50 A. The ancillary facilities include a set of high quality filters, an adjustable diaphragm, a transmission type monitor chamber, etc. The X-ray generating system is capable of generating those filtered X-ray beams as recommended by the National Bureau of Standard (NBS) or as shown in the ISO-4037 report, regarding to the effective energy and dose rate ranges.

(B) γ-ray Sources

To obtain a wide range of γ-energies and outputs, a large number of sealed radioisoiope sources, such as Ra-226, Cs-137 and Co-60, etc. were used to provide precise exposure rate.

The above-mentioned X and γ-photon sources are or can be equipped with calibration benches or stands. Devices for easy and reproducible position, such as trackes, laser beam sources, etc. are available. The remote control systems can be installed whenever necessary.

(C) Ionization Chambers

Ionization chambers calibrated against the primary standards at the NBS or the National Physical Laboratory (NPL) of the U.K. are refered to as the secondary standard chambers at the PDCL. Some secondary standard ion chambers, together with two free-air ion chambers are properly used and maintained. Redundant principle is practised whenever possible.

Ion chamber calibrated against the secondary standard ion chambers are referred to as the laboratory standard instruments which are used to calibrate the field-use dosimeters or survey meters.

(D) Electrometers

There are many feedback type high precision electrometers in the PDCL. Some of them have been calibrated traceable to NBS. Like ion chamber, the electrometers in the PDCL are classfied as secondary standard, laboratory standard and field-use.

(E) Other Equipment

Additional equipment required for ionization chamber calibration include standard charge/voltage sources, three terminal calibration capacitors, high quality barometers, thermometers, timers, humidity measurement instruments, atmospheric communication testing chamber, etc. Many self made devices are installed to position various types of chambers in optimum location with a view to meeting requirements of all radiation charactistics so that most accurate results are obtained based on which to determine the realistic condition.

OPERATION OF THE PDCL

In order to fulfill PDCL's responsibities and to provide for the soonest possible instrument calibration, a disciplined procedure is established to follow. The protocol of the PDCL has described in detail the procedures for calibrating, reporting and record keeping for each class of device. Also classes are defined into which each device will fall. These procedures have also delineated explicit steps to assure utmost accuracy in calibration. All calibrating activities have to be in conformity with laboratory protocol.

RADIOACTIVITY CALIBRATION SYSTEM

According to the characteristics of the radioactivity, e.g. radiation types, physical and chemical forms, and intensity, there are many different qualitification and quantification methods in determining the activity of a radioisotope.

The RCS has had many activity measuring systems which mainly include alpha spectrometric system, 2π-α counter, 2π-β counter liquid scintillation spectrometric system, gamma spectrometric system, 4π-γ ionization chamber, and 4π-β, γ coincidence counting system.(5)

EXPERIENCE WITH THE RDRCL

Experiences gained from establishing and operating the RDRCL

in these years show that the calibration of radiation measuring instruments is a high-tech endeavor and rather costly work. It is important that the personnel selected for establishing and operating the RDRCL are adequately qualified and are given adequate status and remuneration. It also needs to invest large fund for equipment and site preparation. A rigid protocol and a list of operational procedures must be clearly defined and strictly followed. The supports from other relevant high technical industries and the close contact with other national laboratories are essential parts to elevate the level of the laboratory.

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PERFORMANCE CHARACTERISTICS OF SELECTED "BIODEGRADABLE" LIQUID SCINTILLATION COCKTAILS.

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High efficiency, low background, and quench resistance are primary considerations in the selection of a liquid scintillation (LS) cocktail, along with stability of the cocktail with time and reproducibility of the results obtained. These criteria are often compromised to acquire aqueous sample compatibility, or to avoid adverse flammability or toxicity characteristics. Disposability may also affect cocktail selection, since some low level radioactive waste burial sites are refusing to accept toluene- or xylene- based LS wastes, and other solvents are being considered for exclusion. Certain commercially available cocktails purport to circumvent this disposal problem by being "drain disposable" with local approval.

Efficiencies, backgrounds, quench resistance, and sample stability for ten of these cocktails are presented below. Since many LS counters are calibrated to commercially available toluene based standards prepared to American National Standards Institute (ANSI) specifications, data is also presented for one of these formulations (1). The aqueous sample capacities and the effects of sample volume on efficiency and sample stability are presented by other authors or are readily available from the manufacturers, and vary with the nature of the aqueous material (2).

The tested cocktails consisted of Ready Safe (Beckman Instruments Inc.), Ecolume (ICN Biomedical, Inc.), Solvent Free (Isolab, Inc.), Ecoscint (National Diagnostics), Optifluor, Optifluor-O, and Polyfluor (Packard Instrument Company), Biosafe and Biosafe II (Research Products International), and Ecolite (WestChem, Inc.). These cocktails were compared to spectrophotometric grade toluene (Mallinckrodt Chemical Works; SpectrAR) with 3.92 g/L PPO and 0.08 g/L bis-MSB (Omnifluor; NEN-Dupont), since this is comparable to the ANSI standard referenced above.

A Packard Model 300 C/D LS counter was used for data acquisition. The external standard ratio was used for quench determination. Both large diameter (20 ml) and mini (7 ml) vials composed of either low potassium glass or polyethylene (PE) were used. In order to minimize external effects from adapters, a grommet type adapter system was utilized with the mini vials. This system consists of a plastic ring attached to each extremity of the mini vial, and has no effect on the shape of mini vial quench curves, which are diameter dependent, but unaffected by vial or adapter materials. It does, however, slightly improve the unquenched efficiency for mini vials, while lowering background from the adapter system and facilitating static removal (3).

In order to determine the backgrounds on the cocktails, three replicates of each vial/cocktail combination were counted ten times each for ten minutes. As indicated in Table I, the backgrounds were typically greater than that of the toluene formulation. The Optifluor-O backgrounds were comparable to the ANSI toluene reference, but this cocktail lacks aqueous sample capabilities. The detergent-based Biosafe showed high backgrounds in all vials prepared from three different

sample bottles, which were not seen in the solvent-based Biosafe II. The most probable explanation for these excessive backgrounds is the presence of a very long lived chemiluminescent material (4). Lowest backgrounds were noted in PE mini vials employing grommet type adapters.

TABLE I COCKTAIL BACKGROUNDS

COCKTAIL	Glass mini vials	Plastic mini vials	Glass large vials	Plastic large vials
(For cocktails available as of February, 1986)				
Toluene	18.3±1.1	12.3±1.2	22.0±1.6	11.9±0.9
Optifluor	17.0±1.0	13.4±1.2	23.6±1.3	13.1±1.7
Ecolite	17.2±1.3	13.7±1.7	25.3±1.5	13.8±0.9
Ecoscint	19.4±1.9	14.9±1.3	35.4±1.8	21.5±1.8
Ecolume	22.9±1.5	18.5±1.0	40.9±1.8	30.6±1.9
Solvent Free	19.2±1.6	14.4±0.7	30.2±2.1	18.9±0.9
Biosafe	75.5±3.4	71.6±2.1	595 ± 12	668 ± 34
(For cocktails available as of June, 1987)				
Toluene	16.6±1.2	12.7±0.9	20.0±1.6	12.7±1.1
Ready Safe	18.5±1.6	12.8±1.4	25.7±2.1	16.2±2.0
Biosafe II	17.5±1.1	12.7±0.9	22.8±1.1	14.9±1.3
Polyfluor	17.9±1.8	12.5±0.8	22.4±1.7	13.6±1.4
Optifluor-O	16.8±1.2	11.7±1.4	16.8±1.2	11.7±1.4

TABLE II COCKTAIL TRITIUM EFFICIENCIES

COCKTAIL	Glass mini vials	Plastic mini vials	Glass large vials	Plastic large vials
(For cocktails available as of February, 1986)				
Toluene	58.3±0.9	59.0±0.2	57.8±0.5	59.6±0.2
Optifluor	45.1±0.3	45.7±0.2	44.5±0.4	46.1±0.1
Ecolite	42.6±0.4	43.3±1.4	40.3±0.3	46.6±0.4
Ecoscint	46.4±0.2	47.8±0.9	44.3±0.7	50.0±1.6
Ecolume	45.0±0.5	46.3±0.2	41.5±1.1	46.7±1.3
Solvent Free	29.1±0.1	26.7±0.3	27.1±1.7	21.4±0.7
Biosafe	26.6±0.2	22.1±0.7	21.4±0.1	23.6±0.6
(For cocktails available as of June, 1987)				
Toluene	60.7±0.4	60.6±1.1	59.7±0.2	60.0±1.1
Ready Safe	49.5±0.3	52.9±0.3	49.0±0.8	52.3±0.3
Biosafe II	40.8±0.2	43.0±0.5	41.8±0.2	43.2±0.1
Polyfluor	43.2±1.2	43.6±1.6	42.0±0.8	44.7±0.7
Optifluor-O	57.9±0.3	57.9±0.2	58.3±0.7	57.4±0.5

After the background was determined, 30 μL [methyl ^{3}H]thymidine (15.8 Bq/μL) or 150 μL ^3H_2O (22.5 Bq/μL) was added to all of the cocktails except the toluene samples, which received 30 to 100 microliters of tritiated toluene (32.7 Bq/μL). After mixing, the three replicates of each vial/cocktail combination were counted twice for five minutes each to determine the maximum counting efficiencies for tritium (Table II). The

Optifluor-O showed the best maximum efficiency, with the caveat that it is not designed for aqueous samples. The Ready Safe formulation performed best for this type of sample. Optimum counting conditions for these cocktails were obtained in mini glass vials equipped with grommet adapters, or in large PE vials.

Quench curves were prepared by counting the sample vials for 10 minutes and noting the quench number and counts per minute (CPM). A quantity of nitromethane was added to the vial, which was then recounted for an additional 10 minutes. This procedure was performed 10 times until the vial was heavily quenched. Use of the tested cocktails in equipment calibrated for DPM determination with ANSI toluene standards resulted in 4-8% errors for all non-toluene cocktails during efficiency determinations. These errors were reproducible and cocktail specific, and are summarized in other work (5).

In order to provide an indication of quench resistance, the logarithm (ln) of the efficiency was plotted versus the volume of nitromethane for each cocktail. Slopes were then calculated on these curves for each vial/cocktail combination. As shown in Table III, the detergent based cocktails demonstrated the best resistance to nitromethane quenching, followed by the solvent-based formulations. While color (optical) quenching was not addressed in this experiment, the differences between chemical and optical quench curves noted by others probably apply (6).

TABLE III SAMPLE STABILITY AND QUENCH RESISTANCE

COCKTAIL	Glass mini vial	Plastic mini vial	Glass large vial	Plastic large vial	Q curve slopes
(For cocktails available as of February, 1986)[1]					
Optifluor	----	----	1.4%	31.3%[2]	-1.7×10^{-2}
Ecolite	----	----	0.0%	2.0%	-1.5×10^{-2}
Ecoscint	----	----	+0.9%	5.8%	-1.4×10^{-2}
Ecolume	----	----	+2.0%	1.7%	-1.4×10^{-2}
Solvent Free	----	----	+5.0%	4.5%	-9.5×10^{-3}
Biosafe	----	----	8.8%	2.8%	-8.8×10^{-3}
(For cocktails available as of June, 1987)[3]					
Toluene	2.7%	49.2%	1.0%	14.8%[4]	-3.7×10^{-2}
Ready Safe	0.6%	2.0%	0.1%	1.1%	-1.4×10^{-2}
Biosafe II	1.0%	1.5%	0.6%	0.8%	-1.5×10^{-2}
Polyfluor	1.3%	1.6%	0.7%	1.4%	-1.9×10^{-2}
Optifluor-O	1.4%	7.9%	0.8%	0.8%	-2.7×10^{-2}

(1) % loss of original CPM in 28 days.
(2) Thick wall Polyethylene vials.
(3) % loss of original CPM in 19 days.
(4) Thin wall Polyethylene vials.

To measure cocktail stability with time, samples of each cocktail were counted in both glass and PE vials at intervals throughout a one month period. Three replicates of each cocktail were counted for ten minutes each, averaged, and expressed as a percentage of their original counts per minute. Loss of activity was apparent for several samples in

PE vials (Table III). The toluene cocktail lost both volume and activity at rates related to the thickness of the polyethylene vial walls. Optifluor lost activity at the rate of approximately 10% of the initial cpm every 8 days, and retained only 70% of the original counts at 30 days. No volume losses were apparent. This activity loss, noted by others, was not seen in a different lot number of Optifluor provided by Packard in response to the above observation.

In conclusion, selection of any of these "drain disposable" cocktails will result in a degradation of performance for unquenched to moderately quenched tritium samples, relative to a toluene reference standard. However, since many other commercial cocktails trade maximum theoretical performance characteristics for aqueous sample holding capabilities, this may not be a serious disadvantage. Of greater significance, however, the quench curves for these cocktails vary sufficiently from the toluene quench curve that 4-8% errors will be introduced through use of toluene tritium standards for DPM operation. Also, the detergent based cocktails show significant handling shortcomings relative to the other preparations. These cocktails are quite viscous, and require repeated vortex mixings and long dark adaptation periods. The possible presence of long lived chemiluminescent components may affect suitability of these cocktails for many applications.

Finally, although most of these commercial vendors present data stating that their cocktails are "drain disposable", and that tritium and carbon-14 samples may be disposed of via the sanitary sewer system if they meet de minimis guidelines (7), it should be noted that drain disposal is also governed by local regulatory agencies. No criteria were evaluated as to the suitability of a particular cocktail for disposal into the sanitary sewer system.

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A NEW CALIBRATION SOURCE USING A THIN ION-EXCHANGE MEMBRANE.

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INTRODUCTION

In order to calibrate radiation protection instruments with large area detector such as floor contamination monitor, hand-foot-cloth monitor, large area calibration source with good uniformity is necessary. And accompanying with the dismantlement of reactor, various kinds of calibration sources shall be also needed for radiation detectors which are used for measuring the radioactivity of various shapes of solid wastes or contaminated matters.

Thin ion exchange membrane source is useful as such calibration sources by the following reasons : 1) it can be produced easily, 2) large area source with good uniformity can be obtained and 3) its shape can be changed easily and freely.

A thin ion exchange resin membrane source was introduced by Ballard[1] to calibrate contamination instruments. In JAERI, a thin large area plane source with good uniformity was independently developed using a different type of ion exchange membrane. This paper describes the production method and the characteristics of the ion exchange membrane source.

PRODUCTION OF MEMBRANE SOURCES

1) Ion exchange membrane[2),3),4)]

The ion exchange membrane used for producing calibration sources was developed in JAERI and is called R-27. The membrane has carboxyl as radical in the base material polyethylene so that it is adaptable for any radioactive cation. The distribution of radicals in the membrane is quite uniform and the dispersion of membrane thickness is small. The membrane is chemically stable and it is also proved that the membrane has still sufficient strength when it is irradiated with γ-rays of approximately 10^5 Gy. Therefore, the membrane is quite suitable for producing practical calibration sources.

2) Production method

The ion exchange membrane sources are produced as follows: The ion exchange membrane is first soaked in a radioactive solution which is prepared by dissolving standard radioactive solution (obtained from LMRI or Amersham Co.) in pure water. And some non-

radioactive carrier was added to improve the uniformity. A buffer solution is also added into the solution so that PH value of the solution may not vary so remarkably. After being left in the radioactive solution, the membrane is then rinsed with pure water and dried naturally.

Several types of sources were trially produced by this method. The nuclides of the produced sources are as follows: ^{90}Sr-^{90}Y, ^{204}Tl, ^{147}Pm (pure β emitters), ^{106}Ru-^{106}Rh, ^{60}Co, ^{137}Cs (β and γ emitters), ^{241}Am (α and γ emitters). The shape of the produced sources are circle 50 mm or 100 mm in diameter and rectangle 100 mm $\times$ 150 mm.

3) Determination of radioactivity

The radioactivity of the pure β membrane source was determined using a proportional counter calibrated by a thin paper damped with radioactive standard solution. The radioactivity of membrane sources emitting γ-rays was measured with Ge(Li) detector or NaI(Tl) detector. Counting efficiency of the detector was determined by the measurement in the source geometry of plane and point[5]. This is because any point source can be calibrated using a standard point source.

After determination of the radioactivity, the membrane source is mounted on a suitable supporting tool.

CHARACTERISTICS OF THE MEMBRANE SOURCE

Characteristics of the membrane source was investigated for trially produced sources.

1) Uniformity of the radioactivity on the membrane source

The uniformity of the radioactivity on the membrane sources was checked for all produced membrane sources by the autoradiography method using large area industrial X-ray film. The optical density of the X-ray film irradiated by the membrane source contacting directly with the film was measured with densitometer Macbeth TD-504. It was found that the variation of optical density was less than 5 % from the average. Therefore, the uniformity of radioactivity on the produced membrane source was proved to be practically sufficient. The uniformity of the membrane source produced using radioactive solution without carrier was poor and so it was found that non-radioactive carrier is quite important factor to produce uniform membrane sources.

2) Self-absorption effect of the α membrane source

^{241}Am α source was first produced using 25 μm ion exchange membrane. The α energy spectrum of the source was measured with Si-SBD and it was found that no α peak of 241Am (5.486 MeV) is observed because of self-absorption effect. Then an extremely thin

(1 μ m) membrane was used for producing ^{241}Am α source. The spectrum of the source was also investigated. As shown in Fig.1, the spectrum of 1 μ m α source is similar to that of a standard ^{241}Am α source (manufactured by LMRI). 1 μ m α source, however, is rather difficult to handle because of its fragility and so more strong membrane is desirable for practical purposes.

Another type of ion exchange membrane, which has 100 μ m total thickness with several μ m radical grafted surface, was also tried to be used for producing ^{241}Am α source. The spectrum of this source was compared with the standard ^{241}Am α source and was found to be in good agreement with that of standard source (See Fig.2). Therefore, up to now, this 100 μ m ion exchange membrane grafted superficially can be regarded as the most suitable ion exchange membrane to produce practical α sources.

3) " Wipe-off " factor of the membrane source

The " wipe-off " factor, which is quite important for practical calibration source, was also checked by the following experiment. After rubbing the surface of the ^{60}Co membrane source (50 μ m thick) several ten times with smearing filter paper, the source and the filter paper were both measured with GM counter. No difference was observed in the counting rates of them before and after rubbing. This experiment result shows that the " wipe-off " factor of the membrane source is negligibly low.

CONCLUSION

It was found that a new calibration source using a thin ion exchange membrane R-27 developed in JAERI satisfies the uniformity of radioactivity on the membrane with the variation of less than 5 %. Superficially grafted 100 μ m ion exchange membrane was found to be the most suitable to produce practical α source because it has no fragility and causes self-absorption effect very little. And the " wipe-off " factor of the ion exchange membrane source was also found to be negligibly low.

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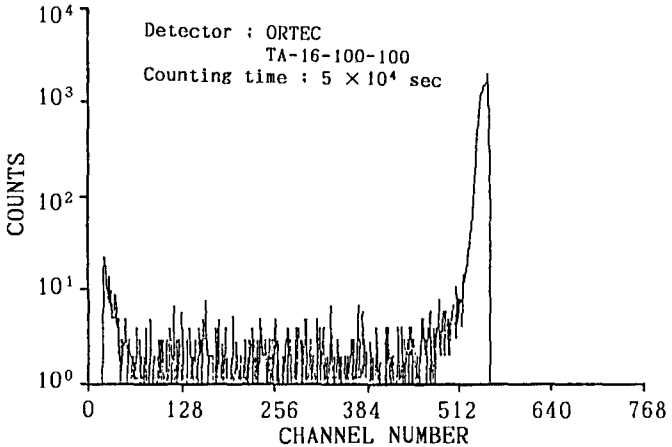
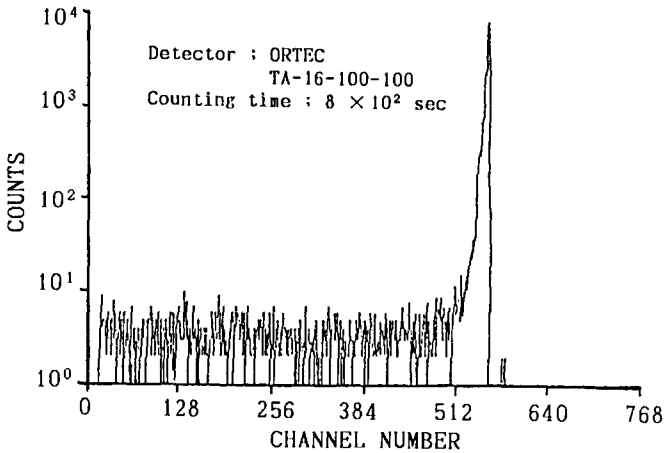
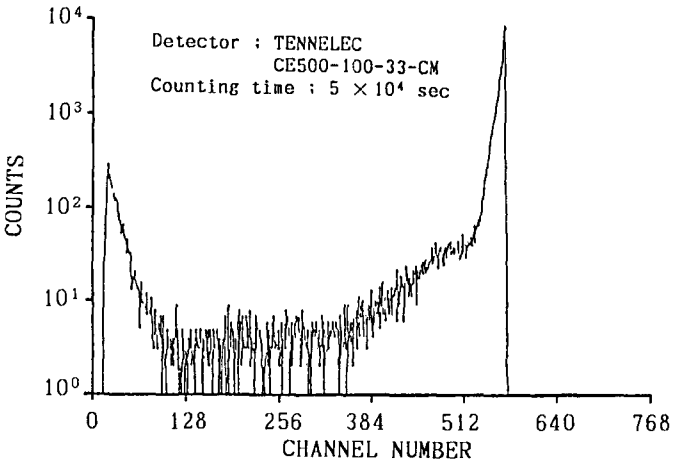


a) 1 μm ion exchange membrane source

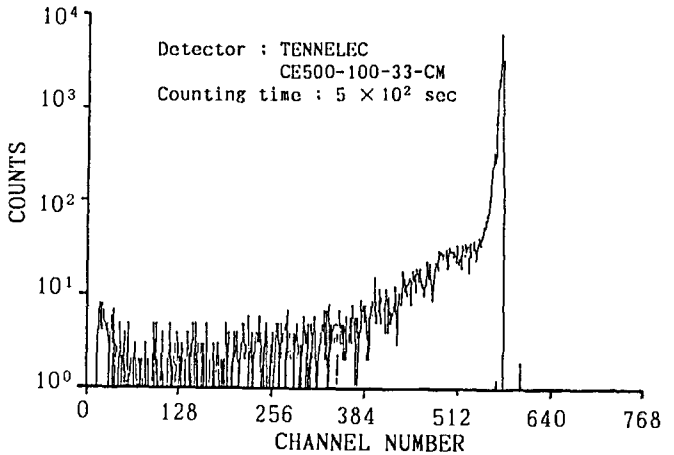


b) Standard ^{241}Am α source (LMRI; EAP-4)

Fig.1 Comparison of ^{241}Am α spectrum between 1μm ion exchange membrane source and a standard source.



a) Superficially grafted ion exchange membrane source (100 μm thick)



b) Standard ^{241}Am α source (LMRI; EAP-4)

Fig.2 Comparison of ^{241}Am α spectrum between superficially grafted ion exchange membrane source and a standard source.

DETECTION LIMITS OF GAMMA-RAY SPECTROSCOPY SYSTEMS USED FOR RADIOASSAY

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ABSTRACT

The ability of a radioassay system to quantify the activity in a given sample is determined by the counting time, background conditions, and detection efficiency. The detection limit of a radioassay system can be specified in a number of ways, each of which requires that the measurement process be strictly defined and controlled. Since spectroscopy systems used for radioassay of gamma-ray emitters present unique "Background" conditions with each spectrum, the requirement of strict background quality control is not met. A comparison of methods currently used to calculate detection limits for gamma-ray spectroscopy systems has been performed. The appropriateness of the use of each method has been evaluated with a series of controlled experiments. A method to properly specify and determine the detection limit for a gamma-ray spectroscopy system has been demonstrated. This method provides a more appropriate technique for determining the detection limit and, therefore, has a significant impact on regulations and instrumentation for both radiobioassay and assay of environmental samples for gamma-ray emitters.

A DOSE-EQUIVALENT RATE METER FOR ENVIRONMENTAL RADIATION SURVEYS

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INTRODUCTION

In 1969 a scintillation dose rate meter was developed |1| which is well suited for measuring the exposure rate for photon energies above 20 keV. The air equivalent response of the scintillation detector is achieved by coating the plastic scintillator (NE 102A) in the probe with a thin layer of ZnS |1|, |4|.

In report 39 |2| ICRU recommended new quantities for measurements in radiation protection with external sources. For environmental monitoring, these quantities are the ambient dose equivalent H*(10) and the directional dose equivalent H'(0.07). If these new quantities are introduced for practical measurements in radiation protection, instruments must be modified to directly indicate the ambient dose eqivalent or the directional dose equivalent.

In 1986 the PTB began carrying out experimental investigations to ascertain whether the response of the scintillation detectors could be modified to obtain an indication of the ambient dose-equivalent rate independent of the photon energy. At the same time the electronics of these instruments was improved to overcome constructional shortcomings. The dependence of the reading on the temperature was reduced electronically and the influence of the dark current of the photomultiplier on the more sensitive ranges was diminished.

INDICATION OF THE AMBIENT DOSE-EQUIVALENT RATE WITH A SCINTILLATION DETECTOR

An energy independent reading of the ambient dose equivalent rate with a scintillation dose rate meter is achieved if the quotient of the output current of the photomultiplier and the ambient dose equivalent rate is independent of the photon energy.

As shown in |1| the response of the scintillation detector as a function of photon energy can be influenced by coating the plastic scintillator in the probe with a thin layer of zinc sulfide or other inorganic scintillators of appropriate thickness.

This technique is also applied to obtain the scintillation detector's response which is needed for an energy independent reading of the ambient dose equivalent rate. Research work has been carried out on the instrument which is suitable for measuring radiation levels in the order of those in the natural environment.

multiplier is high and the dark current is in the order of the signal current. This current also depends on the temperature and increases with increasing temperature.

To eliminate the dependence of the signal on the temperature, the gain of the photomultiplier must be controlled by the temperature of the photomultiplier-scintillator combination. The contribution of the dark current to the signal current can be reduced by subtracting its value.

All information needed for controlling the high voltage and the gain of the photomultiplier in all ranges as a function of temperature is stored together with information on the corresponding dark current in an EPROM. For each range and each measured temperature of the detector between $-20^{o}C$ and $+40^{o}C$, this information is retrieved from the memory and used to control the high voltage supply and to subtract the corresponding value of the dark current from the signal current.

As it is impossible to measure the family of characteristics to be stored in the memory in a reasonably short time, P. Seyfried has developed a special algorithm |6| which allows the whole family of characteristics to be computed from a few measured values.

The first instrument equipped with this new system has shown temperature-dependent deviations in all ranges not exceeding a maximum value of + 5%. This could be improved by using more measured values to compute the family of characteristics.

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|2| ICRU-Report 39, 1985: Determination of Dose Equivalents for External Radiation Sources. ICRU, Bethseda, Ma. 20814, USA

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|5| Großwendt, B.; Hohlfeld, K.; Kramer, H.M.; Selbach, H.-J.: Konversionsfaktoren für die ICRU-Äquivalentdosisgrößen zur Kalibrierung von Strahlenschutzdosimetern, PTB-Report, PTB-Dos-11, 1985.

|6| Seyfried, P.: Publication in preparation

DESCRIPTION OF GENERAL CIRCUIT

A schematic diagram of the dose rate meter, which consists of two parts, the detector probe and the control unit, is shown in Fig. 2. The instrument has 8 ranges from 30 nSv/h up to 100 μSv/h full scale which are selected either by the autoranging mode or by the manual mode. To avoid overloading of the photomultiplier, the output current of each range is limited to a maximum value of 3 nA by setting the high voltage of the photomultiplier to an appropriate value. This signal current, which is proportional to the dose rate, is indicated on the meter in the control unit. Selectable time constants are provided.

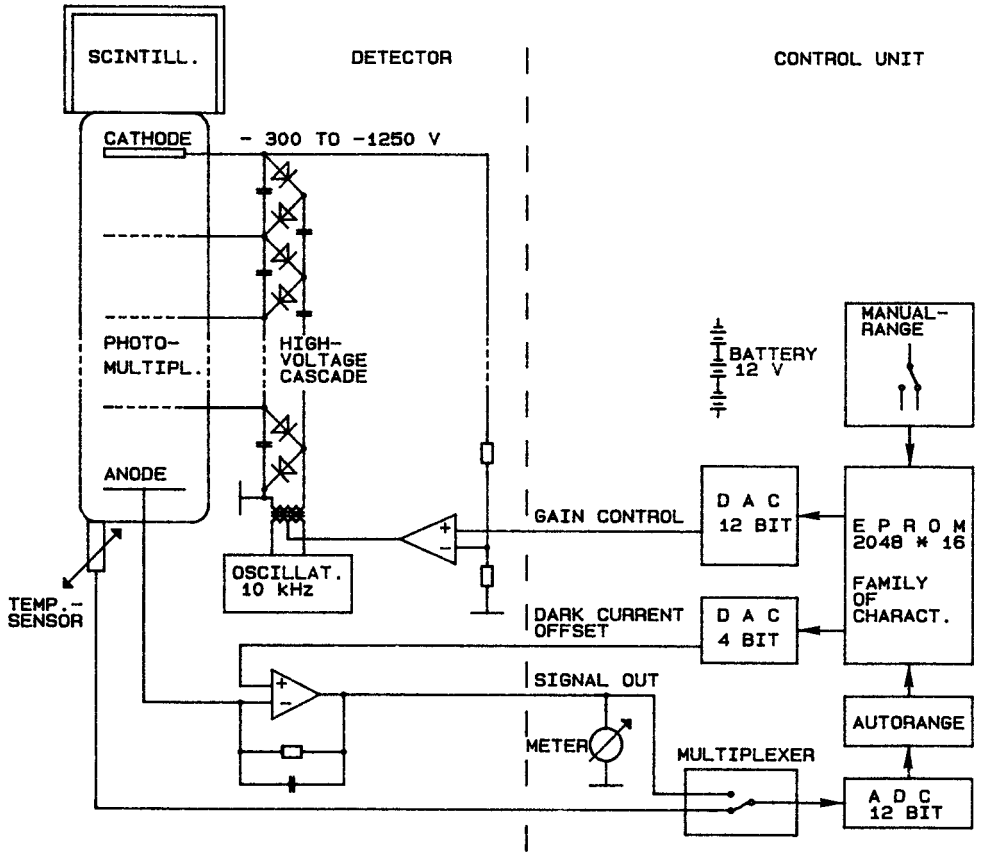

Fig. 2 Schematic diagram of the dose-equivalent rate meter

There are two disadvantages in this operation mode of the photomultiplier. With increasing temperature the signal current decreases for a fixed dose rate, and not only the amplification of the photomultiplier but also the fluorescence yield of the scintillator is dependent on the temperature. Besides this, the dark current of the photomultiplier may contribute to the signal current in the more sensitive ranges where the gain of the photo-

This instrument has a cylindrical plastic scintillator in the probe which is 75 mm in diameter and 75 mm in height |3|, |4|. In the course of the work various mixtures of inorganic scintillators containing ZnS, $CaWO_4$ and Gd_2O_2S were used to coat the cylindrical scintillator to obtain the desired response. The response of the scintillation detector was measured with X-reference radiation of series A |5| and γ-radiation of caesium-137 and cobalt-60. Dividing the readings of the dose-rate meter by the ambient dose equivalent rates at the same place in the calibration field yields the response of the instrument as a function of the photon energy. The ambient dose equivalent rate in the calibration field was calculated from the measured photon dose equivalent rate using the conversion factors given in |5|. The best approximation which we achieved to the desired response is shown in Fig. 1. From 20 keV up to 1.25 MeV the reading is within +4% and -20%, independent of the photon energy, normalized to the response at an energy of 662 keV.

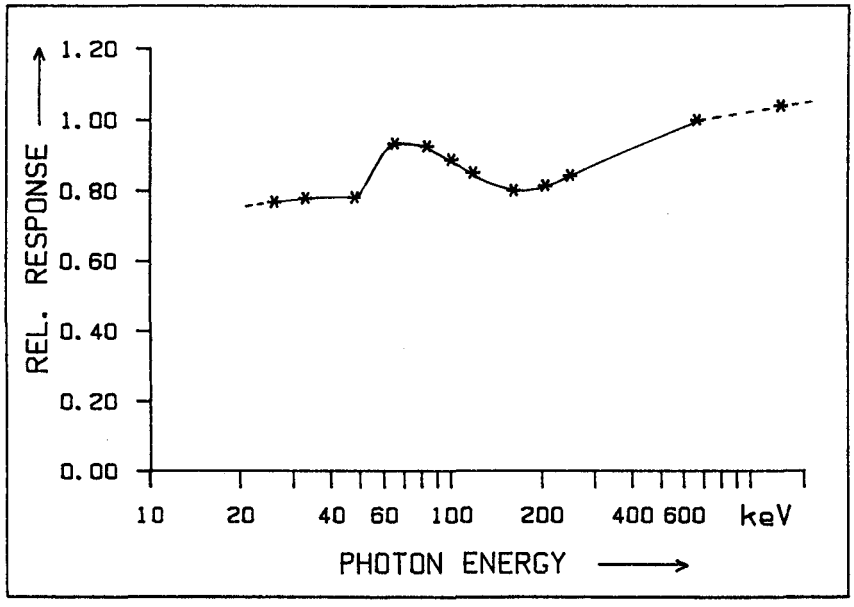

Fig. 1 Relative reponse of the dose-equivalent rate meter as function of the photon energy

As pointed out in |1| the decay time of phosphorescence after irradiation of the inorganic scintillators with X-rays should be within the order of a few seconds. Measurement have shown that 1 minute after irradiation, the yield of the phosphorescence is about $5 \cdot 10^{-4}$ of the intensity of the fluorescence. In this case the influence of the phosphorescence on the reading can be neglected shortly after irradiation.

MEASUREMENTS OF PHOTOELECTRONS FROM A PHOTOMULTIPLIER PHOTOCATHODE AND THEIR APPLICATIONS TO DOSIMETRY

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INTRODUCTION

In gaseous ionization detectors, W-value defined as an average energy expended per ion-pair, plays an important role for the measurement of absorbed dose. However, there does not exist the same kind of quantity, Wp, which should be defined as an average energy expended per emitted photon, in the case of scintillators. It is very difficult to absolutely measure such a quantity with present techniques using of scintillators and available photomultipliers (PMT). In the PMT, it is quite difficult to measure the collection efficiency of photoelectrons at the first dynode and the multiplication gain of successive dynodes chain, although the quantum efficiency of the photocathode is measurable. On the other hand, the number of photoelectrons from the photocathode is easily measurable with a well calibrated charge sensitive amplifier. We recently measured numbers of photoelectrons with three combinations of PMT and NaI(Tl) for the first time(1). The PMT was operated as a photodiode (PD-mode). The average energy expended per collected photoelectron, Wpe, was almost 100 eV in all the three combinations. Here, we describe the measurements of photoelectrons with a NaI(Tl) coupled to a PMT. Also we show the application of this technique to radiation dosimetry using a plastic scintillator coupled with a PMT and a charge integrator.

NUMBER OF PHOTOELECTRONS

Measurements of the photoelectrons from the photocatode were made using a 3"o X 3" NaI(Tl) scintillator coupled to a PMT. In the PD-mode, the first dynode (Dy1), second one (Dy2) and focusing grid (G) were connected together as a collector to collect the photoelectrons, while the photocathode (K), the anode (A) and all the other dynodes were connected as a cathode as shown in Fig.1. A low noise charge sensitive preamplifier (CSPA) was connected to the collector with a charge terminator which was used to calibrate the amplifier system by a high precision mercury pulser. The signal from the CSPA was fed to a main amplifier for shaping and further amplification. The semi-Gaussian shaping was made with differentiating and integrating time constants of both 2 usec. The amplifier system with a charge terminator was calibrated with the method used in the previous experiments(1). The complete collection of photoelectrons was

assured by measuring the pulse heights of the photo-peak due to the gamma-rays from Cs-137 as a function of applied voltage to the cathode. The pulse height of the photo-peak was completely saturated above -20 volt as shown in Fig.2. The energy dependence of pulse heights and resolutions was measured with the gamma-ray sources of Cs-137, Mn-54, Na-22, Co-22, K-40, Tl-208, and Pu-C. Also, these measurements were repeated with a same combination of a PMT and a scintillator in the normal operation mode of a PMT (PMT-mode). The two typical pulse height distributions are shown in Fig.3. Figure (A) and (C) are gamma-ray spectra from Co-60 and Pu-C with the PMT-mode, and (B) and (D) with the PD-mode, respectively. The pulse heights were determined by least-square fitting of measured pulse height distributions to a Gaussian plus a quadratic function or two Gaussians. The converted pulse heights to the number of photoelectrons were fitted against the energy of gamma-rays by the least-squares fit. The number of photoelectrons was obtained to be (9.94±0.005) electrons/keV in this combination. The Wpe was 100.6 eV. This result excellently agreed with the previous results(1).

ENERGY RESOLUTIONS IN PD-MODE AND PMT-MODE

In the case of the PD-mode, the PMT is operated without the multiplication of electrons at the dynode chain and the measurements are made with a high gain charge sensitive amplifier system. Therefore, the signal to noise ratio is an important factor to limit the experiments. Two typical examples measured with the PD-mode and the PMT-mode are shown in Fig.3. The energy resolutions of the PMT-mode are slightly better than those of the PD-mode at the gamma-rays from Co-60, but are almost comparable at about 6 MeV (Pu-C). The noise width obtained from the pulse height distributions of the test pulse well agreed to be 420 electrons in the full width at half maximum(fwhm) as a noise equivalent charge(2). The noise width mainly depends on the input capacitances of the PMT and the amplifier, and also on the leakage current of the PMT.

APPLICATION TO DOSIMETRY

Plastic scintillators (PS) are similar to tissue in composition and density, and are also itself radiation detectors. The PS had been used to measured the tissue equivalent absorbed dose. In that case, the measurements were made by the PMT-mode with a combination of a PS and a PMT. The combination was calibrated with the gamma-rays of the known energies. The measurements of the absorbed dose were made firstly by taking a pulse height distribution with a pulse height analyzer, secondly by calculating the total pulse heights weighted with counts per channel, and thirdly converted the total pulse heights to the absorbed energy using the energy per channel. However, the method described here may be much easier, since the absorbed energy is simply calculated from N X Wpe, where N is the total number of photoelectrons. The experimental apparatus used in the present measurements is shown in Fig.4. A PS (NE102A), of which the diameter is 3 cm and the thickness is 1 cm, was used to

measure both Wpe and energy depositions. A PMT (R1307, Hamamatsu Photonics Co.) was coupled to the PS as a photo-diode. The PS was wrapped with aluminized mylar sheet as the reflector of photons. In order to measure the Wpe, the collector was connected to a low noise charge sensitive amplifier system (A) at first. The Wpe of the combination was measured to be 430±20 eV with the gamma-rays of the known energies. In order to measure the absorbed energy, the collector was connected with a charge integrator (B), while the cathode was held at -50 volt. A well calibrated capacitor (C) was 9.52 pF. The charge of photoelectrons was integrated for a period (Ti) and discharged periodically by a relay. The output waveform was recorded on a strip-chart recorder. The total charge was calculated from the maximum voltage (Eo). The total integrated charge due to background radiations and leakage current between the cathode and the collector was measured to be (0.67±0.1) pC for the integration time of 2 sec. A Co-60 gamma-ray source of 7.9 mCi was used for irradiation of the PS. The measured net charges at 29.7 cm and 139.7 cm were (10.9±0.1) and (0.53±0.01) pC for 2 sec, respectively. These are easily converted to absorbed dose rates with the Wp and are 115 and 5.4 mrad/h, respectively. Also the absorbed dose rates at several points between above two points were measured. The dependence of those on the distance between the source and the PS was assured to be well fitted to the inverse square law.

DISCUSSION

The preliminary results seem to suggest that this method is effective to the absolute measurement of tissue equivalent dose with the PS. In this method, a high voltage power supply, which is normally used to operate a PMT, is not necessary and the collection efficiency of photoelectrons does not depend on the fluctuation of biasing voltage. These also suggest a possibility making of a small size dosimeter without a high voltage power supply. In this experiment a charge integrator is used to measure the total charges of collected photoelectrons, but it is also possible to use pulse techniques to get the total charges. There may exist two faults in this method. The Wpe of the PS was determined with the Compton edges of the gamma-rays with the known energies. The standard deviation of the Wpe for the PS was large due to uncertainty in determining the Compton edges, even though the low noise charge sensitive amplifier, of which the noise width was 495 + 2 electrons in fwhm, was used for the measurements. Furthermore, the number of photoelectrons in the PS was about one fourth of that in NaI(Tl). However, the uncertainty in the Wpe is possibly avoidable by using several monochromatic electron sources for the energy calibration. In the latter case, photo-detectors with much higher quantum efficiency is desirable.

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Fig. 1

Schematic diagram of the experimental equipment

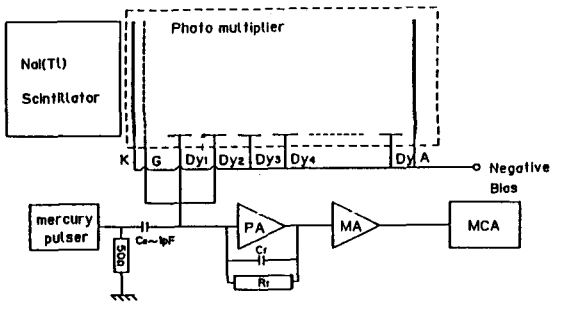


Fig. 2

Saturation characteristics in PD mode

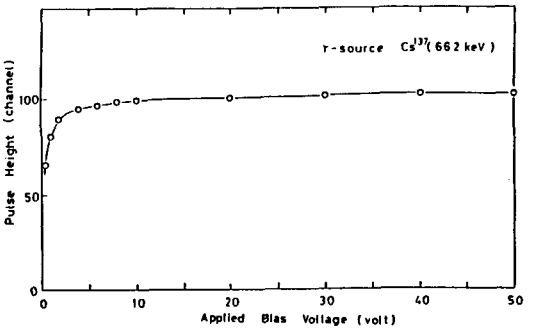


Fig. 3

Typical spectra of gamma-ray with PMT and PD mode

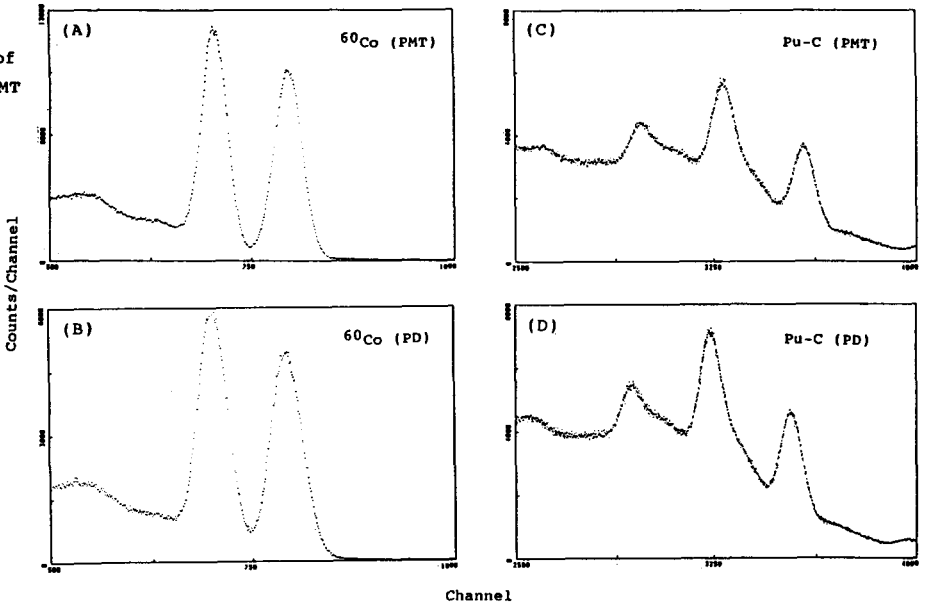
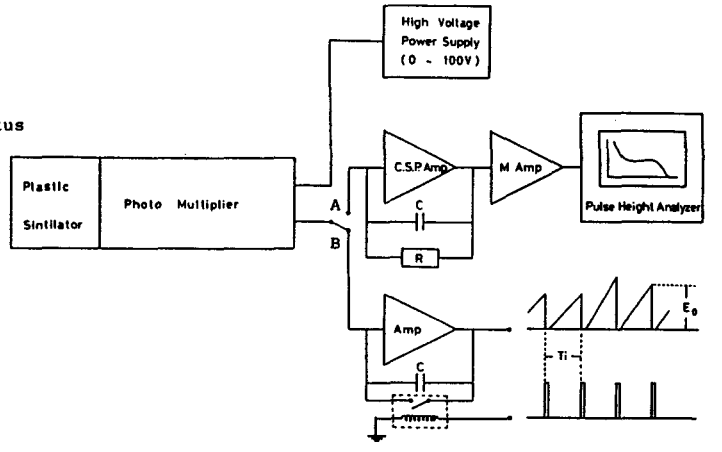


Fig. 4

Schematic diagram of the apparatus for absorbed dose measurement



HIGH RESOLUTION SCINTILLATOR/PHOTODIODE DETECTORS FOR ROUTINE MONITORING

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For many years the practical application of photodiodes in scintillation detectors has been limited by the relatively high levels of noise associated with their junction capacitance and the charge amplification process. Recently devices have become available with more suitable characteristics, which have allowed the development of detectors for routine radiation monitoring.

DETECTOR SIGNAL/NOISE

The equation for the noise at the output of a system consisting of a photodiode, FET charge sensitive preamplifier and main amplifier with CR-RC pulse shaping has been given by Delaney (1980). The noise contains several components and the variation of noise with pulse shaping time shows a definite minimum indicating an optimum pulse shaping time for a given photodiode/FET configuration. This time has been measured experimentally and is in good agreement with theoretical predictions.

Some of the most important components in the noise equation are dependent on the junction capacitance of the photodiode, which is roughly proportional to the diode sensitive area. However the size of the signal is determined by the light which can be coupled onto this area. We have therefore investigated the signal/noise ratio as a function of the proportion of the face area of a large scintillator optically coupled to a given photodiode. Areas of the scintillator which are not coupled are coated with a diffuse reflector, such as magnesium oxide powder, so that some of the light finally emerging suffers multiple internal reflections. For a given face area of scintillator there is an optimum sensitive area for the diode, and the signal/noise ratio can be predicted quantitatively using a simple theoretical model.

It appears also that to obtain a good optical coupling from a scintillator, it is important for the window to be in contact with the silicon surface. The type of diode which has an epoxy resin window provides the best results in spite of having a poorer transmission in the UV part of the spectrum.

Resin window photodiodes are available as PN or PIN devices. It has been pointed out by Groom (1984) that PN diodes have an associated series

resistance which acts as an extra source of thermal noise. We have measured this series resistance for PIN diodes and find it to be much smaller than for the PN variety. PIN diodes are also able to withstand a greater reverse bias, enabling a greater reduction in junction capacitance and hence a significant reduction in the other noise components.

The photodiode output signal also depends on the spectral matching of the light emitted by the scintillator and the overall diode response. Table 1 shows a comparison of the signal strengths for 662keV photons from 137Cs measured for various scintillators coupled in turn with silicon grease to a photomultiplier and a resin window photodiode. The scintillators were cut into cubes of side 1cm and one face polished on moistened paper. The photomultiplier had a bialkali photocathode with peak quantum efficiency at 420nm, while the photodiode has a broad absorption band peaking at some 900nm. The advantages of using CsI(Tl) with the photodiode and NaI(Tl) with the photomultiplier are clearly seen.

Table 1. Comparison of observed pulse-heights with various scintillators on photomultiplier and photodiode.

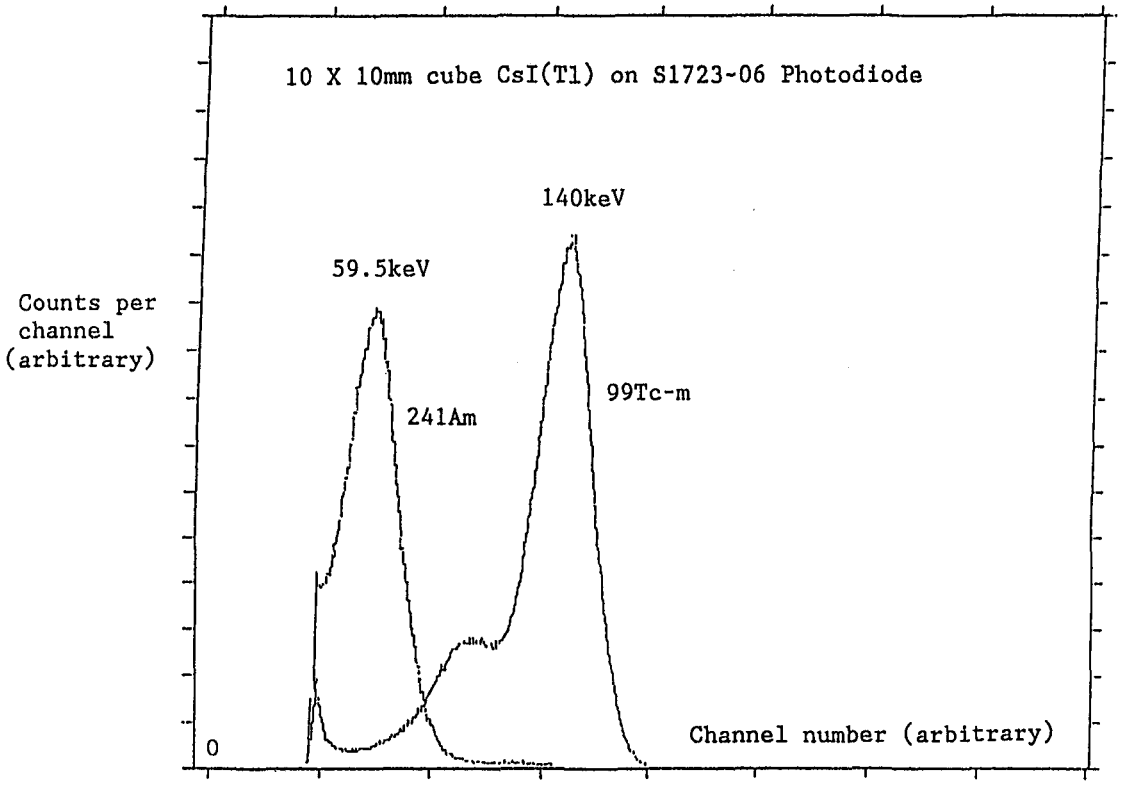
Scintillator	Reported light output as a % of NaI(Tl)	Pulse Height Measured on PM as % of NaI(Tl)	Pulse Height Measured on PD as % of NaI(Tl)
NaI(Tl)	100	100	100
CsI(Tl)	41*	31	148
CsI(Na)	65 - 83*	84	82
BGO	8+	-	-

* Nuclear Enterprises Inc. Brochure No 126P (1980)
\+ Furukhi M.R. Mat. Res. Soc. Symp. Proc Vol 26 (1963)

DETECTOR PULSE HEIGHT SPECTRA

Only one commercially available photodiode (Hamamatsu Photonics S1723-06) meets these requirements and it happens to have a sensitive area of $1cm^2$. The performance of various scintillators has been studied using this device together with a high quality pulse amplification system (Ortec) with pulse shaping time of 10μs and a 4000 channel analyser (Nuclear Data). Some results for CsI(Tl) are shown in Fig.1 and Table 2. The width of the peaks is due largely to the electronic noise and corresponds to a FWHM of 38keV. The photopeak of 241Am can clearly be resolved.

Fig.1 Pulse height spectrum for photons from 241Am and 99Tc-m



For the 2" diameter X 1" high scintillator the proportion of the light which can be directed onto the $1cm^2$ photosensitive area is subtantially reduced and this is shown in Table 2 by the reduced channel number of the photopeak. However the noise width as indicated by the FWHM is relatively independent of the crystal dimensions.

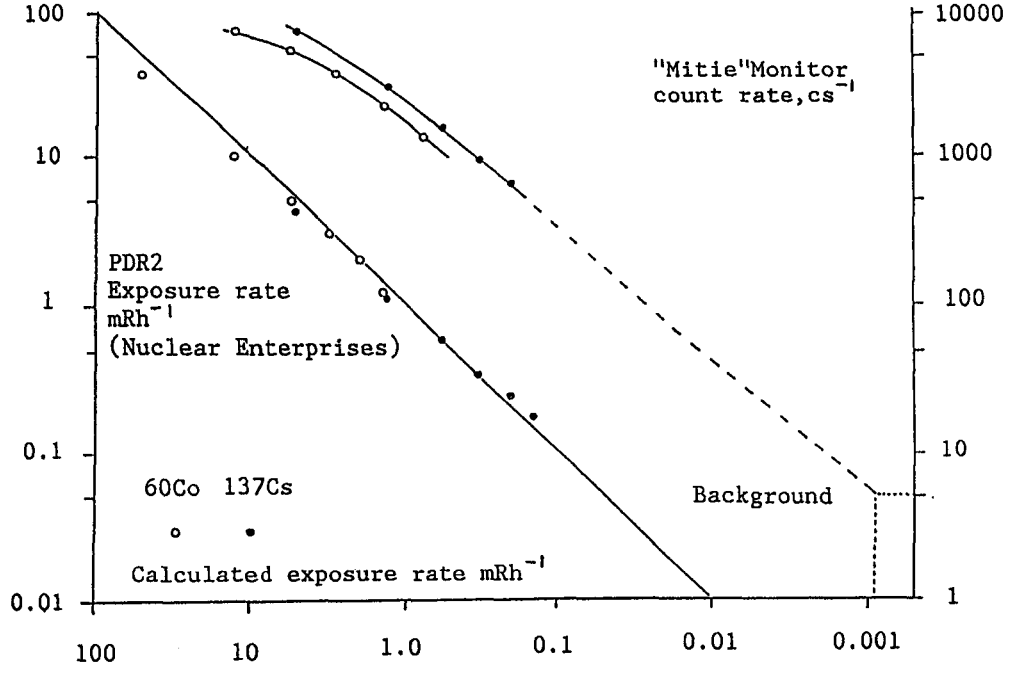
Table 2 Effect of increasing face area of CsI(Tl)

Crystal Dimension mm	Photopeak Channel No	FWHM channels	Resolution %
7x7x12.7	378	21.5	5.7
10x10x10	365	21.0	5.8
20 dia x 10	288	23.5	8.2
25.4 dia x 12.7	231	20.5	9.0
50.8 dia x 25.4	114	19.5	17.1

PROTOTYPE MONITOR

To demonstrate the promise of such detectors, we have constructed a prototype we call the "mitie" monitor. The detectors are detachable and plug into a box, measuring some 13 X 8 X 5cm.,which contains a 9V battery, charge amplifier, main pulse shaping amplifier and a digital ratemeter displaying counts per second. On the front panel the threshold energy can be adjusted and an analogue output is provided for spectrometry. Fig. 2 shows some results obtained using a 1" diameter X 1" thick CsI(Tl) detector compared to those for a commercial instrument using a compensated GM counter. The sensitivity of the "mitie" is much greater as expected for a scintillation counter, and with suitable integrating time can measure satisfactorily at the $1\mu Rh^{-1}$ level.

Fig.2 Comparison of response to photons from 137Cs and 60Co



CONCLUSIONS

The prototype can conveniently be held in one hand, is robust and the pulse height is inherently stable. By eliminating the need for a photomultiplier and a high voltage supply, it is relatively inexpensive to produce. With further development we may expect the scintillator/photodiode detector to replace present scintillation detectors in many current applications.

ADVANTAGES AND LIMITATIONS OF Si SURFACE BARRIER AND CdTe DETECTORS FOR MEASURAMENT OF DIAGNOSTIC X-RAY SPECTRA.

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INTRODUCTION

The knowledge of diagnostic X-ray spectra is very important to determine and minimise the dose to the patient. To this purpose and also for image optimisation it is also necessary to know the spectra of the photons trasmitted from a phantom or from the patient in the actual conditions of a radiological examination. X-ray spectrometry by means of germanium detectors can be performed ,at the present time, by well established methods. One of the major problems , encountered in this type of measuraments is due to the high photon fluence typical of diagnostic X-ray beams. Counting rates, pile-up and pulse distortion are usually reduced by: a) very narrow collimations and large focus detector distances, (b) decreasing of X-ray tube current (10-50 μA) ,whenever possible. It is then worthwhile to investigate on detection systems whose efficiency is much lower than that of the Ge detectors and which are capable to realise a compromise between accuracy of measurement and simplicity of use as happens with Silicon surface barrier and CdTe detectors (Pani et al. 1986, Di Castro et al. 1983). In this work diagnostic X-ray spectra measured with both detectors are compared in the energy range 10-100 keV. Detector work conditions and the stripping procedures utilized are also analysed and discussed. The advantages of these spectrometric system are : operation at room temperature, simplicity of use with photon flunce up to 10^7 photons /mm^2·s, and response in fairly good agreement with that typical of a Ge detector. Some difficulties arise in the use of these detectors as they require for the detected spectra correction procedures that are more complex than in the case of Ge detector.

EQUIPMENT AND METHODS

An n-type silicon surface barrier detector (Ortec AB-16-25-300) 300 micron thick,totally depleted with an area of 25 mm^2 was used in this study. It is usually employed to detect charged particles but it has good characteristics also to detect X-ray photons such as a good energy resolution that varies from 5.4 to 6.3 keV in the range from 20 to 400 keV. The cadmium telluride detector was grown in Italy in the I.N.F.N. Laboratory of Bari. It was produced with travelling heater method (THM) and it has a sufficient energy resolution (9% at 122 keV) and a size of 5x5x1.7 mm . CdTe detectors are not at present available with volumes larger than 0.5 cm^3 and with high purity. Cdte and Si detectors have a low detection efficiency that make them quite suitable for application in high intensity photon radiation beams

which may exceed the detection capability of the spectrometric system such as in a direct measurements of diagnostic X-ray spectra. They do not need the cryogenic cooling which often limits the choice of a suitable experimental geometry. All measuraments of distributed spectra were compared with ones obtained from a Ge high purity detector (PGT IGP 105) with a size of 100 mm^2 x 5 mm. The X-ray equipment consists of a metrological tube Philips MG421 with a costant potential in the range 30-420 kV. The focus-detector distance used in this study was 1 meter. A collimation with 1 mm of aperture diameter was used for the Ge detector. The collimation for Si and CdTe detectors was chosen at 4 mm of aperture diameter to limit the irradiation area to the detector active area. Energy calibration and efficiency was determined for each detector using standard point sources in the energy range from 5 to 200 keV. In figures 1 and 2 are shown pulse height distributions of two typical X-ray qualities,and the presence of spurious effects for Si and CdTe detector respectively. These effects are principally due to the escape from the depletion layer of Compton photons, K-photons, photoelectron and Compton electron and due to the incomplete charge collection. The influence of these effects depends on detector active volume , its atomic number and purity. To obtain the true bremsstrahlung spectrum impinging on the detector a carefull analysis is needed to determine the influences of spurious effects on the detector response. This analysis consists of calculating the transport of radiation through the detector. To this aim a Monte Carlo method based on "analog procedures" was used. The techniques and the general methods employed in the Monte Carlo program for the solution of transport problems are discussed elesewere (2). After the Monte Carlo analysis a "stripping procedure" is applied to correct the measured spectra.

STRIPPING PROCEDURE

The stripping procedure for CdTe and Si detector is summarized in the following formula:

$$N_t = \left\{ N_d(E) - \left[t_k(E+E_K) * N_t(E+E_k) \right] - \left[\sum_{E_o=E_{c,o}}^{E_{max}} h_s(E,E_o) N_t(E_o) \right] - \left[\sum_{E_o=E}^{E_{c,o}} h_m(E,E_o) N_t(E_o) \right] - \left[\sum_{E_o=E}^{E_{max}} R N_t(E_o) \right] \right\} * \varepsilon(E)^{-1}$$

The formula consists of the subtraction of four terms from the number of detected photon of energy E ($N_d(E)$). The first term represents the number of K-photons that escape from the detector.The presence of this effect has to be negleted for Si detector because it is negligible for incident photons energy greater than 12 keV and for the presence of the detector noise below. The second and third term represent the number of events of energy E due to electron escape and due to single and multiple scattered photon escape respectively. These correctionshave been completely neglected for CdTe detector because Compton scattering becomes considerable only for incident photon energy greater than 150 keV. The last term represents the events introduced from theincomplete charge collection. The charge trapping is the main contributing factor to this effect for CdTe. Such correction can

be neglected in Si for its high purity. Finally correction for photopeak efficiency $\mathcal{E}(E)$ is applied. A more detailed desription of $h(E)$, $r_K(E)$ and R is discussed elsewere (1) (2).

RESULTS AND DISCUSSION

Measured and corrected spectra for CdTe and Si detector are shown in figure 1 and 2 respectively. The stripping correction effect is impressive for both detectors. Figure 1 shows a sufficient agreement between the 100 kV spectrum detected by CdTe and the same obtained with Ge detector. It was possible to analyse with CdTe detector only heavily filtered spectra because of the detector noise (20 keV) and the higher errors introduced by the stripping increasing the number of the channels involved. Typical diagnostic X-ray spectrum is shown in figure 2 obtained with Si detector. There is a good agreement with the spectrum measured with Ge. Only a difference can be noticed in the energy range of W-K lines due to the lower resolution of Si. The higher accuracy of Si detector response and stripping correction is also shown in figure 3 where the tungsten K-lines affect the energy distribution much less than CdTe detector.
In conclusion the Si surface barrier detector offers very interesting advantages optimizing the compromise between accuracy and simplicity of use in the energy and fluence rate range of radiodiagnostics.

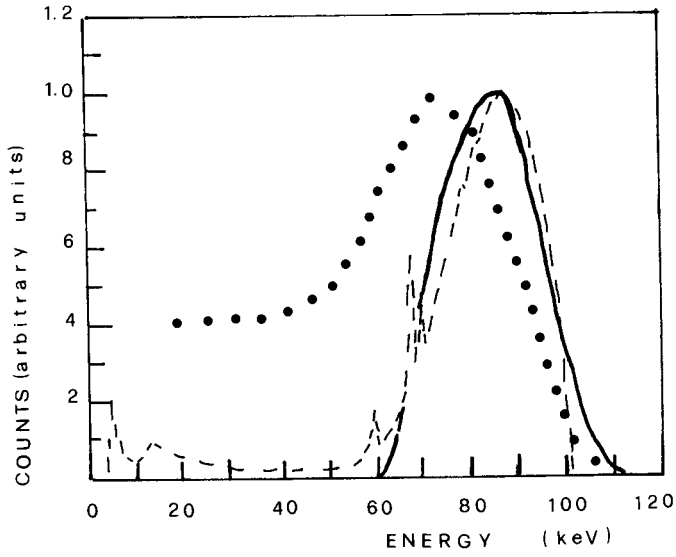

Fig.1 X-ray spectra obtained with the CdTe detector at 100 kV filtered with 4 mm Al+ 5mm Cu. Measured spectrum (dot); corrected spectrum (full line); same spectrum obtained with Ge detector (broken line).

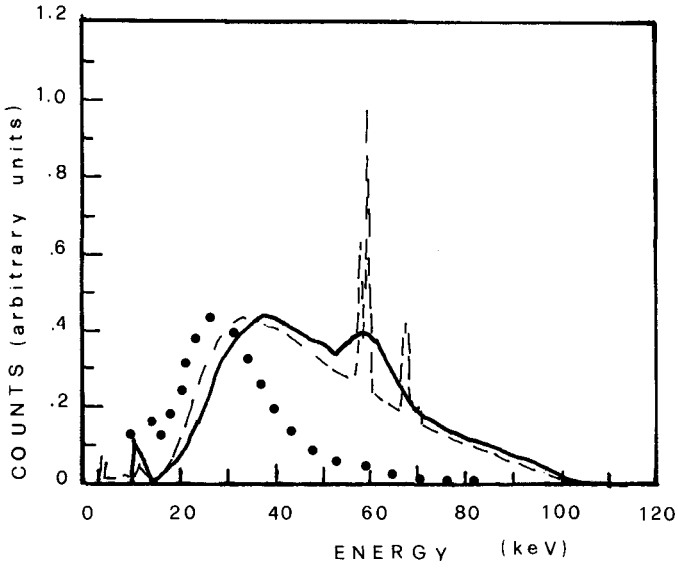

Fig.2 X-ray spectra obtained with Surface barrier detector at 100 kV filtered with 2.5 mm Al. Measured spectrum (dot); corrected spectrum (full line); same spectrum obtained with Ge detector (broken line).

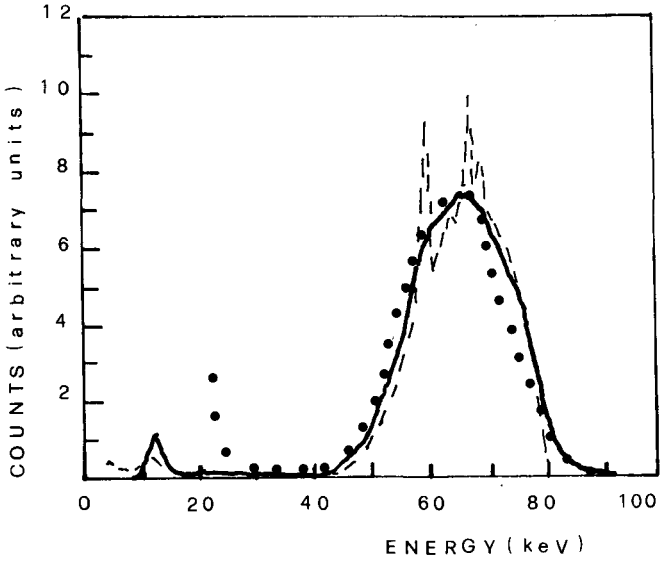

Fig.3 80 kV filtered X-ray spectrum (4 mm Al+ 2 mm Cu).A comparison between corrected spectra obtained from Si (full line), CdTe (dot) and Ge (broken line) detectors.

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OBJECTS OF A STABLE BEAM CALIBRATION SYSTEM BASED ON PHYSICAL CONSTANTS

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Calibration of protection-level dosemeters is a legal requirement in many countries[1] and in the UK is required by Regulation 24 of the Ionising Radiations Regulations 1985[2]. Two important features of the testing required under this regulation are that 1) the calibration is traceable to a national standard and 2) the accuracy of the calibration is known.

A major difference between the requirements for the calibration of clinical dosemeters for use in radiotherapy and of protection-level dosemeters is that in the former case there are a few dosemeters requiring calibration to a high standard of accuracy while in the case of protection-level dosemeters, there are a great number of dosemeters requiring calibration to a lower, but known standard of accuracy. The performance of dosemeters used for protection purposes must be verified on a regular basis, and in view of the numbers of dosemeters and doserate meters involved, a robotic system presents many advantages, as it is no longer permissible to use dosemeters the characteristics of which have not been verified.

In the system described, dosemeters are placed in known positions on measuring platforms (fig 1). On leaving the calibration cell, the operator initiates the calibration procedure, which can be wholly or partly robotic. Measuring sequences can be chosen so that each of two detectors can be placed at each position and exposed in turn, or alternatively a single detector can be exposed at each chosen position along the axis of the beam.

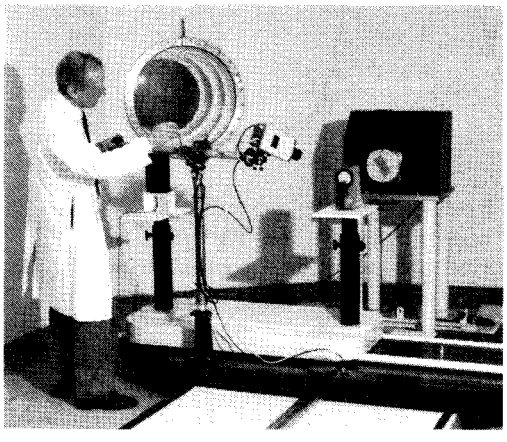

fig 1 GENERAL VIEW SHOWING MEASURING PLATFORMS

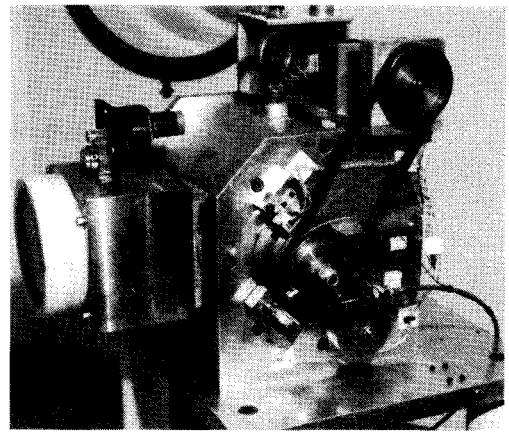

fig 2 SOURCE HOUSING AND CONTROL MECHANISM

Conventional dosemeter calibration is based on the transfer of calibration factors from a calibrated secondary standard dosemeter which itself requires to be calibrated regularly against a national standard, so that the calibration of the instruments calibrated against it are also traceable. The secondary standard dosemeter itself is subject to possible loss of its calibrated status due to severe environmental conditions, damage or component failure. It is necessary to maintain a reference source to check the calibration of the secondary standard dosemeter.

The philosophy of a stable-beam calibration system is to use for calibration purposes the output from a radioactive source the air kerma rate from which at a given distance, once determined, can be predicted by the use of physical constants, so as to eliminate the intermediate secondary standard dosemeter interposed between the primary standard and the dosemeter under calibration by using the radiation beam from the source as the actual calibration standard. While this eliminates the need for the use for the secondary standard dosemeter for each calibration, there may however remain the need to validate the accuracy of the system at regular intervals of say a week or a month by checking the air kerma rate at a fixed point in the beam using a calibrated secondary standard dosemeter. These measurements will ensure that that changes have not arisen due to mechanical inconsistencies arising from shocks to the system or wear and tear, or to unpredicted changes in air kerma rate arising from the decay of the radioactive source not following the predicted values due to the presence of source impurities, and are required in the UK to satisfy the requirements of NATLAS and NAMAS.

Inaccuracies occurring in calibration against secondary standard dosemeters of known accuracy can arise due to uncertainties regarding the source in the case of X-ray generators, to errors in the secondary standard itself, and to positional inaccuracy, arising from uncertainties in the determination of the exact postion in the photon beam where the air-kerma rate is determined, together with the need to position the dosemeter under calibration with comparable accuracy. If an X-ray source is used, there is the further requirement to position a reference dosemeter, to allow the air-kerma rate to be normalised between dosemeter readings.

While at this stage of its development, a stable-beam system cannot itself be regarded as a secondary reference standard, the performance of such a system is substantially better than that of a tertiary standard, and the degree of consistency of measurement which can be achieved is such that only occasional reference to a secondary standard is required. The calibration standard in a stable-beam system consists of the photon beam itself, emitted by a radioactive source or sources of photons of suitable energy or energies, the characteristics of the source and of the beam itself being known to a high degree of accuracy. Once the air-kerma rate at a given point on the axis of the beam has been established, the air-kerma rate at that point can be predicted at

a future date by means of the decay constant of the given radionuclide and the reading of the instrument under calibration compared with this value.

The use of the air kerma rate at a point in the beam from a radionuclide source as a reference standard presupposes the availability of a stable scatter-free beam, the characteristics of which have been verified by a national standardising laboratory.

CALIBRATION USING A STABLE BEAM SYSTEM

The use of a stable beam system for calibration consists therefore of the following initial steps:

The system is set up with one or more radionuclide sources. The beam is checked for electron contamination and the photon energy checked.
The air-kerma rate on the axis of the beam is measured to check for any deviation from the inverse square law prediction and entered into the system computer, together with the date and decay constant of the source (fig 3).
The air-kerma rate is measured across the beam at various positions along the axis to ascertain uniformity of air-kerma rate over given measuring volumes, in order to establish accuracy limits for radiation transducers of different sizes (fig 4).

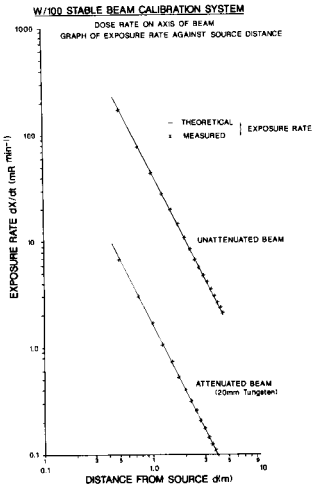

fig 3

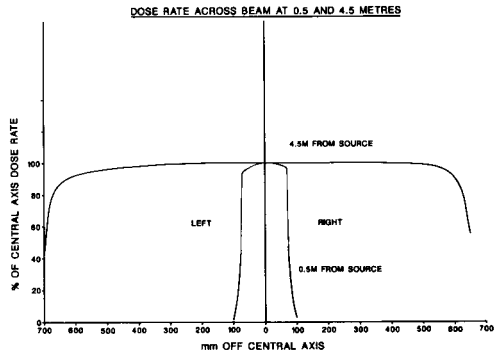

fig 4

CALIBRATION PROCEDURES

Dosemeters under calibration are then set up on one of the two measuring platforms and moved to the limit of distance along the axis of the beam (Z-axis). The reading of the instrument is then observed and entered into the computer, and the detector moved to another position.

The time required for each observation is limited by the response time of the instrument, and is pre-programmed before

the measurement, the detector being driven automatically to the next position after a given time. When the series of measurements is completed, the computer produces a calibration certificate for the instrument which includes all relevant information: the measurement data entered by the computer and the details of the instrument entered by the operator. In the case of ionisation chambers under calibration, details of temperature and pressure are also entered automatically by the computer, and the correction factor applied so that the calibration produced is correct at 20°C and 1013 mbar (1.013×10^5 Pa).

Source housings (fig 2) are available to permit the use of two sources of high photon energy; for sources of lower photon energy, an alternative source housing is being developed. The lower energy sources are intended to fit on to the axis of the calibration system in front of the main source housing. Examples of nuclides evaluation of which it is proposed to carry out to assess their suitability for calibration purposes include ^{241}Am, ^{57}Co and ^{125}I.

A wide range of tests may be required for complete calibration. Some, such as those relating to warm-up times and the response time of the instrument are important for measurement of rapidly changing radiation intensities. It is also important to validate instrumental integrity over the range of physical conditions under which the instrument is likely to be used, and over the full range of intensity and photon energies to which the instrument may be exposed. The calibration process consists of a range of pre-programmed measurements, which are compared with the standard data stored in the computer. A summary of the results from a full calibration of a range of instruments can then be used to evaluate their suitability for use for a particular purpose.

The purpose of developing this system is to attempt to set up a calibration facility which satisfies the NAMAS and NATLAS requirements as set out in publications M1S1, M1S2 and M1S3, B0021, B0102 and M1 to M5[3]. The first system was installed at Bristol General Hospital in what was previously the Cobalt Teletherapy Suite in the old Radiotherapy Department in October 1987.

These calibration systems are being developed in collaboration with Messrs Pantatron Engineering Limited, Gillingham, Kent.

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A SELF-CONTAINED ENERGY & PULSE SHAPE LIQUID SCINTILLATION SPECTROMETER

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INTRODUCTION

The increase in nuclear energy applications and the use of technically enhanced sources of radioactivity engenders a need for rapid, low-level techniques to detect and quantify alpha radionuclides. Existing methods often require extensive chemical separations, ion exchange and thin film deposition onto flat surfaces. Counting is performed with surface barrier or gas flow proportional instruments. Occasionally tracer radionuclides are needed to determine the variations in chemical procedures or plating efficiency while prepared standards are used to correct for self-absorption and backscatter. The existing methods may not be feasible for some applications or for field measurements.

Liquid scintillation alpha spectroscopy is a viable alternative to the methods described above. Coupled with solvent extraction techniques, alpha emitter recovery and counting efficiency can approach 100%. The 4π counting geometry eliminates backscatter and self-absorption. Pulse shape discrimination (PSD) signals may be used to segregate alpha particle from beta-gamma decay events. The concept is called Photon Electron-Rejecting Alpha Liquid Scintillation (PERALS) spectrometry. PSD alpha scintillation spectrometers have been developed and applied to the assay of uranium and thorium concentrations in phosphate fertilizers [Bo79,Me80] and ^{210}Po in a uranium mill circuit [Ca81,McK83].

MODULAR PERALS SPECTROMETER

The Radiation Measurements Facility at Arizona State University has attempted to develop a portable PSD alpha spectrometer [Kl81]. The PSD circuit used in the portable system was designed by Thorngate [Th77,Th78]. However, the electronics suffered two instability problems, "walk" and "jitter". Walk refers to the trigger level of a measurement with respect to the amplitude of the peak; for example, a trigger level of 0.25 volt occurs at greater percentage of a 1.0 volt signal than a 10.0 volt signal. Jitter refers to the changes in the trigger level timing

due to fluctuations in the signal. To compensate for the instabilities it was necessary to perform frequent adjustments of several interacting potentiometers. A high speed oscilloscope and a pure alpha-emitting sample were needed to perform the adjustments. Therefore, it was decided to reevaluate the electronic schemes published by Thorngate and others. The objectives were to determine if the electronics in the PSD circuitry could be improved and the instabilities eliminated.

The resulting electronic circuit incorporates the ideas of Alexander and Goulding [Al61] as well as Thorngate. Figure 1 is a block diagram of the new electronic scheme. Referring to Figure 1, the photomultiplier tube anode pulse (point A) is integrated by the pre-amplifier. The signal branches (point B) into an upper and a lower channel and an output which is delayed for 400 nanoseconds. The delay allows the integrated signal to reach its peak voltage (point E). The upper and lower channels each contain a percentage of the non-delayed signal to be used in the compatators. The upper channel contains a larger percentage of the signal than the lower. The signals in the upper and lower channels (points C & D) are fed to separate comparators and produce a fixed fraction of the signal (points F & G) for use in the time-to-amplitude-converter (TAC). The comparators trigger as the rise of the delay signal crosses the voltage settings determined by the upper and lower channels. The time difference between the comparator trip points is processed by the TAC and produces a PSD spectrum (point H) [Ca85].

Figure 2 is a diagram of the modular PERALS spectrometer. The spectrometer is six NIM modules wide. It contains a high voltage power supply, a photomultiplier tube, a sample holder, and the PSD and pulse-height amplification electronics. The unit fits into a standard 6 or 12 wide NIM bin. As shown in the figure, outputs from the PSD portion of the system may be sent directly to a multichannel analyzer (MCA) to provide gross alpha and beta/gamma decay information. In addition, the PSD signal may be used to gate the MCA and reject all beta/gamma decay events. Thereafter, only the alpha decay events from the energy pulse-height amplifier will be accepted by the multichannel analyzer. The resulting alpha energy spectrum does not contain any beta/gamma interference.

Two figures of merit defined by Thorngate (Th77), the beta-gamma Rejection Ratio and the Background counts, may be used to evaluate the performance of the system. The Rejection Ratio is the unit's ability to reject the beta/gamma decay events occurring within the sample. Background is the unit's ability to reject the environmental beta/gamma events. The new PERALS spectrometer exhibits a slightly improved Rejection Ratio over Thorngate's (our system = 99.98% , Thorngate = 99.95%) but had a higher, yet acceptable background (our system = 0.046 cpm, Thorngate = 0.002 cpm). The new system has less jitter and walk and does not require frequent adjustment. It is hoped that the system will be commercially available in the future.

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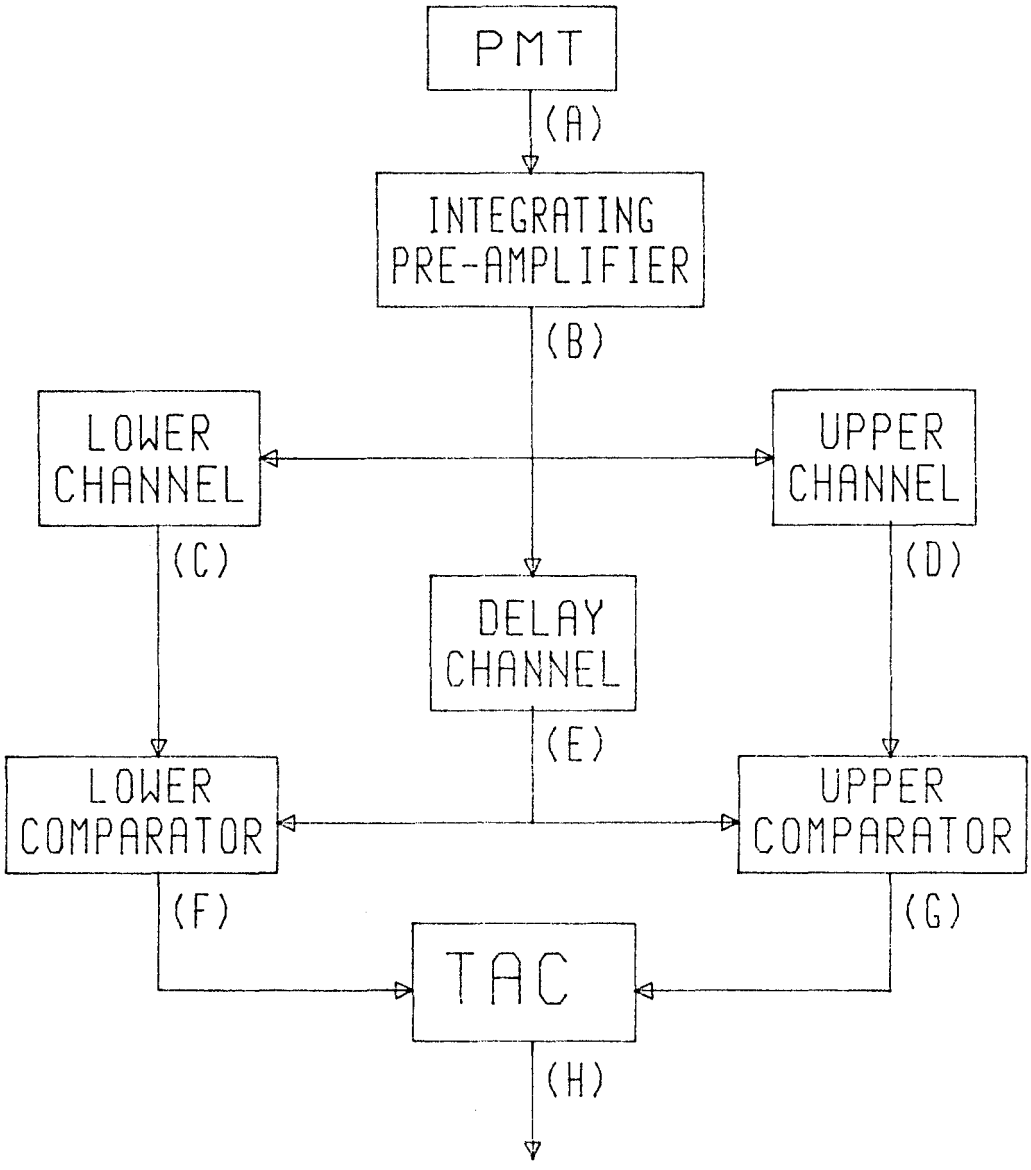

FIGURE 1. BLOCK DIAGRAM OF NEW ELECTRONICS

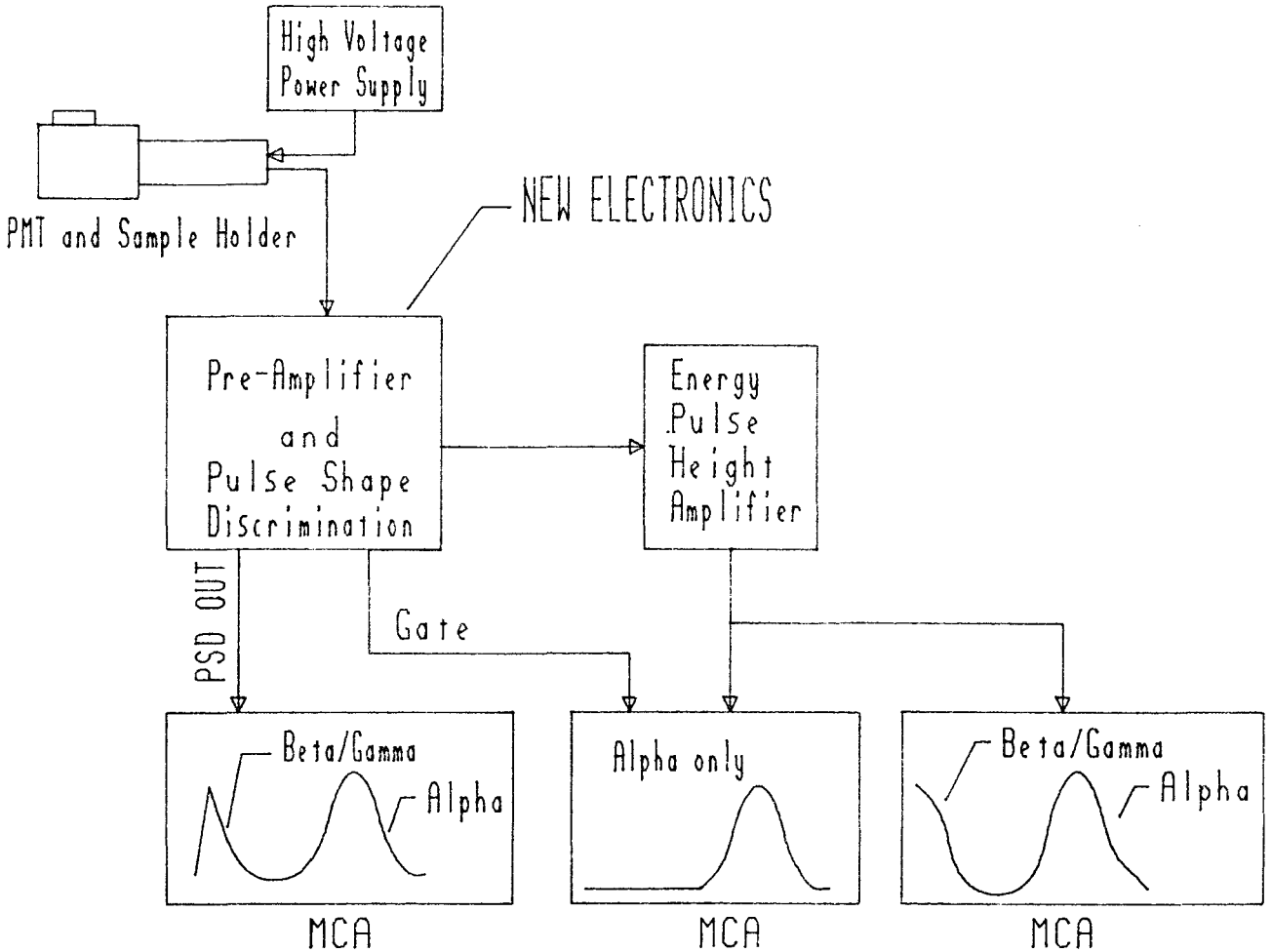

FIGURE 2. PERALS SPECTROMETER

AN INEXPENSIVE DOSIMETER CALIBRATOR

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ABSTRACT

A large Radiology department may have many room monitors, dosimeters, and survey meters. The cost of having all these devices calibrated yearly can amount to a large sum of money.

This paper describes a purpose built, inexpensive calibrator which uses 241-Am as the source of radiation. The calibrator can be used to monitor changes in the response of any of the radiation detection devices in the department. It is designed to provide a range of exposure rates. The system is very reproducible and can accommodate a wide range of ion chamber shapes.

241-Am emits a range of alpha particles and a 59.5 keV gamma ray, which falls nicely in the range of typical diagnostic x-ray energies. 241-Am has a half life of 432 years which provides a good degree of reproducibility without the need for constant decay correction.

The use of the system is for monitoring changes in response of calibrated devices, and should not be considered to have eliminated the need for proper calibration from time to time by a Standards Laboratory.

SIX CHANNEL LOW BACKGROUND ALPHA COUNTER

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Frequently, it is required to analyse a fairly large number of samples for weak alpha radioactivity in the field of radiation protection, health physics, environmental monitoring, food inspection, nuclear medicine, radiobiology and geological prospecting. The sample is always separated or enriched by some radiochemical processes and deposited to form a thin planar source, finally put it in a low background assembly for radioactivity measurement. In order to lower the minimum detection limit, the counting time is extended greatly, especially for weak alpha activity, always up to 24hours. For this reason, a sort of practical instrument must be stable, reliable, low background, unsensitive to disturbance, easy to operate, able simultaneously to measure several samples, able to continuously operate longer than 24 hours and low production cost, that will suit the demand very well.

Silicon- gold surface barrier semiconductor detectors are chosen for alpha detectors, because which are more firm, less auxiliary circuits and smaller in size than scintillation counters, and less sensitive to disturbance, less auxiliary installation and smaller in size than grid ionization chamber. The instrument consisting of six detectors, might simultaneously and separately count six samples. Each detector look on to a planar source which is put into a shallow hole on a drawer.

The signal from the detector is fed to a charge sensitive amplifier, then to alinear complementary voltage amplifier and a discriminator. All these linear circuits are designed in a low power consumptive mode and allow to work at wide voltage range. The lasting current of each channel linear circuit is less than one milliampere at 7-9 volts.

In order to reduce the noise, ordinary charge sensitive amplifier utilizes high mutual conductance FET (field effect transistor) and hence large drain current is needed. We use high β transistors with small steady current and get less noise to meet this instrument feature. The discriminator consist of a FET, a thyristor and a transistor. The input PN junction of the thyristor biases the FET. Once the FET is set into conducting sufficiently, it will trigger the thyrister. The detector and linear circuit of each channel is put into an electric magnetic shielding to reduce the interference from each other and the disturbance from outside.

The outputs of six linear circuit channels are fed to six storage counters, which are made of CMOS integrated circuit for reasons of saving power and reliability. Each storage counter has four decimal digits and a overflow signal. There are only one decoder and one display with liquid crystal for all six storage counters. They also share a self-test device and an auto-timing device, which bases on a quartz time circuit and the time interval are $\frac{1}{2}$, 1, 5, 10, 20, 60, 100, 400, 1000, 1400 minutes optionally.

Whole insrument is supplied by a set of six storage batteries or six dry cells, with neither voltage stabilizer nor step-up device, so that no extra power is expended. Non-linking with mains power obtain two advantage: to avoid disturbance from mains network, from which majority interference come; to avoid losing the data just in progress, this situation may happens in a mains supplied instrument, while the mians break off unexpectedly. Fresh recharged batteries will last 500 hours or longer. If the voltage supply lower below some level, a special sign will appear.

The outward appearance of the instrument is same as a NIM standard bin, with size 480mm(W)x240mm(H)x340mm(D). The effective sensitive size of detectors may be 12 mm, 16 mm or 20 mm in diameter optionally. The performance index of mean instrument back-background is 6, 8 or 10 counts per 24 hours respectively, and detection efficiency for planar 2π source is larger than 60%, 65% or 70% respectively. The actual measured mean background is 2.4, 3.2, or 4.7 counts per 1400 minutes, and efficiency of 7 mm diameter

Pu-239 planar source is 65.4%, 74.2% and 79.1% respectively. Up to now, forty sets of six channel instrument are working satisfactorily.

THE DEVELOPMENT OF ENERGY COMPENSATED GEIGER MULLER DETECTORS. FOR THE QUANTITY AMBIENT DOSE EQUIVALENT

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and

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National Radiological Protection Board

ABSTRACT

Many national authorities are considering the adoption of the dose-equivalent quantities proposed in ICRU publication 39. Two of these, ambient and directional dose equivalent, are appropriate for use in survey instruments. Energy-compensated Geiger Muller detectors have been designed to measure the quantity ambient dose equivalent over the energy range 40 keV to 1.25 MeV. These detectors complement existing exposure measuring devices. The general principles of design, materials and construction are described for four types, which cover in total the dose rate range from 1 $\mu Sv\ h^{-1}$ to 10 $Sv\ h^{-1}$.

RADIATION PROTECTION AT HIGH ENERGY HEAVY ION ACCELERATORS

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INTRODUCTION

The radiological impact in the environment of high energy accelerators is given by the following main contributions:

1. prompt radiation consisting mainly of neutrons which originate from shields;
2. activated air released from the accelerator building;
3. skyshine of neutrons emitted into the air above the accelerator.

Radiation exposure of the public due to activated ground water was found to be of no concern. Simple relationships have been derived which allow the calculation of the three dominant contributions to the radiation impact of heavy ion accelerators operating in the energy region between 0.4 and 2 GeV/nucleon.

CALCULATION OF THE NEUTRON DOSE RATE OUTSIDE SHIELDS

Assuming that the ion beam interaction zone can be regarded as a point neutron source (Fig. 1) the dose rate at the outer surface of the shield can be expressed in terms of the differential neutron yield per ion of the source, the effective fluence to dose conversion factor and the effective dose transmission factor of the shield. For thick shields (> 500 g/cm^2) the contribution of source neutrons below 100 MeV to the dose rate at the outer surface of the shield is only a few percent. Therefore to a good approximation the lower limit of the source neutron energy to be taken into account is chosen to be $E_0 = 100$ MeV. The effective quantities were calculated by folding the appropriate data for monoenergetic neutrons with measured and calculated spectra of neutrons produced in high energy heavy ion reactions [1]. Since the effective dose transmission factor for $\rho \cdot d > 500$ g/cm^2 can be described in terms of an exponential function and a dose buildup factor we obtain:

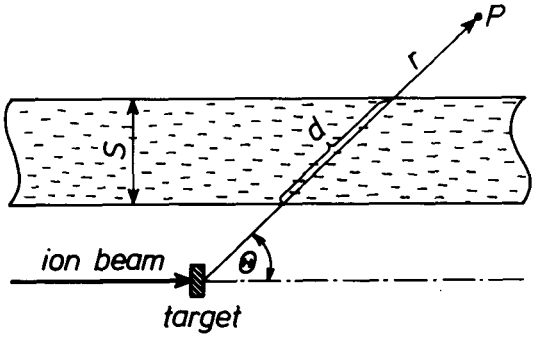

Fig. 1 Shielding geometry

$$\dot{H} = K \cdot \frac{1}{r^2} \cdot J \cdot \exp\left[- (\rho \cdot d) / \Lambda \right] \qquad (1)$$

The parameters K and Λ depend on the specific ion energy and on the neutron emission angle. Equ. (1) allows the straightforward estimation of the shield thickness $\rho \cdot d$ from the given dose rate limit. The necessary data are quoted in reference [1]. For comparison this method was applied to the lateral shielding of proton accelerators where other published data are available.

TABLE 1 Lateral shielding of multi-GeV proton and heavy ion accelerators by ordinary concrete or soil. E_i specific ion energy in GeV per nucleon; [$\rho \cdot d$] = g · cm^{-2}

Reaction	$\dot{H}(\Theta = 90°) \cdot r^2/J$ ($\mu Sv \cdot h^{-1} \cdot cm^2 \cdot s$)	Comments
p thick iron target	$0.16 \cdot E_i \cdot \exp[-(\rho \cdot d)/108]$	this work
	$0.39 \cdot E_i \cdot \exp[-(\rho \cdot d)/103]$	O'Brien-method
	$0.16 \cdot E_i \cdot \exp[-(\rho \cdot d)/117]$	Moyer-method
Ne thick iron target	$5.7 \cdot E_i \cdot \exp[-(\rho \cdot d)/106]$	this work
	$2.6 \cdot E_i \cdot \exp[-(\rho \cdot d)/114]$	this work

The various shielding formulas for protons are in satisfactory agreement. For thick concrete shields ($\rho \cdot d \approx 1000$ g/cm^2) the calculated dose rates differ by no more than a factor of two. For heavy ion beams the lateral neutron dose attenuation length is nearly identical to the proton value but heavy ion reactions yield much more neutrons than proton interactions at the same specific ion energy. This is simply a consequence of the larger number of interacting nucleons in heavy ion collisions. With respect to radiation protection this means that for the same specific ion energy and the same beam intensity a multi-GeV heavy ion accelerator requires considerably more shielding than a proton machine.

ACTIVATION OF AIR

With respect to radiological impact only a few gaseous radionuclides are of concern. The most important reactions and their effective cross sections are summarized in Table 2.

TABLE 2

Effective cross sections σ_{eff} (> 15 MeV) in mbarn of nuclear reactions induced in air by secondary particles produced by interactions of 2 GeV/u heavy ions.

		Targetnuclide		
Radionuclide	$T_{1/2}$	N 14	O 16	Ar 40
Be 7	53.3 d	2.9	2.5	-
C 11	20.4 min	16	8.2	-
N 13	9.96 min	17	3.4	-
O 15	2.03 min	-	34	-
Ar 41	1.83 h	-	-	660

The production of gaseous radionuclides can be estimated using an average spectrum (integrated over the emission angle) of secondary particles-predominantly neutrons - emitted from the beam interaction zone:

$$\dot{N}^{+} = n \cdot Q\ (>15\ \text{MeV}) \cdot \sigma_{eff}\ (>15\ \text{MeV}) \cdot r_{eff} \qquad (2)$$

$\dot{N}^{+}$	production rate.
$Q\ (>15\ \text{MeV})$	source strength of neutrons with energies above 15 MeV.
$\sigma_{eff}\ (>15\ \text{MeV})$	effective cross section of the activation reaction for neutrons above 15 MeV.
r_{eff}	effective radius of the activated air volume depending on the shape of the volume and on the neutron angular distribution.

The effective cross sections were derived by folding the energy dependent cross sections of the nuclear reactions with the average spectrum of neutrons with energies above the lowest threshold (> 15 MeV). The influence of neutrons below 100 MeV is significant because of their very large contribution to the total source strength and of the strong variation of the cross sections in this region. The yield of Ar 41 was estimated from the flux density of thermal neutrons in the accelerator void. From the yield of radionuclides the annual activity release from the different parts of the accelerator was estimated and used as the source term for the atmospheric dispersion. The main radiological impact is due to gamma submersion in the plume. The dominant contributions to the annual dose equivalent are caused by the positron annihilation radiation of the short lived radionuclides C 11, N 13, O 15. For a heavy ion beam of $J = 1 \cdot 10^{9} s^{-1}$, 2 GeV/u Ne-ions absorbed in a thick iron target, the total annual activity release from a 1000 m^3 target room (ventilation rate: 10 h^{-1}, beam time: 6000 h/a) is of the order of magnitude of 10^{12} Bq. At the location of maximum radiological impact the annual dose due to gamma submersion is about 1 µSv (stack height: 30 m) which is far below the dose limit for the public. Areas however where high intensity ion beams are absorbed should be supplied with an air circulation system in order to minimize the activity release and to keep the collective dose equivalent of the public as low as reasonably achievable.

SKYSHINE

In designing the shielding of the accelerator area it is important to consider neutrons which pass into the atmosphere and are scattered back to the earth thus contributing to the radiological impact. The skyshine dose rate can be described using the importance function of neutrons with given energy and emission angle for a certain distance from the source. Physically the importance function is the dose equivalent per source neutron passing into the atmosphere. The data of ref. [2] were used for the evaluation of effective importance functions by averaging over the neutron emission angle and by folding the data for monoenergetic neutrons with the average spectrum of neutrons emitted from the interaction zone of a 2 GeV/u Ne-beam (Table 3). It is essential

to take into account low energy neutrons (< 10 MeV) because they largely contribute to the skyshine. The importance functions in ref. [2] contain the contribution of photons produced in neutron interactions with air. The skyshine dose rate at distance r from a beam interaction zone without roof shielding is expressed as:

$$\dot{H} = Q(\Delta\Omega) \cdot I_{eff}(r)/r^2 \qquad (3)$$

$Q(\Delta\Omega)$ total source strength of neutrons emitted into the solid angle $\Delta\Omega$.

$I_{eff}(r)$ effective skyshine function at distance r from the source.

TABLE 3 Effective skyshine importance function (2 GeV/u Ne-beam)

r (m)	50	100	300	500	1000
$I_{eff}(r)$ $(\frac{cSv \cdot m^2}{neutron})$	$2.3 \cdot 10^{-13}$	$2.3 \cdot 10^{-13}$	$2.0 \cdot 10^{-13}$	$1.4 \cdot 10^{-13}$	$5.0 \cdot 10^{-14}$

Using the data in Table 3 the skyshine dose rate caused by a target area without roof shielding was calculated from equ. (3) assumming that a beam of 10^9 s^{-1} 2 GeV/u Ne-ions is absorbed in a thick iron target (Table 4). The effective source strength $Q(\Delta\Omega)$ of neutrons above 0.1 MeV emitted from the beam interaction zone into the atmosphere is about $6 \cdot 10^{10}$ s^{-1}. The influence of a roof shielding on the skyshine dose rate was evaluated in ref. [2]. Using these attenuation factors the results given in Table 4 are obtained. The results show that at high energy accelerators with large experimental areas at ground surface level the skyshine problem needs very careful consideration with respect to the annual dose equivalent of the workers and of the population living in the vicinity of the accelerator site.

TABLE 4 Skyshine dose rate (μSv/h) caused by an experimental area without and with an ordinary concrete roof shielding

	r (m)	50	100	300	500	1000
ordinary concrete roof shielding (g/cm^2)	---	200	50	4.8	1.2	0.11
	100	48	12	1.2	0.29	0.026
	200	10	2.5	0.24	0.060	$5.5 \cdot 10^{-3}$
	300	2.0	0.50	0.048	0.012	$1.1 \cdot 10^{-3}$

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ASSESSMENT OF OCCUPATIONAL EXPOSURES AROUND HIGH-ENERGY PROTON ACCELERATORS

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INTRODUCTION

At CERN, the European Laboratory for Particle Physics, three high-energy proton accelerators are in operation: a 600 MeV Synchro-cyclotron (SC), a 28 GeV Proton Synchrotron (PS) and the 450 GeV Super Proton Synchrotron (SPS).

At present the Large Electron Positron (LEP) accelerator is under construction and will start in 1989 initially operating at 51 GeV. In the meantime the 31 GeV Intersecting Storage Rings (ISR) for protons have been decommissioned (1984). In conjunction with the accelerators mentioned above occupational exposures to various types of radiation are of concern. As around nuclear reactors, the main source of radiation causing personal exposures is gamma radiation from activated accelerator components during accelerator shutdown periods. During operation of the accelerators, exposures to hadrons (mainly neutrons), covering a wide energy range in experimental areas, and to high-energy muons downstream of primary proton beam targets have to be considered. Outside controlled radiation areas occupational exposures are derived from a system of passive thermoluminescence detectors (TLD) used for area monitoring, while inside controlled radiation areas individual dosimeters are worn. This paper describes the systems used at CERN for monitoring of exposures.

INDIVIDUAL MONITORING

Personal monitoring at CERN is still based on films. For gamma, beta, thermal neutron and muon monitoring the double-coated Kodak RM type 2 film is used. For evaluating the different radiation components, the CERN film badge contains the following filter combination: (1) plastic 80 mg/cm^2, (2) plastic and aluminium 89 mg/cm^2, (3) open window, (4) tin 1070 mg/cm^2, (5) lead 1260 mg/cm^2, (6) cadmium 1150 mg/cm^2 and (7) plastic 355 mg/cm^2. While the exposure free-in -air of a ^{137}Cs source serves as a reference for the calibration, the interpretation of the optical density pattern behind the various filters, as a result of a irradiation in the photon field of induced radioactivity, takes the backscatter from the wearer's body into consideration. Hence, a good match of dose results between the film badge and the pocket dosemeter is assured [1].

The presence of beta radiation or thermal neutrons is checked by comparing the optical densities behind the two thin plastic and aluminium filters or behind the cadmium and tin filters, respectively. Thermal neutrons are of no concern for personal exposures at CERN, while beta doses mainly occur during the assembly of large high-energy physics experiments containing depleted uranium.

Muons cause a blackening of the gamma film about 20% lower than expected from a ^{137}Cs exposure for the same dose equivalent [2]. Personal exposures to muons are only occurring in limited quantities.

Personal dosimetry of hadrons (mainly neutrons) is still carried out with the Kodak NTA nuclear emulsion, since the stray radiation field outside the shielding of the CERN accelerators is composed of a broad spectrum of hadrons (up to the energy of the protons accelerated). Calibrations in such fields are carried out as a routine. Fading of latent tracks in the NTA film sealed by the NRPB under nitrogen is sufficiently low to enable the use of the nuclear emulsion during two-month periods. A two-monthly distribution period is further justified by the low personal neutron doses normally encountered at CERN [3]. The (conservative) conversion factor used is 14 tracks per mSv and mm^2.

Attempts to replace the NTA film by solid-state nuclear track detectors like LR115 and CR39, to overcome e.g. the fading problem, have been made since 1972 [4,5,6]. High background and poor reproducibility did discourage routine use at CERN.

AREA MONITORING WITH PASSIVE DETECTORS

The dose distribution outside controlled radiation areas is determined with a TLD system having a high neutron-to-gamma sensitivity. The system used consists of ^{6}LiF and ^{7}LiF (Harshaw TLD-600 and TLD-700 chips) inside a cylindrical polyethylene moderator of 12.5 cm Ø x 12.5 cm, so that the neutrons after slowing down are detected by the $^{6}Li(n,\alpha)^{3}He$ reaction. The read-out of the ^{6}LiF detector is corrected for gamma and charged particle background using the read out of the ^{7}LiF detector, almost insensitive to thermal neutrons.

A network of 178 TL monitors of this type is in use at present on both CERN sites on an annual read-out basis. The detectors are calibrated in the stray field itself using Andersson and Braun neutron rem counters and argon-air filled high-pressure ionization chambers at 13 different positions as reference detectors. The ^{6}LiF calibration factor for stray neutrons is about five times higher per mSv than for gamma radiation. The system is in use since 1973. The two pairs of ^{6}LiF and ^{7}LiF detectors have so far been read out using a Harshaw Atlas reader. The same detectors will be evaluated from 1988 onwards, using the Alnor hot nitrogen Dosacus reader and corresponding detector holders.

The neutron energy dependence of the response of such a small moderator system is not ideal with respect to dose equivalent measurements, but the system can still be used since the stray neutron spectrum below 10 MeV does not show too strong variations at sufficiently large distances (>100 m) from the shielding of primary proton beams [7].

EVALUATED OCCUPATIONAL EXPOSURES

Occupational exposures, as determined by both systems for the CERN population of 7363 persons (4791 under film-badge control) at the end of 1986, are summarized in Table 1. The collective dose for both CERN personnel and outside contractor staff under film badge control is given, together with the collective dose derived from isodose distributions as measured by the TLD system outside controlled radiation areas. An example of such an isodose distribution measured on the CERN Meyrin site is presented in Fig. 1. The collective dose derived assumes a presence of 21% of the time on the CERN site with no corrections applied for working indoors, so that a considerable overestimation may be present.

The collective dose as given in Table 1 for the film badge results contains a contribution of neutrons (for CERN staff an average of 12%), while for the collective dose due to stray radiation, as measured with the TLD system, the average neutron contribution is about 83%. Table 1 shows that although the main source of occupational exposure is work on activated accelerator components inside controlled radiation areas (target and primary beam areas), the collective dose due to stray radiation, as determined by the TLD area monitoring system, is not negligible. The total detriment due to CERN operation, expressed in terms of total collective dose (film badge + area monitoring results), could be compared to the total number of protons accelerated annually at CERN by the PS. This accelerator has been the principal source of protons at CERN during the periods considered, whereby it supplied protons to ISR and SPS as well. The collective dose per accelerated proton, as given in Table 2, diminished between 1976 and 1986 by a factor of about 7, partly due to improved accelerator design and maintenance practices, as well as accelerator shielding.

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Table 1

Year	Film-badge results		TLD area monitoring	
	Collective dose (man.Sv)[1]	Average individual dose (mSv)[2]	Collective dose[3] (man.Sv)	Average[3] individual dose (mSv)
1976	4.16		0.81	0.16
1977	4.62	1.06	0.83	0.17
1978	3.65	0.77	0.80	0.16
1979	4.06	0.84	0.70	0.14
1980	3.08	0.52	0.46	0.09
1981	2.08	0.41	1.30	0.24
1982	1.83	0.34	1.47	0.22
1983	1.68	0.29	0.63	0.09
1984	1.59	0.27	0.79	0.10
1985	1.67	0.30	0.61	0.08
1986	1.52	0.27	0.38	0.05

1) CERN personnel and outside contractors
2) CERN staff only
3) Total CERN population outside controlled radiation areas

Table 2
Collective dose versus number of accelerated protons

Year	Total collective[1] dose (man.Sv)	PS accelerated protons ($\times 10^{19}$)	Dose in Sv per proton ($\times 10^{-19}$)
1976	4.97	2.76	1.80
1977	5.45	3.66	1.49
1978	4.45	6.18	0.72
1979	4.76	6.99	0.68
1980	3.64	5.86	0.60
1981	3.38	5.08	0.66
1982	3.30	8.27	0.40
1983	2.31	9.68	0.24
1984	2.38	10.93	0.22
1985	2.28	9.78	0.23
1986	1.90	7.15	0.27

1) Film badge + TLD area monitoring.

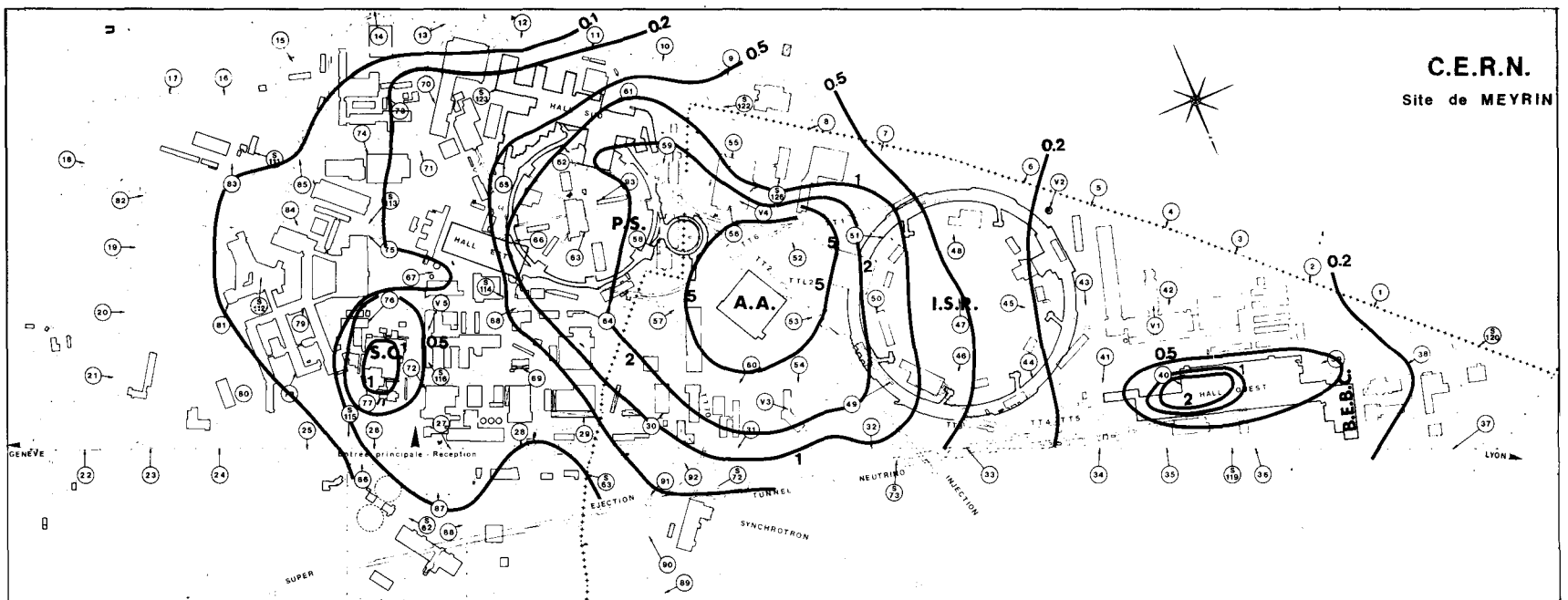

Fig. 1 Isodose distribution for 1985 (total dose in mSv) CERN Meyrin site

THE RESPONSE OF SURVEY METERS TO PULSED RADIATION FIELDS†

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The response of most survey meters to steady radiation fields is fairly well known and documented. However hardly any data is available in the literature regarding the response of these instruments to pulsed radiation. Pulsed radiation fields are encountered e.g., in the vicinity of linear electron accelerators or klystrons. Linear accelerators can deliver pulses up to a few microseconds long at repetition rates of a few hundred Hz. The fraction of operating time during which the beam is on (i.e., pulse width × repetition rate) is called the duty factor (D.F.). Typically duty factors for linear electron accelerators are less than 0.001. These small duty factors impose severe limitations on the radiation detection instruments as is shown below. The peak or instantaneous intensity (I_p) will be $I_p = I_{av}/D.F.$, or more than 1000 times the average intensity (I_{av}).

An instrument that ordinarily responds well to the average dose rate spread out evenly in time may not be able to cope with such a high dose rate. Instruments which have long dead times such as Geiger Mueller and proportional counters tend to become saturated in such fields and only count repetition rate. Ionization chambers are less influenced however, they must be operated with adequate voltage to overcome recombination losses. Scintillation survey meters may become non-linear at higher dose rates for pulsed radiation because the photomultiplier cannot handle the instantaneous currents that are required. Because of the need to test the response of different radiation detection instruments to pulsed fields, a pulsed x-ray facility has been built (I_p87). A brief description of this facility is given below along with tests of several different instruments.

Figure 1 shows a partial cut-away view of the pulsed x-ray facility. The major part of the x-ray tube is the electron gun which provides a stream of pulsed electrons that can be accelerated towards a combined target-window located directly below it. The window consists of aluminum 510 μm thick plated on the vacuum side with a layer of gold 6.4 μm thick. The frequency of the electron pulses can be varied by an internal pulser from 60 to 360 Hz with pulse widths from 360 ns to 5 us. The pulse amplitude can be varied over a wide range of currents. An external pulser can be used to obtain other frequencies or special pulse shapes. The voltage across the gun can be varied from 0 to 100 kV. The major part of the x-ray tube is enclosed in a large walk-in cabinet made of plywood lined with 0.32 cm thick lead, thus adequate shielding is provided.

A precision ionization chamber* (Farmer type) mounted directly below the x-ray tube facilitates remote readout of radiation levels

*Model 30-351, Victorean Inc., Cleveland, OH 44104

inside the cabinet. Experimental data indicated that this chamber and the Radcal ion chamber* (used with a 2025AC radiation monitor) were in good agreement therefore, the readings taken with the Radcal are considered to be the "true dose rate". The Radcal has a fairly uniform response between 30 keV and 1.33 MeV.

A brief study was made of the response of a cylindrical GM counter to pulsed radiation. Because of a non-pulsed dark current from the x-ray tube, it was not possible to study this in great detail. However, 5 mR/hr of non-pulsed x-rays at 80 kV gave 410 counts/second. An additional 5 mR/hr of pulsed x-rays at 100 Hz (pulse width = 2 μS) gave an additional 87 ± 11 counts/second. This shows that even at 5 mR/hr the GM counter is essentially only counting pulse repetition rate.

The "Radector II" consists of a pressurized argon filled ionization chamber (Neher White) which has a uniform response for photon energies between 80 keV and 1.2 MeV. Its response to pulsed x-rays (pulse width = 1 μS) at 100 kV (with aluminum filtration 1.6 mm thick) at 100 Hz is shown in Figure 2. The response departs significantly from linearity for dose rates above 1 mGy/h. This is probably inherent in Neher-White chambers due to the low collecting voltage. Part of the spectrum of the pulsed x-rays was below the flat response range of the Radector and this caused it to read only 45% as high as the Radcal at low dose rates.

The SLAC scintillation survey meter is built in house and consists of a Bicron air equivalent plastic scintillator doped with arscenic to increase the effective atomic number to that of air. Its decay constant is in the nanosecond range. Figure 3 shows its corrected response to pulsed x-rays (pulse width = 2 μS) at 70 kV and 120 Hz. The response begins to deviate from linearity above 1.0 mGy/h. There are several causes of non-linearity in photomultipliers. It is not certain but we believe the cause in this case is due to the photocathode resistance. The data has been corrected for the small difference in spectral response of the instrument from the ionization chamber.

The Xetex 303B is specifically designed for use around pulsed radiation sources. It consists essentially of a calcium fluoride scintillator and a vacuum photodiode and has a response fairly independent of energy between 60 keV and 2 MeV. It's response to pulsed x-rays (pulse width = 2 μS) at 70 kV and 120 Hz is shown in Figure 4. The response is quite linear up to 40 mGy/h. The data is corrected for the somewhat low response of the 303B to this x-ray spectrum.

*Model 20X5-180 Radcal Corporation, Monrovia, CA 91016.

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†Work supported by the U.S. Department of Energy under Contract DE-AC03-76SF00515.

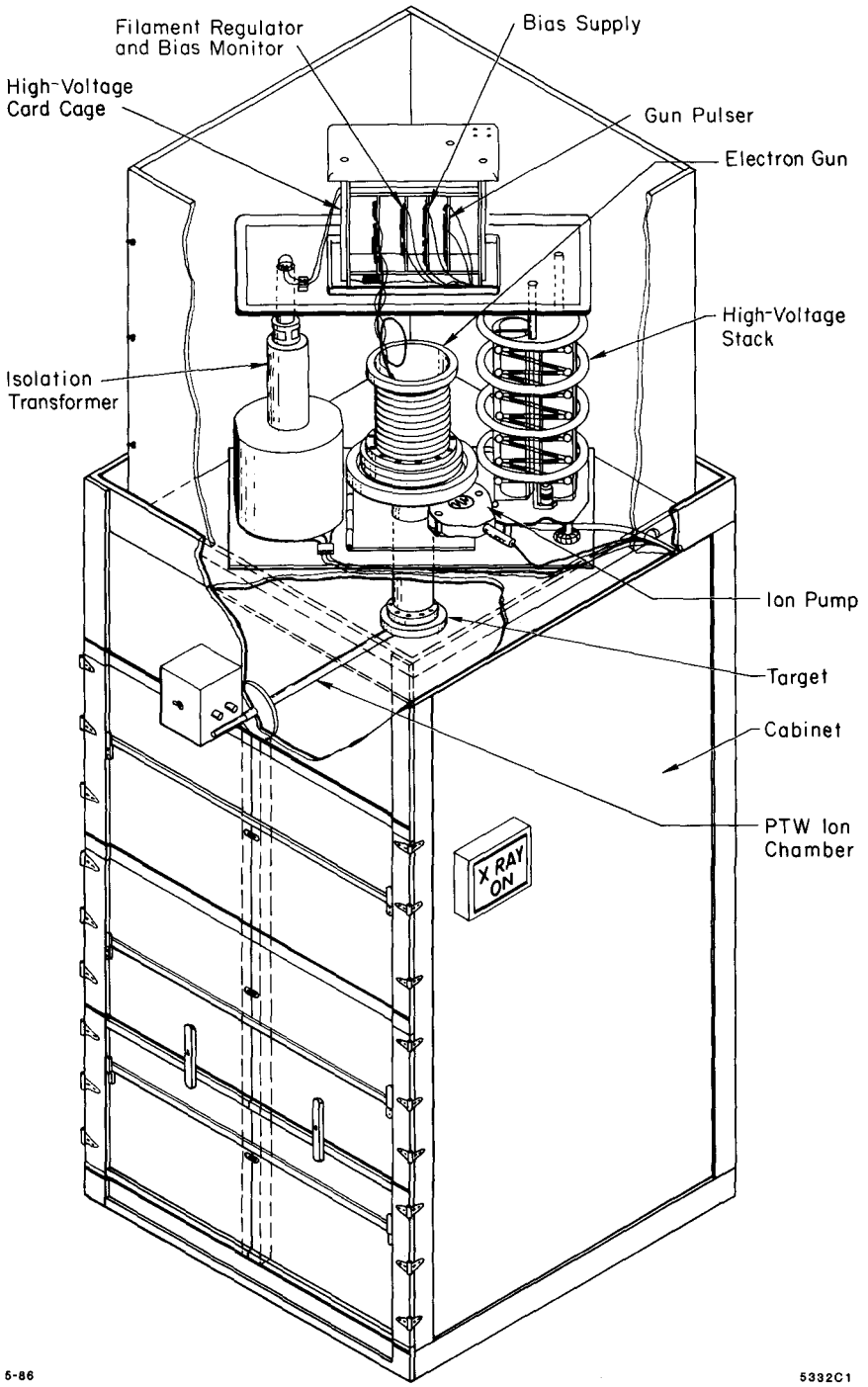

Fig. 1. The SLAC pulsed X-ray Facility.

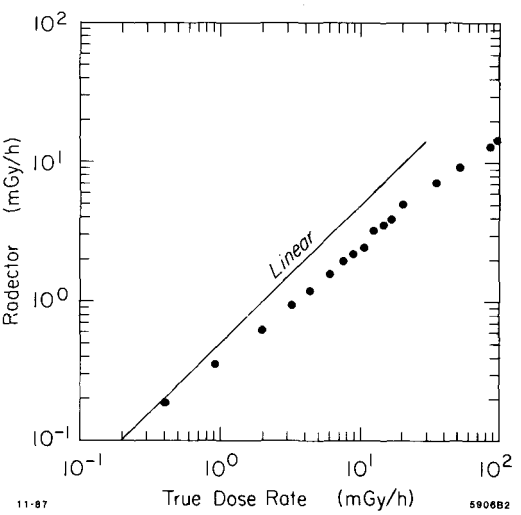

Fig. 2. Response of Radector.

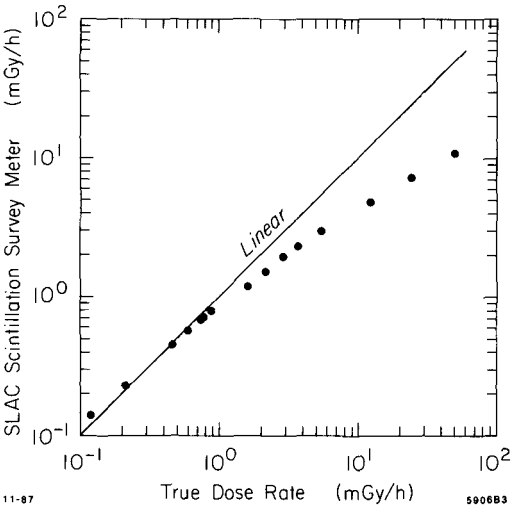

Fig. 3. Response of SLAC Scintillation Survey Meter.

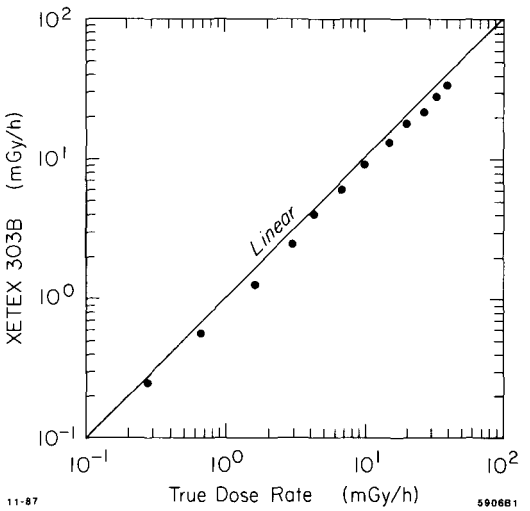

Fig. 4. Response of XETEX 303B.

RADIATION ENVIRONMENT IN THE TUNNEL OF A HIGH-ENERGY PROTON ACCELERATOR AT ENERGIES NEAR 1 TeV

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Neutron energy spectra, fluence distributions and rates in the FNAL Tevatron tunnel are summarized. This work has application to radiation damage to electronics and research equipment at high energy accelerators, as well as to radiological protection. Preliminary studies and related work are described elsewhere [1, 2, 3].

EXPERIMENTAL ARRANGEMENT

Figures 1 and 2 show the experimental arrangement in the A-17 region of the Tevatron tunnel. A room-temperature straight section (11.95 m) is shown which had a controlled N_2 leak in order to study neutron production as a function of beam-gas interaction rate during coasting beam conditions. Pressure was measured by a calibrated gauge near the leak and was controlled over the range 10^{-8} to 10^{-5} torr. The pressure in the adjoining cryogenic magnet chains, about 1×10^{-11} torr or less, was negligible by comparison. A "triangular" N_2 pressure distribution was assumed, *i.e.*, a linear decline in both directions from the leak, going to zero at the cryogenic interfaces. The interaction rate within the warm section was calculated to be 1.02×10^{-2} interactions per g cm^{-2} of N_2 and per passing proton, based on a total cross section for N of 238 mb.

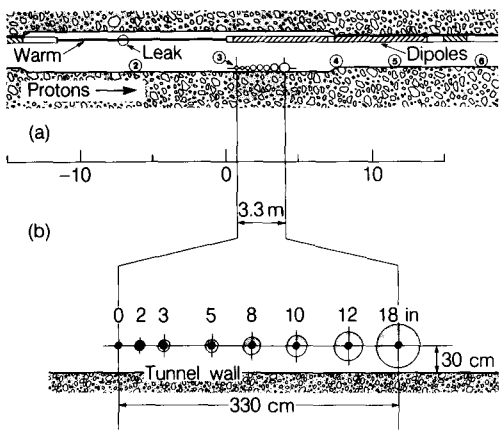

Fig. 1. Plan view of experimental setup in the tunnel near A-17. (a) overall view; (b) spectrometer geometry.

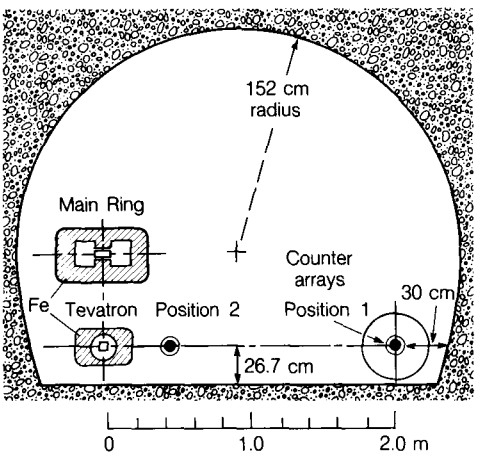

Fig. 2. Tunnel cross section near A-17.

A Bonner multisphere neutron spectrometer [4] was used, consisting of eight LiI(Eu) scintillation crystals (12.7 mm diam $\times$ 12.7 mm high), enriched to 99% ^{6}Li, and surrounded by spherical polyethylene neutron moderators ranging in diameter from 0 to 45.72 cm. These detectors were coupled through photomultiplier tubes to an 8-input 4096-channel analyzer. For spectral unfolding, the response functions of Sanna were used [5] in the program LOUHI [6]. For some runs the eight detectors were fitted with identical 12.7-cm spheres to study the longitudinal distribution of neutron fluence.

The Tevatron and Main Ring (MR) share a common tunnel (Fig. 2) [7]. Because the MR produced elevated radiation backgrounds, detectors were gated off during MR operation which accelerated protons from 8 to 150 GeV. A standard current toroid served as beam monitor.

Preliminary measurements were made in 1985 at a different location (A-48; not shown here), a low-loss region (small β) representing "quiet" coasting-beam conditions of accelerator operation. Here, the spectrometer was deployed 14.2 - 17.6 m downstream of a 4-m long warm section containing a nominal pressure of 2×10^{-8} torr. Measurements were made during Tevatron operation at 150 and 800 GeV, and during MR operation (only) at 150 GeV.

NEUTRON FIELD IN THE TUNNEL

Fluence rates as functions of gas pressure were measured along the tunnel over the range $-19 \leq z \leq 35$ m using identical 12.7-cm moderators surrounding each LiI scintillator. Figure 3 shows an example of a linear fit relating the counting rate to the gas pressure in the warm straight section for a 900-GeV proton beam. The slopes and intercepts from such fits for each location were converted to fluence rates per proton *passing* the measurement point and are plotted separately to show the distribution along z in Fig. 4. It is evident that the slope data (Figs. 3, 4a) are related to primary interactions within the warm section. There are variations (S. D. $\approx$ 20%) which are not understood between the measurement sets shown (Fig. 4a). The longitudinal distribution of intercepts (Fig. 4b) was found to be very similar in shape to that of the slopes, but the variance is much larger.

Representative neutron spectra made using the Bonner spectrometer under five different conditions are shown in Fig. 5. The remarkable features of these spectra are: (a) the prominent peak in all distributions that lies in the few-hundred keV region; (b) the relatively few high-energy neutrons (about 10% and 1% of fluence above 2 and 50 MeV, respectively); and (c) the average quality factor in the range $\bar{Q} = 6.9$ - 7.6.

With the spectrometer deployed as in Fig. 1 during 900-GeV operation, the *slopes* from fluence-vs pressure plots (Fig. 3) were used as input data to LOUHI. This procedure gives the spectrum of neutrons *unambiguously* produced in cascades initiated by primary interactions of protons on N_2 of the warm section. The prominent peak is centered at about 360 keV. The spectrum derived from *intercepts* of the same plots is related to primary interactions on materials in the vicinity other than introduced N_2. The peak is centered at about 75 keV and can be compared with spectra obtained downstream of A-48 (1985) at a gas pressure such that interactions in the warm section should not dominate. The spectra shown are for 150 and 800 GeV Tevatron operation and for MR-only operation at 150 GeV. The peaks are centered at 280, 135 and 240 keV, respec-

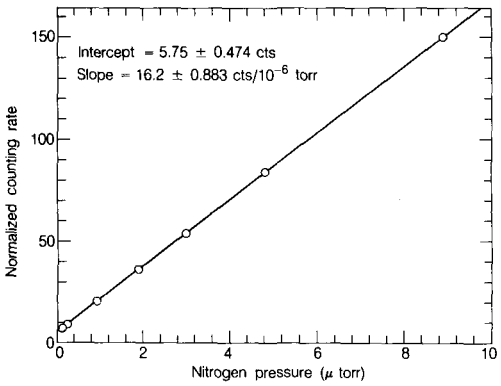

Fig. 3. Example of neutron fluence at 2m plotted *vs* N_2 pressure in warm section A-17 for 900-GeV protons.

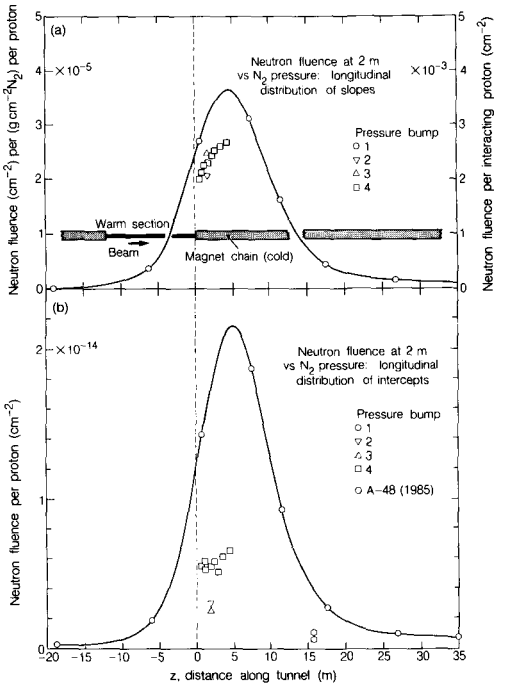

Fig. 4. Neutron fluence at 2m from beam line: (a) Per *passing* 900-GeV proton and unit N_2 target thickness (left ordinate) or per *interacting* proton (right ordinate), as derived from slopes of fluence-pressure plots; (b) per *passing* proton, as derived from intercepts of fluence-pressure plots.

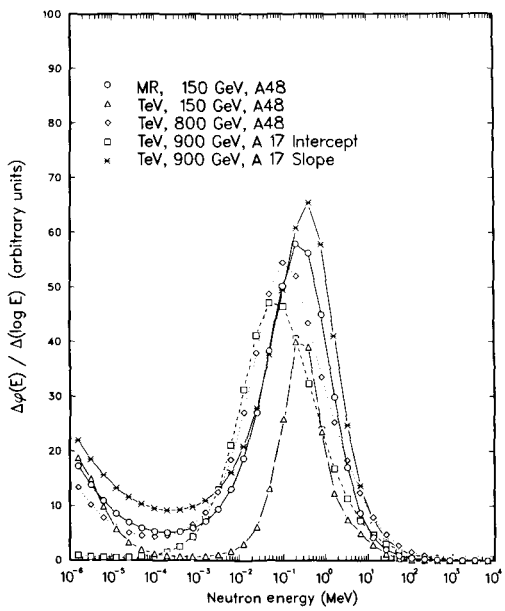

Fig. 5. Unfolded fluence spectrum at 2m from beam line for four types of Tevatron run and one Main-Ring run.

tively. We do not fully understand the differences between peaks but point out that the primary interactions that do not occur on N_2 most likely occur either on H_2 or He in cryogenic sections or with machine materials.

Monte-Carlo simulations performed by Gabriel et al. (not shown here) are in good agreement with the spectrum of Fig. 5 derived from slopes [3]. They furthermore indicate that about 80% of the neutrons at 200 cm from the beam line are albedo neutrons, *i.e.*, scattered from tunnel walls. This was tested experimentally by repeating certain runs at 39 cm from the beam line (Position 2, Fig. 2). The result was consistent with the expected radial distribution for the direct fluence, assuming that the albedo fluence is uniform across the tunnel section.

Integration over the "slope" z-distribution of Fig. 4a and correcting for the albedo fluence gives a value 10 neutrons produced per 900-GeV proton passing A-17 and per g cm^{-2} of N_2 target. A similar calculation based on an integration over the intercept z-distribution (Fig. 4b) gives 7.4×10^{-9} neutrons produced per passing 900-GeV proton. We caution the reader that this latter value is not well understood, is subject to the vagueries of machine operation and will likely vary widely from place to place around a given accelerator ring.

The above observations suggest a common "filter" for the neutrons, regardless of the nature of the original interactions which produce the parent cascade. Prominent parts of the filter must be the iron magnet yokes as well as the concrete tunnel lining. The neutron field dominates the tunnel radiation field in terms of absorbed dose to tissue. Because of their capability of producing lattice defects and transmutations, neutrons of energy $E_n \geq 150$ keV are the most important potential cause of radiation damage to solid state electronic devices in accelerator tunnel environments similar to those studied [1].

This work was partially supported by the U. S. Department of Energy under Contract No. DE-AC03-76SF00098 with the University of California.

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25 YEARS RADIATION PROTECTION PRACTICE AT LOW ENERGY PARTICLE ACCELERATORS

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ABSTRACT

Small high yield particle accelerators, such as neutron generators and minicyclotrons, are used in various areas of physical, chemical and medical research. Since they are strong sources of neutrons and γ-radiation a number of radiation protection problems must be solved to achieve a proper protection of workers and environment. In our paper we give recommendations to solve these problems according to our long practice.

The main problems at neutron generators concern the behaviour of tritium. Our paper informs about our experiences with the leakage of tritium into the environment with respect to transport, storage, installation and change of high activity tritium targets. Furthermore, we inform about the release of tritium to the environment during normal operation of neutron generators and our measures to reduce the releases. Finally, the procedures suitable for the conditioning of tritium in pump oils, cooling water and absorption cartridges are described. These procedures were also applied to the safe decommissioning of a neutron generator at the end of 15 years of operation.

At minicyclotrons three radiation protection problems are dominating: proper design of shields, induced activity and exposure to workers. In our paper we give recommendations for shield design on the basis of our extensive attenuation measurements performed in the last twenty years. Furthermore we give recommendations for the planning of Ventilation Systems which are in harmony with the ALARA-principle. Since minicylotrons are mainly used to provide short-lived radionuclides the handling of the radionuclides possibly results in high whole body and hand doses to workers. Our paper gives a dose record for the years 1971 to 1988.

NEUTRON YIELD OF MEDICAL ELECTRON ACCELERATORS

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Shielding calculations for medical electron accelerators above about 10 MeV require some knowledge of the neutron emission from the machine. This knowledge might come from the manufacturer's specifications or from published measurements of the neutron leakage of that particular model and energy of accelerator. In principle, the yield can be calculated if details of the accelerator design are known (1). These details are often not available because the manufacturer considers them proprietary. A broader knowledge of neutron emission would be useful and it is the purpose of this paper to present such information.

Patterson (2) reported that a fast neutron source placed in a cavity with thick concrete walls produced a nearly uniform field of thermal neutrons inside the cavity. They found that the thermal neutron fluence rate is given by the simple relation

$$\dot{\phi} = \frac{kQ}{S} \qquad (1)$$

where $\dot{\phi}$ = thermal neutron fluence rate (n cm^{-2} sec^{-1})
Q = fast neutron emission rate (n s^{-1})
S = inside surface area of the cavity (cm^{-2})
and k is a constant equal to 1.26 ± 0.10.

A similar relationship has been found for scattered fast neutrons (3,4) and for scattered photons (5) when the source is a gamma ray source. While Eq. 1 would be expected to be strictly valid only for spherical rooms, Patterson found that it worked well for a cubical cavity. McCall (4) used the Mobte Carlo Code MORSE and measurements to show that for both thermal and scattered fast neutrons, the typical radiation therapy electron accelerator rooms also gave adequate agreement with this representation. An obvious modification is to rewrite Eq. 1 so that Q is the number of neutrons produced per photon rad delivered at the isocenter (n/rad) and $\dot{\phi}$ is (n cm^{-2} rad^{-1}). Using the above, it is possible to measure the thermal neutron fluence per photon rad and the dimensions of the room and calculate the neutron yield of the accelerator. It should be noted that if the thermal neutron detectors are calibrated by an exposure in a cavity in concrete with the aid of Eq. 1 and then used in the therapy room measurements, k cancels out in the calculations.

During the course of the last 10 years, the author has made measurements of the neutron yield, Q, for many accelerators of different types. Some measurements were made while doing neutron surveys of the accelerators as a consultant. The rest, and larger fraction, have been made by mailing gold foils to medical physicists and asking them to make an exposure. The gold foils, along with the appropriate information concerning the exposure and rooms, was then mailed back for counting. The author is very grateful to the large number of people who have assisted in this project.

The gold foils used, 2.54 cm diameter by 0.00254 cm thick, were counted with a pancake G-M counter in a lead shield. Sufficient counting time was always employed so that counting statistics contributed less than ±3% S.D. The overall precision of the measurements is difficult to estimate. From MORSE calculations, it is believed that variation in room size and shape did not contribute more than ±10% S.D. Variations in the calibration of the accelerator output should conservatively be less than ±5% S.D. Normally, the Cd difference method was not used. Measurements in 10 therapy rooms gave a Cd ratio of 2.1 ± 0.2 which was indistinguishably different from that in our calibration facility. An overall precision of ±15% S.D. is a reasonable estimate.

Systematic errors are also involved. The foils were calibrated in a concrete cavity with a calibrated ^{238}PuBe source. Any error in this value is reflected throughout the measurements. It is possible that variations in concrete composition can produce different values of the constant k. This might be a function of the chemical composition of local sand and aggregates.

Table I and Table II give the results of measurements on electron linear accelerators and betatrons, respectively. In general, nominally identical accelerators gave very similar neutron yields, with a few exceptions. Popular machines tended to have more nearly the same neutron yield than those where only a few were made. This was presumably because the less popular machines were still undergoing engineering changes from one serial number to the next. It should be noted that the numbers in the column labelled "Energy" are those provided by the user. Often the manufacturer guarantees a certain depth dose characteristic e.g., percentage of maximum dose rate at 10 cm depth in water and the listed energy is only nominal.

It is striking that there is so much variation in neutron yield from one manufacturer to another for machines operated at the same energy - e.g., by a factor of 1.5-3.0 at 18 to 25 MeV. This is believed due to the following reasons:

1. Varying beam loss before the electrons strike the target.
2. Choice of material for the target and flattner.
3. Deviation from the nominal energy in order to attain the desired depth dose performance.

These results are good enough for many shielding calculations, e.g., when it is contemplated replacing an existing accelerator with a higher energy machine in the same room.

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Table I. Neutron Yield of Linear Accelerators

Type	Energy (MeV)	Neutron Yield Per Photon Rad At Isocenter	Comments
Varian Clinac 18	10	6.5×10^{8}	
	10	5.7×10^{8}	
	10	5.6×10^{8}	
Toshiba LMR-15	10	4.6×10^{9}	
Siemens Mevatron XX	14	7.7×10^{9}	
	14	9.8×10^{9}	
	14	8.2×10^{9}	
	14	8.0×10^{9}	
	14	8.0×10^{9}	
Varian Clinac 20	15	1.7×10^{10}	Water Phantom in Beam
	15	9.7×10^{9}	
	15	8.9×10^{9}	
Varian Clinac 20	18	2.0×10^{10}	Neutron Shielding in Therapy Head
	18	2.2×10^{10}	
	18	3.5×10^{10}	No Neutron Shielding
	18	3.5×10^{10}	
Mitsubishi	18	2.3×10^{10}	
Varian Clinac 1800	18	2.9×10^{10}	
	18	2.8×10^{10}	
	18	3.0×10^{10}	
	18	2.9×10^{10}	
AECL Saturne (Therac 20)	18	4.1×10^{10}	Heavy Concrete Unknown
	18	5.8×10^{10}	
	18	5.0×10^{10}	
	18	4.8×10^{10}	
	18	5.5×10^{10}	
Philips SL-75-20	15	1.5×10^{10}	
	16	7.9×10^{10}	
	17	1.1×10^{11}	
	18	8.0×10^{10}	Borated Polyethylene Door in Room
	18	7.1×10^{10}	
Siemens KD	18	2.2×10^{10}	
	20	3.2×10^{10}	Foil was Close to Large Polyethylene Door
	20	2.4×10^{10}	
	20	2.7×10^{10}	
Varian Clinac 2500	24	2.8×10^{10}	
	24	3.1×10^{10}	

Table I (Cont.)

Type	Energy (MeV)	Neutron Yield Per Photon Rad At Isocenter	Comments
Varian Clinac 35	25	1.2×10^{11}	Older Design
	25	6.6×10^{10}	Ilmenite Concrete Ceiling and Wall
	25	6.6×10^{10}	
CGR Sagittaire	25	4.8×10^{10}	
	25	5.2×10^{10}	
	25	4.5×10^{10}	
	25	6.5×10^{10}	
	25	5.5×10^{10}	
	25	3.8×10^{11}	Made in USA of Diff. Manufact.
CGR Saturne	25	1.0×10^{11}	
AECL Therac 25	25	6.0×10^{10}	
	25	4.1×10^{10}	
Philips SL-25	25	8.6×10^{10}	
	25	6.4×10^{10}	

Table II. Neutron Yields of Betatrons

Type	Energy (MeV)	Neutron Yield Per Photon Rad @ 1m From Target	Comments
Siemens	18	1.8×10^{10}	Barite Concrete Inserts
	18	1.1×10^{10}	
	18	3.4×10^{10}	
Allis-Chalmers	22	5.6×10^{9}	
	24	1.3×10^{10}	
	25	1.1×10^{10}	
	25	8.8×10^{9}	
Shimadzu	25	1.4×10^{10}	
Brown Boveri	32	6.5×10^{9}	
Siemens	42	3.2×10^{9}	Barite Concrete
	42	5.3×10^{9}	
Brown Boveri	45	3.7×10^{9}	
	45	2.0×10^{9}	Heavy Concrete Unknown
	45	3.3×10^{9}	
Scanditronix*	50	6.5×10^{10}	
	50	6.7×10^{10}	

*Note: These two machines are microtrons rather than betatrons.

ρ-Be FAST NEUTRON ACTIVATION EXPERIMENT OF CONCRETE

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ABSTRACT

Concrete is widely used as a shielding material of a radiation operating facility, But induced radioactivuty in concrete irradiated by neutrons becomes a new radiation source, which increase the radiation exposure of maintenance workers and the radioactive waste. These are the problems at facility operation and dismantling. The development and popularization of large scale accelerators makes these problems more serious while they produce many high energy neutrons.

We measured the induced radioactivuty of concrete irradiated by neutrons, which were produced by 20, 30 and 40-MeV proton bombardment of a beryllium target, The neutron spectra were also measured, and the experiment was performed at the SF cyclotron of the Institute for Nuclear Study, University of tokyo.

ACTIVATION PRODUCTS OF A HEAVY ION ACCELERATOR

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Abstract: Heavy ions - carbons to uranium - are accelerated up to 20 MeV/μ by the UNILAC (universal linear accelerator).

Parts of the accelerater - the beam ducts, slits, apertures, Faraday-cups or the experimental devices are sometimes hit by the accelerated ions. The induced radioactivity is analyzed by γ-spectrometry.

Shortly after the irradiation a lot of nuclides is found - partly depending of the accelerated ions - whereas after a cooling time of some month the determined nuclides are mostly known as activation products of proton accelerators.

The UNILAC is able to accelerate heavy ions - carbon to uranium - up to 20 MeV/μ. Almost each of the elements has been accelerated during the past years. A large part of the beam time very heavy ions - heavier than gold - are sent to the experimental devices.

γ-Spectra of parts dismantled from the accelerator or the beam pipes were measured. Short time after the end of irradiation nuclides as Se-75, Rb-83, Ag-110m or some Hf-isotopes are found depending on the material of the irradiated part and the accelerated ions. Generally only the well known nuclides near to the iron region are seen after some month of cooling time (1), as it is shown in the figure.

There will be given some examples of γ-spectra as a function of cooling time.

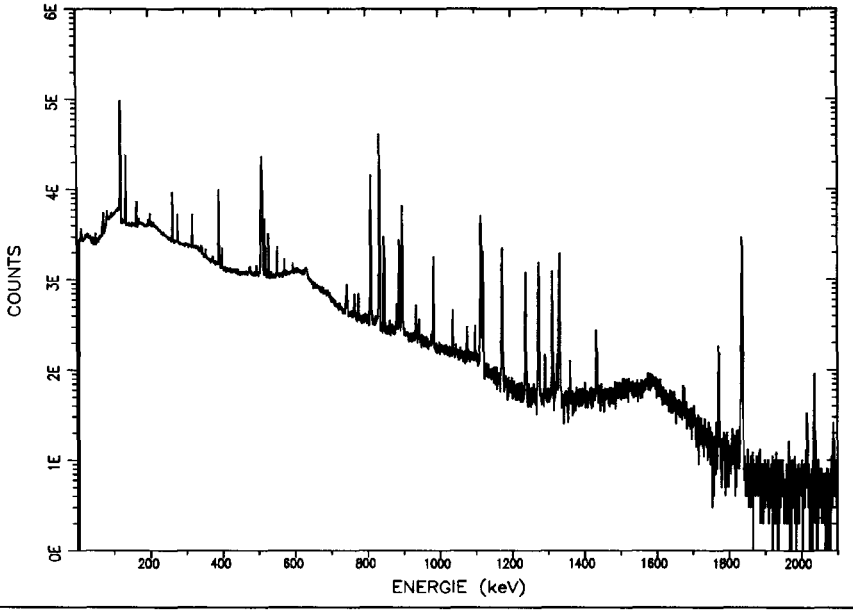
It has not been tried to calculate the activity of the irradiated parts on behalf of the variety of accelerated ions, different energies from 5 MeV/μ to about 20 MeV/μ and the unknown portion of the beam hitting slits etc.

1.) F. F. Szlavik, Activation of a 42 MeV Cyclotron; Health Physics of Radiation Generating Machines; CONF 86 02 106

bellows dismantled January 29th, 1987

measured February 26th, 1987

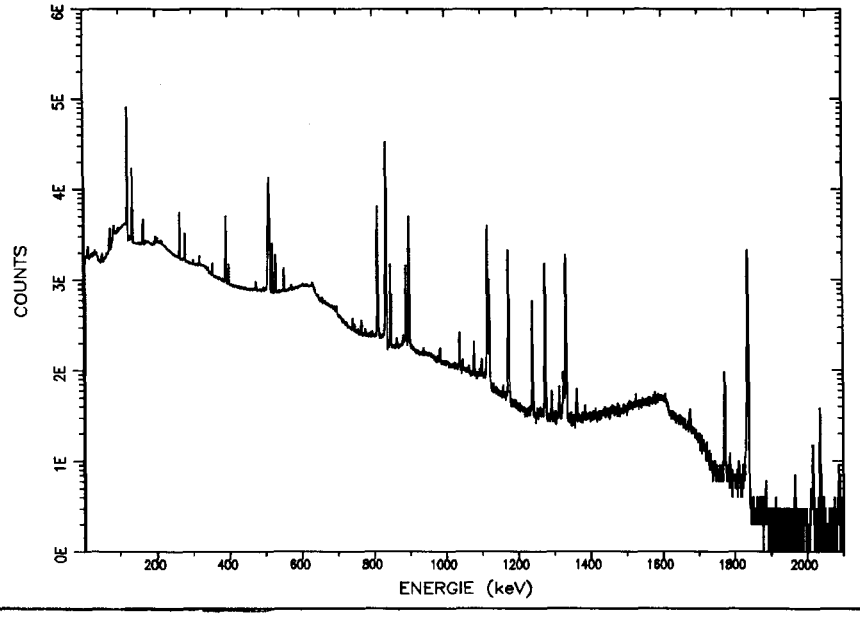
measuring time: 10515 s



bellows dismantled January 29th, 1987

measured May 18th, 1987

measuring time: 10515 s



EFFECT OF HYDROGENOUS SHIELDS ON THE AVERAGE ENERGY OF NEUTRON SPECTRA*

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In a previous work[1], the authors and colleagues studied problems of photoneutron transport in concrete rooms using the Monte Carlo transport program, MORSE[2] and supporting experimental measurements. From this work, an empirical method of calculation was developed based on the average energy, $\bar{E}$, of the neutron spectrum. This method has been described in NCRP 79[3]. Using this method, it is possible to calculate the effect of metal shielding and also scattering inside a concrete room based on $\bar{E}$, and to provide a suitable conversion factor for converting fluence to either dose equivalent or to absorbed dose. For subsequent shielding, TVL's for polyethylene or concrete were given for two different geometries, one from the accelerator moderated by a heavy metal (sphere geometry) and the other from these same spectra modified by room scattering (long geometry). It should be noted that these TVL's are for shields where the thickness is great enough that there is an exponential decrease, and thus would overestimate attenuation in the first few centimeters (see, for example, Fig. 1). The neutron spectra used in this paper are as follows: 15W = photoneutrons from 15 MeV on tungsten, 25W = photoneutrons from 25 MeV on tungsten, Cf = ^{252}Cf and 14W4W = photoneutrons from 14 MeV on tungsten surrounded by 4 inches (10 cm) of tungsten.

While the TVL's given in NCRP 79 are correct for a given $\bar{E}$, provided the shield is thick enough, they do not give any insight into how the average energy of the neutrons changes with depth in either concrete or polyethylene shields. It was felt that some such information would be informative, and is the subject of this paper.

When penetrating heavy elements, such as iron, lead or tungten, the average energy of the neutron spectrum decreases nearly exponentially at first, and then approaches a constant value at large thicknesses. This is expected since the major mechanism for energy loss is through non-elastic scattering [inealstic, (n,2n), (n,p), etc.]. In hydrogenous materials, however, the energy loss in most energy ranges is dominated by elastic scattering on hydrogen nuclei. Thus, the macroscopic behavior may be quite different.

Using the MORSE code, we have studied photoneutrons penetrating heavy metals such as might be found in a medical accelerator, and then subsequently striking a concrete or polyethylene shield. These spectra have been used in two different geometries. The first is a spherically symmetric shield such as might be added around the accelerator head to reduce neutron fluence, or might be applicable to the accelerator room itself which has a direct view of the accelerator. The second geometry is that of a long tunnel or duct with the absorber at the end, representing the door at the outer end of a maze. In the former case, the neutron spectrum

*Work supported by the U.S. Department of Energy Contract DE-AC03-76SF00515.

incident on the absorber is that from the accelerator (through the heavy metal shield) plus the room scattered neutrons. For this case, a range of values for $\bar{E}$ are of interest representing the different source terms. In the latter case, only room scattered neutrons reach the absorber. For this case, only a single value for $\bar{E}$ is needed since the $\bar{E}$ of neutrons reaching a door down a maze is always in the neighborhood of 100 keV regardless of the accelerator energy.

We have not found any simple empirical relations to describe the effect of hydrogenous shields on $\bar{E}$. Instead, we present some graphical data which should be of some help in analyzing measurements. The spherical geometry results are shown in Fig. 2 for concrete and Fig. 3 for polyethylene.

The graphs cover the range of thicknesses which might be required for medical accelerator shielding. From the figures, we note that there is always an initial decrease in $\bar{E}$ followed by a slow hardening of the spectrum with increasing depth in the shield. This is consistent with the removal of the lower energy neutrons early in the shield.

Figure 4 shows the average energy outside a polyethylene shield in a geometry such as a door into a maze. Again, we see the rapid decrease in the $\bar{E}$ followed by a gradual increase in $\bar{E}$. The attenuation of dose equivalent through such a door closely follows an exponential with the TVL's as given in NCRP 79. When using a fluence detector to determine neutron leakage through the door, it may be easier to measure with the door open and calculate the result with the door closed rather than try to correct for the effects of the changing spectrum through the door.

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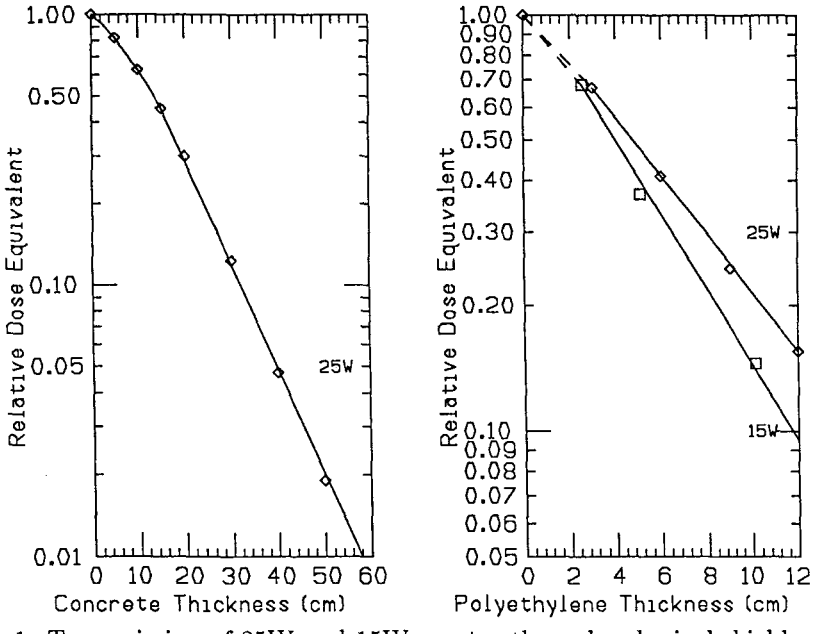

Figure 1. Transmission of 25W and 15W spectra through spherical shields.

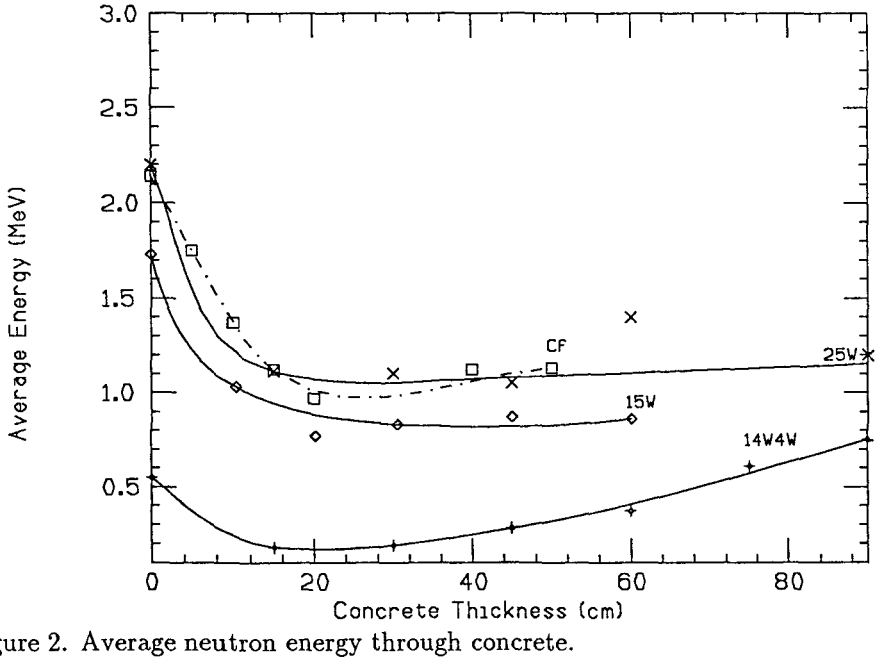

Figure 2. Average neutron energy through concrete.

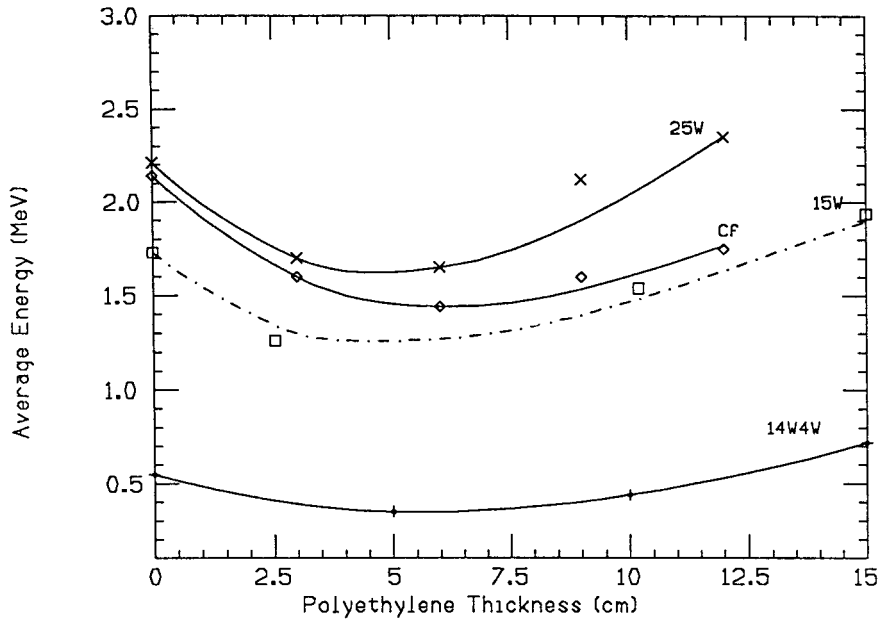

Figure 3. Average neutron energy through polyethylene.

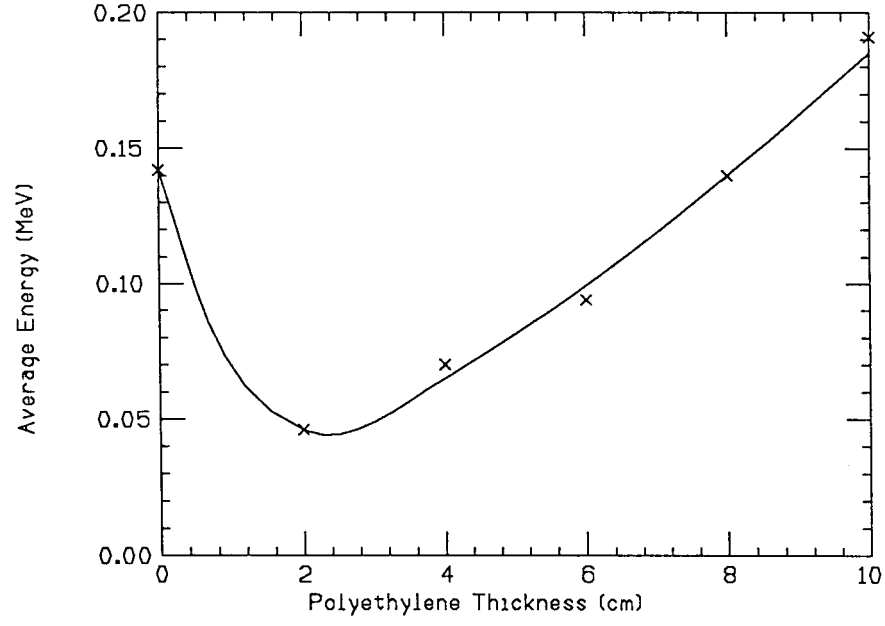

Figure 4. Average neutron energy in polyethylene for a scattered neutron spectrum ($\bar{E} = 140$ keV) as an input.

BROAD BEAM ATTENUATION AND HALF VALUE LAYER DETERMINATION IN BARYT CONCRETE FOR 150-400 KV X-RAYS

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1. INTRODUCTION

Using broad beam geometry, the attenuatio properties for baryt concrete, made out of domastic baryt and on our own prescription (1), have been studied for constant Y-rays potentials from 150-400 KV. The density of the used baryt concrete was $3,6x10^3$ kg/m^3. From the experimentally obtained attenuation curves, the half value were determined, and presented in the paper.

2. GEOMETRY, EQUIPMENT AND INSTRUMENTATION

The measuring geometry is shown in Figure 1. This geometry is to some extend result of our own experience in this field (1) but also it meets all requirements of the IEC recommendations for determination of the protective properties of materials used for the protective shielding against X-radiation of radiation quality up to 400 KV (3).

As a source of X-radiatio two types od X-ray equipments, with constant tube voltage, were used. Physical characteristics of the useful beam are shown in table 1.

The attenuation material consisted of sheets of baryt concrete 50x50 cm square. Its thickners varied from 1 to 5 cm, and density of samples was $3,6x10^3$ kg/m^3.

Data were taken to establish adequate size of the beam to insure "broad beam" conditions. Results are shown in Figure 2. A 40cm square field was found to be adequate one, what agrees rather well with other authors.

Measurements were performed by Victoreen ionisation chambers model 130, 227, 683 nad 228.

3. RESULTS

The Baryt concrete attenuation curves for kilovotages from 150 to 300 KV are shown in Figure 3, and those for 300, 350 and 400 KV in Figure 4.

From these experimentaly obtained curves the half value layers for the examined baryt concrete were determined. Otained results

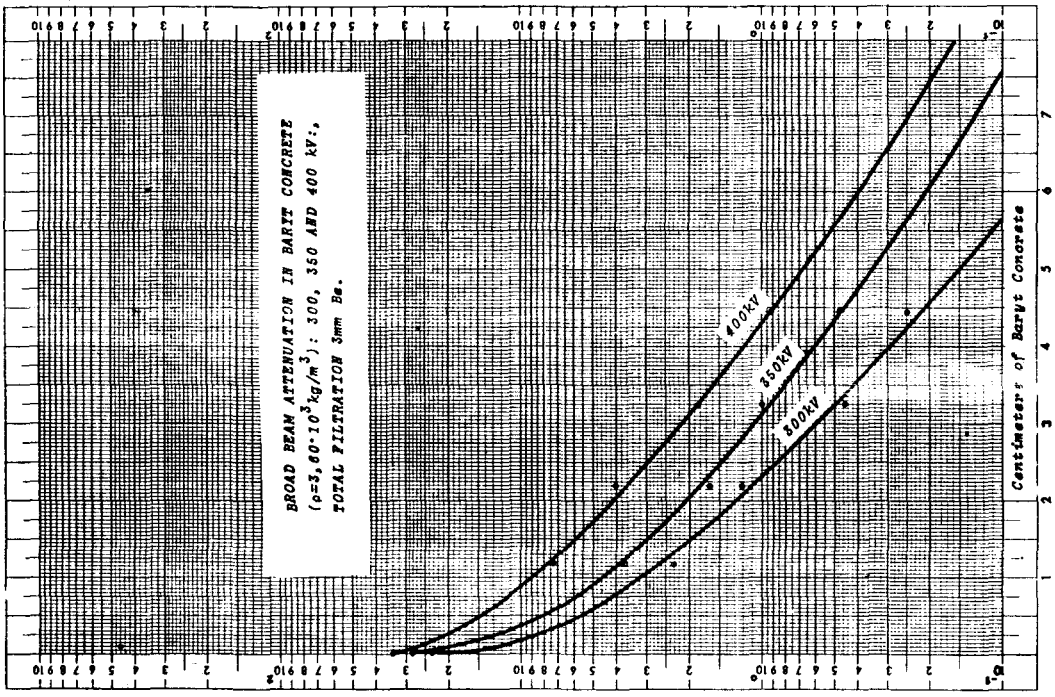

Fig.4. Broad beam attenuation in Baryt Concrete

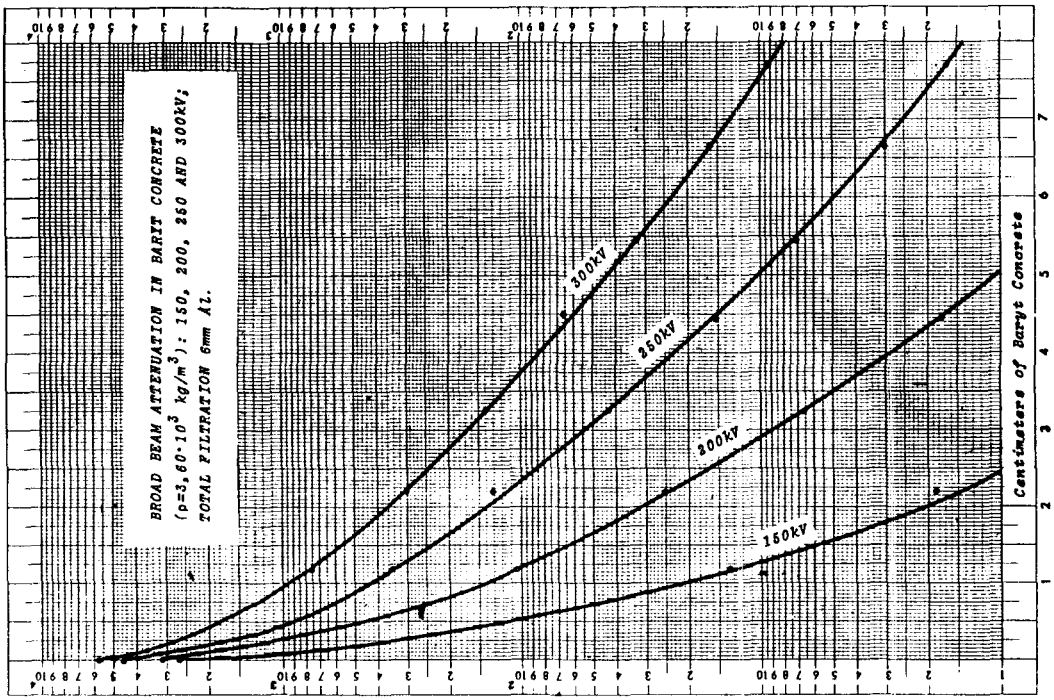

Fig.3. Broad beam attenuation in Baryt Concrete

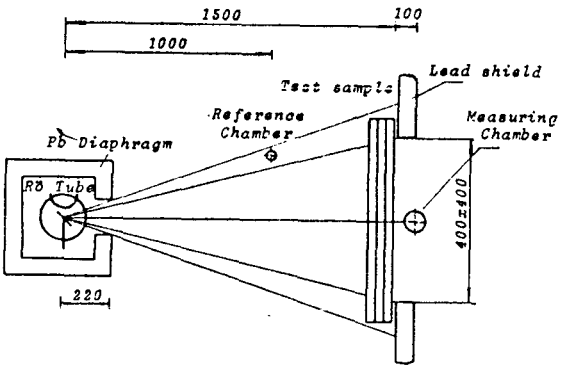

Fig.1. Geometry for Baryt Concrete broad beam attenuation measurements.

Table 1. Physical characteristics of the useful beam

KV	Total filtration	First half-value layer $(d_{1/2})_1$ in mm	Second half-value layer $(d_{1/2})_2$ in mm	Homogenity coefficient $(d_{1/2})_2/(d_{1/2})_1$	C/kg·s·mA at 1 m	Rö tube type
150		27,00	73,75	2,73	1617,10	
200	6 mm Al	34,25	101,00	2,95	2805,23	CMA 306
250		42,00	125,00	2,98	4009,00	(ANDREX)
300		45,50	160,00	3,52	5567,44	
300		10,00	23,00	2,30	18366,41	"ISOVOLT 430"
350	3 mm Be	12,50	28,30	2,26	23741,14	(SEIFERT)
400		15,20	32,70	2,15	27397,73	

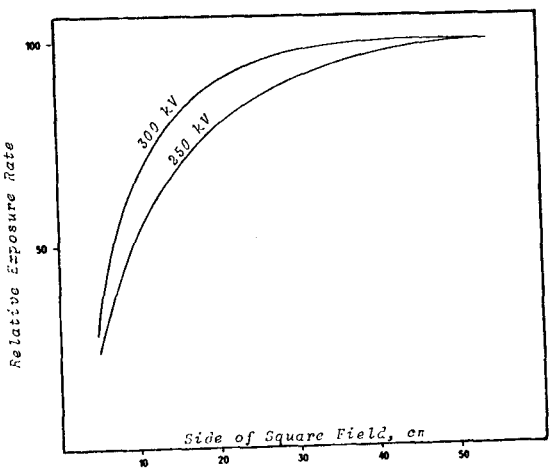

Fig.2. Relative transmitted Exposure Rate as a function of size of square field at 5,46 cm Baryt Concrete Barier.

are presented in table 2. These values were obtained for the end points of attenuation curves.

Table 2. Half-layers values for baryt concrete ($\rho = 3.6 \times 10^3$ kg/m^3) for broad X-ray beam

Voltage (kV)	Values of half-layers (cm)	Remark
150	1.250	Rö tube: CMA 306
200	2.750	Filtration: 6 mm Al
250	4.275	Radiation beam: 40x40 cm
300	6.065	
300	5.950	Rö tube: "ISOVOLT 420"
350	7.275	Filtration: 3 mm Be
400	8.350	Radiation beam: 45x45 cm

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NEW GAMMA-RAY BUILDUP FACTORS INCLUDING THE EFFECT OF BREMSSTRAHLUNG

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ABSTRACT

Despite the simple manner by which the transport equation can be derived, the solution of this equation is, however, difficult to obtain on an exact form unless certain restrictions are to be applied, which may consequently lead to unefficient solutions. This reason, together with the complicated calculations to be done every time the transport equation is to be solved, led to the development of the concept of the buildup factors. Applying the concept of the buildup factors, the radiation transport calculations can be easily reduced to the use of the buildup factors for the nuclear responses being calculated.

The concept of the buildup factors has been considered since 1951 and a large set of buildup factors was obtained. In most of those calculations, an important process in the Gamma-ray transport through media, which is the bremsstrahlung process through which secondary photons are emitted by energetic electrons, was neglected.

The effect of the bremsstrahlung on the buildup factors was investigated in a somewhat approximated manner through a few papers. Nearly, in all of them, the process was considered as the average of multi-collision processes between the electron and the medium atoms. However, because of the importance of this effect, especially at high photon energies and for heavy materials, a more exact treatment is developed in this paper.

In this treatment, the bremsstrahlung is treated as a single-collision process. The calculations are carried out using the Monte Carlo method, in which the complete photon-electron-photon cascade is considered. The dose-buildup factors are calculated for different source geometeries: Point isotropic, plane normal and plane inclined sources and with photon energies of 6.0, 8.0 and 10.0-Mev. Two media are selected, lead and tungesten, as irradiation media.

The results show that the values of the buildup factors are increased by a factor ranging from 120% to more than 200% depending on the initial photon energy, the medium material, and the depth inside the medium. Good agreement is achieved in comparing the results with the experiments.

HUMAN RESPIRATORY TRACT MODEL FOR RADIOLOGICAL PROTECTION - A REVISION OF THE ICRP DOSIMETRIC MODEL FOR THE RESPIRATORY SYSTEM

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The need for standardized values for various parameters describing the inhalation, deposition, retention and translocation of airborne radionuclides in workers for the purpose of deriving exposure limits was addressed in Tripartite Conferences on Radiation Protection from 1949 to 1953 (NVO 1984). At conferences held at Chalk River, Canada, September 29-30, 1949, and at Arden House in Harriman, New York, March 30-April 1, 1953, agreement was reached on a model for calculating radiation doses resulting from inhalation of radioactive aerosols when specific data were not available. This model was used in the 1959 report of the ICRP Committee II on Permissible Dose for Internal Radiation (ICRP 1959). It was assumed that 50% of an inhaled aerosol is deposited in the upper respiratory tract, 25% is exhaled and 25% is retained in the lungs. For soluble particles, it was assumed that the 25% is absorbed and translocated to other tissues in the body. For insoluble particles it was assumed that 12.5% is cleared in 24 hours and the remaining 12.5% is retained in the lungs with a half-time of 120 days.

This simple model of deposition, retention and clearance of inhaled aerosols was the basis for the limits for exposure to radionuclides and for calculations of doses to exposed individuals for both assessment and predictive purposes until ICRP Publication 30 appeared in 1979, using much more sophisticated dosimetric and metabolic models (ICRP 1979). The model used was a slight modification of that published in 1966 by a special Task Group on Lung Dynamics of ICRP Committee II, chaired by Dr. Paul Morrow (ICRP 1966). Major innovations were introduced by this task group, including a deposition model not only based on but using dust sampling data. Deposition was described for three anatomical compartments, which had both physiological and radiobiological implications. The model made possible consideration of both particle size of inspired aerosols and respiratory rate with respect to fraction deposited in each region: nasal, bronchial and pulmonary.

A quantitative kinetic clearance model was introduced that accounted for material deposited in each of the three regions. Perhaps of greatest impact was the classification of chemical compounds according to estimates of their expected tendency to be retained in the respiratory tract. This is the D, W, Y classification; D class for those compounds expected to be cleared from the respiratory tract with a half-time less than 1 day; W class for compounds with clearance half-time of a few days to months; and Y class for compounds that are expected to be retained with half-time of 6 months to years. This model also provided for the transfer of inhaled particles to the thoracic lymph nodes. The effort of the task group was a major scientific accomplishment. It used and expanded upon the total relevant technical data available and reflected the outstanding expertise and extraordinary insight of the members.

*Work supported by the U.S. Department of Energy under Contract DE-AC06-76RLO 1830

The deposition and retention models developed by the task group provided a sound scientific basis for ICRP Committee II to calculate radiation doses from inhaled radionuclides leading to recommendations for Annual Limits on Intake and Derived Air Concentrations that appeared in Publication 30. Task group models went further than Committee II was prepared to go in some areas, such as the transfer of radionuclides to thoracic lymph nodes (Committee II elected to combine the lymph nodes and lungs), but failed to address others, such as the nasal passages. However, the omissions were insignificant compared with the magnitude of the advances in knowledge of inhaled particles stimulated by the task group's report. After publication of the 1966 report, research on the deposition, retention, clearance and translocation of inhaled aerosols intensified with studies on both humans and experimental animals. This new knowledge led to the ICRP's decision to initiate a review of the dosimetric model for the respiratory tract and a possible revision or development of a new model. In 1984, the ICRP appointed a task group of Committee II to undertake the assignment. The members of the task group were selected to assure that the broad spectrum of biological, physiological, chemical, and radiological aspects of inhaled radioactive aerosols could be competently addressed. The members of the multinational task group are Fredrick Cross, Richard Cuddihy, Peter Gehr, Anthony James, John Johnson, Roland Masse, Monique Roy, Willi Stahlhofen, and William Bair, Chairman. In addition, numerous corresponding members were invited to provide technical assistance and to review drafts of the report. The following describes the status of the effort in October 1987. The dosimetry model eventually adopted by the ICRP may be quite different, since much work remains to be done.

Following a review of the current ICRP lung model, a basis was established for a revision. Principal inadequacies of the current model would be addressed, such as calculation of radiation doses to the nasal and oral passages; replacement of the D, W, and Y classification system for clearance of inhaled materials where adequate information is available; and calculation of doses for inhalation of gases. The revised model would use new knowledge of deposition and retention of very small particles (well below 0.1 μm diameter), regional deposition of inhaled particles, the distribution of and absorption of inhaled gases, and clearance kinetics for numerous radioactive compounds determined in humans and experimental animals. Knowledge of the morphology and the physiology of the respiratory tract has increased, the relative regional sensitivities of the respiratory tract to cancer induction is better understood, and dosimetry modeling concepts and approaches have greatly expanded. The major developments in computer technology during the last few years have opened numerous possibilities for not only modeling the intake of radioactive materials but also utilizing the model for projecting and assessing radiation doses. The task group determined that a new model should facilitate calculation of biologically meaningful doses; should be consistent with morphological, physiological and radiobiological characteristics of the respiratory tract; should incorporate current knowledge; meet radiation protection needs and be user-friendly by not being too sophisticated; should be adaptable to development of computer software to allow calculation of relevant radiation doses from knowledge of a few readily measured exposure parameters; should be equally useful for assessment purposes as for calculating ALIs; should apply to all members of the world population; and should consider the influence of smoking, air pollutants and disease.

The task group's approach was to converge separately developed morphological, radiobiological, physiological, deposition, clearance, dosimetric, and

bioassay considerations into a comprehensive multiparameter dosimetric model for the complete respiratory tract. All will be addressed in considerable detail in the report. An anatomical representation of the model is shown in Figure 1, along with the regional compartments that can be identified with respect to measuring deposition, clearance and retention of inhaled aerosols and to the occurrence of neoplastic diseases, the somatic effects of principal concern in radiation protection. These compartments are the extrathoracic, ET, comprising the nose, mouth, pharynx and larynx; the thoracic fast-clearing region, T(f), comprising the 0 through 16th airway generation (trachea through the terminal bronchioli); and the thoracic slow-clearing region, T(s), comprising the 17th through 23rd airway generation (respiratory bronchioli through alveolar sacs) plus the thoracic lymph nodes.

Since the revised lung model is to be applicable to essentially all members of the world's population under both working and nonworking conditions, reference values for numerous physiological parameters will be given. Lung volumes, flow rates and respiratory rates either will be given for Caucasian, Asian and African male and females of varying height and weight as a function of age, or guidance will be provided for deriving these values for use in calculating radiation doses to the respiratory tract under conditions of interest. The model will apply to both oral and nasal breathing, separately and in combination; e.g., at ventilation rates several times the resting value, it will be assumed that half of the intake is through the nose and half through the mouth. The task group considers it necessary that the dosimetric model accommodate a wide range of physiological parameters because they influence the amount of radioactive gases and particles inspired from a contaminated environment, as well as regional and total deposition of particles within the respiratory tract.

The proposed revised model addresses inhalability of aerosols and deposition in extrathoracic tissues such as the nasal passages, pharynx, larynx, and vocal cords. Deposition in the T(f) region is assumed to include material rapidly cleared essentially by mechanical processes from airway generations 0 through 16 (Figure 1). Deposition in the T(s) region includes material slowly cleared from airway generations 17 through 23 by both mechanical and solubilization processes as well as material infinitely retained such as in lymphatic tissues. Calculation of deposition of particles in these regions as well as in various tissues within these regions will be based on morphometric models and experimental data from human subjects inhaling test aerosols over a broad range of particle sizes.

In the proposed model, clearance of particles is competitive, occurring either by mechanical or absorption processes, and is assumed to be nonlinear, with excretion a time-varying factor of the residual amount. Mechanical clearance rates will be obtained from studies with human subjects. For compounds for which reliable human data exist or for which data can be extrapolated from animal experiments, the model will use observed rates of absorption. For other compounds, default values will be used based on the current D, W, and Y classification system. Mathematical models for calculating radiation doses to various tissues of the respiratory tract will be developed incorporating expressions describing the deposition and retention of radionuclides. Rather than treating the lung and lymph nodes as a single organ and calculating an average dose, the revised model will provide for calculating doses to tissues in all anatomical regions identified in Figure 1. To provide some perspective to the calculated doses, the task group will assess the relative sensitivity of these tissues to the induction of cancer

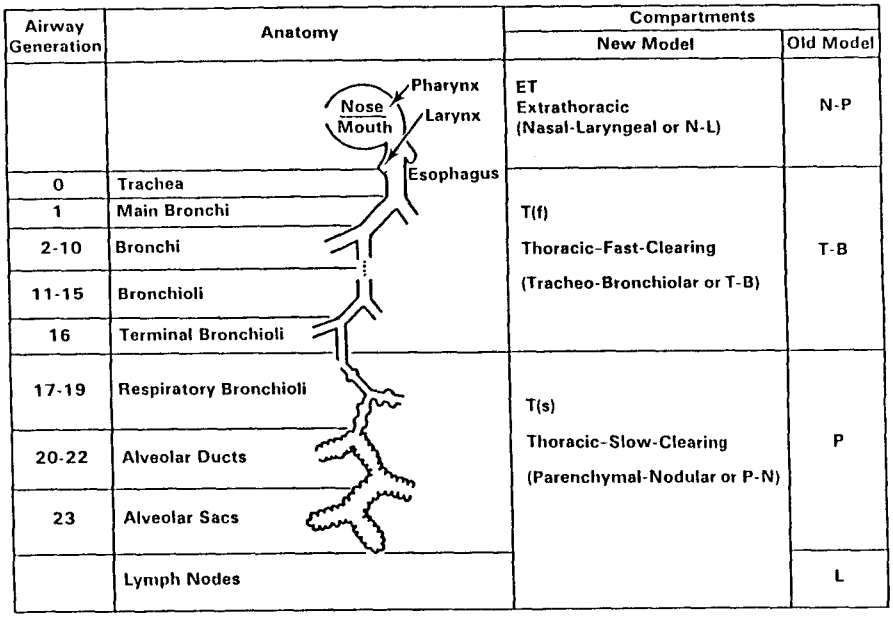

FIGURE 1. Proposed Revised ICRP Lung Model

by radiation. This will also be useful in future partitioning a weighting factor that might be assigned to the total respiratory tract for calculating effective dose equivalent. The calculation of doses will follow the method of ICRP 30 in which the committed dose equivalent in a target tissue is determined by the energy absorbed per unit mass from the radiation emitted from a source organ. Compared with the current model, the proposed model is expected to simplify calculating respiratory tract doses from bioassay data. The task group expects computer software will facilitate but not be necessary for using the revised dosimetric model for the respiratory tract.

This work is being coordinated with a similar effort undertaken by the National Council on Radiation Protection and Measurements in the United States to minimize differences in the respiratory tract dosimetric models adopted by the two organizations.

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OSTEOSARCOMA INDUCTION BY PLUTONIUM-239, AMERICIUM-241 AND NEPTUNIUM-237: THE PROBLEM OF DERIVING RISK ESTIMATES FOR MAN

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Spontaneous bone cancer (osteosarcoma) represents only about 0.3% of all human cancers, but it is well known to be inducible in humans by internal contamination with Radium-226 (1) and Radium-224 (2). Plutonium-239, Americium-241 and Neptunium-237 form, or will form, the principal long-lived alpha particle emitting components of high activity waste and "burnt-up" nuclear fuel elements. These three nuclides deposit extensively in human bone and although, fortunately, no case of a human osteosarcoma induced by any of these nuclides is known, evidence from animal studies suggests that all three are more effective than Radium-226 in inducing osteosarcoma (3,4,5,6). The assumption that the ratio of the Risk Factors, the number of osteosarcoma expected per 10000 person/animal Gy, for Radium-226 and any other bone-seeking alpha-emitter will be independent of animal species has formed the basis of all the important studies of the radiotoxicity of actinide nuclides in experimental animals (7). The aim of this communication is to review the Risk Factors which may be calculated from the various animal studies carried out over the last thirty years with Plutonium-239, Americium-241 and Neptunium-237 and to consider the problems which may arise in extrapolating these Risk Factors to homo sapiens.

Data on osteosarcoma induction by each of the three actinide nuclides are available only for rats, but information for Plutonium-239, 238 and Americium-241 has been obtained in mice and beagles. Some of the beagle studies are not yet completed and some of the rodent data were obtained only at relatively high radionuclide doses; however, sufficient information is now available to make a provisional evaluation of the likely Risk Factors for man for the three most important actinide components of the nuclear fuel cycle.

The radiotoxicity of Plutonium-239 and Americium-241 in rats has been studied by Soviet workers (4,8) and by Taylor et al (3,9) following intravenous injection of activities which yielded an averaged radiation dose to bone of about 7 Gy. For each of these nuclides the various sets of data may be combined to yield single Risk Factors for Plutonium-239 and Americium-241 of 440 (Standard Deviation (SD) 50) and 230 (SD 30) respectively. The difference between these values is statistically significant and suggests that Plutonium-239 is 1.9 (SD 0.2) times more effective in inducing osteosarcoma than Americium-241.

Data for Neptunium-237 are much less extensive and come from studies by Soviet workers (6) and by Wirth (10). A single large US study following inhalation of Neptunium-237 yielded no osteosarcomas (11). From the Soviet and German data Risk Factors for the male and female rat combined have been calculated as 1200 (SD 150) at an average bone radiation dose of 3-5 Gy; 2900 (SD 450) at 0.7-1.0 Gy and 3700 (SD 1200) at <0.15 Gy. These results suggest that Neptunium-237 may be more carcinogenic than Plutonium-239 and that there may be a trend towards higher Risk Factors at lower average absorbed alpha doses to bone.

Studies of the relative radiotoxicity of Plutonium-239 and Americium-241 in mice have been made by the Salt Lake City Group (12,13). These studies show

no clear evidence for an effect of radiation dose to bone on the calculated Risk Factor and, for averaged doses to bone in the range 3 to 13 Gy, Risk Factors of 730 (SD 70) for Plutonium-239 and 270 (SD 50) for Americium-241 have been calculated. In this species Plutonium-239 appears to be 2.7 (SD 0.2) times more effective than Americium-241 for the induction of bone sarcoma.

The most valuable information will come from the as yet uncompleted beagle studies in the United States. However, due to the kindness of the groups in Salt Lake City and Richland in publishing extensive data on the progress of their studies it is possible to make provisional estimates of the Risk Factors for Plutonium-239, Americium-241 and Radium-226 at various dose levels. It must be emphasised that the values which will be discussed in the following paragraphs are provisional since they are based on the present author's analysis of data which have not yet been fully analysed by the original investigators.

The Risk Factors calculated for beagles are shown graphically in Figure 1. All three nuclides show Risk Factors which are significantly higher at low average alpha radiation doses to bone, as compared to higher bone doses. This effect is most marked for Plutonium-239 and is least for

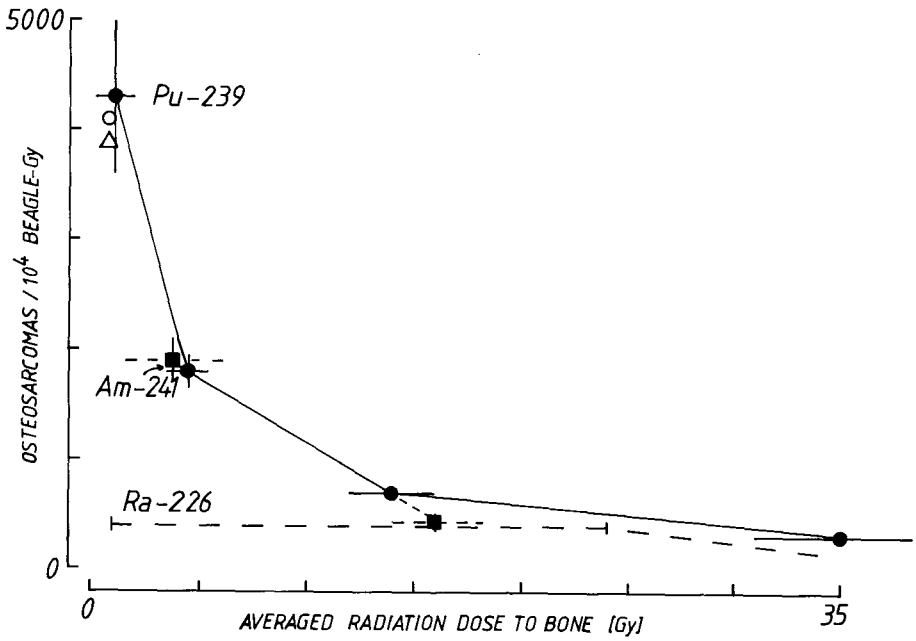

FIGURE 1 Calculated Risk Factors for osteosarcoma induction in beagles as a function of radionuclide and average alpha radiation dose to bone. Closed symbols - intravenous injection (5), open symbols - inhalation (14).

Radium-226. Particularly noteworthy are the very different ranges of bone dose over which the maximum risk is observed for the different nuclides. For Plutonium-239 the dose range is very narrow, about 0.1 to 2 Gy, as compared to about 1.5 to 6 Gy for Americium-241. For Radium-226 the calculated Risk Factor is similar (400 SD 60) up to about 25 Gy but reduces to about 170 and about 60 at average bone doses of 40 and 155 Gy respectively. At average bone doses of about 1.2 Gy Plutonium-239 appears to be about twice as carcinogenic as Americium-241 (Pu/Am = 2.3 (SD 0.2)), but at the two higher doses shown in Figure 1 this difference disappears. The risk Factor for Plutonium-239 in the bone dose range 0.1 to 2 Gy for the Utah beagles, which received single intravenous injections (5), is closely similar to those which may be calculated from the data of Park (14) for beagles which inhaled either Plutonium-239 nitrate or Plutonium-238 oxide (these values are shown as open symbols in Figure 1).

All the Risk Factors discussed above were calculated for adult animals, for young adult dogs Mays et al (7) have calculated a Risk Factor of 7700 (SD 1200) at bone doses <1 Gy, a value about double that shown in Figure 1. Thus the possibility that age may significantly affect the Risk Factor cannot be excluded.

The quite large differences in the ranges of averaged bone dose over which Plutonium-239, Americium-241 and Radium-226 apparently exhibit their maximum risk highlights the importance of a detailed understanding of the microscopic distribution of alpha particle radiation dose and the development of cell-specific dosimetry for each bone-seeking radionuclide (15). Such models are under active investigation for the above nuclides in beagles, but for Neptunium-237 it appears probable that any such cell-specific model will have to be developed from studies in rat bone, a tissue with characteristics rather different from human bone.

The data in Figure 1 show that for radiation protection purposes it is necessary to calculate the Risk Factors from animal studies involving the lowest meaningful bone dose levels available. It must also be recognised that the Risk Ratios for pairs of radionuclides may also vary with the averaged alpha dose to bone. From the human data for Radium-226 (16) at bone doses of <14 Gy, a Risk Factor of between 50 and 100 may be calculated, 4 to 8 times lower than that observed in beagles, and of the same order as the difference in the spontaneous incidence of osteosarcoma in beagle and human. Thus using the risk ratio concept, and assuming that at very low bone doses the relative carcinogenicity of Plutonium-239 and Neptunium-237 are similar in rat, beagle and man, the following Risk Factors for the three actinides in humans are proposed:

Plutonium-239, 238: 1000 to 2000 osteosarcomas/10000 person-Gy

Americium-241: 500 to 1000 osteosarcomas/10000 person-Gy

Neptunium-237: 3000 to 6000 osteosarcomas/10000 person-Gy.

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LYMPHATIC CANCERS AND LEUKEMIA AMONG PLUTONIUM WORKERS

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ABSTRACT

Recent studies of cancer incidence and mortality among plutonium workers suggest elevated risks for certain lymphatic cancers and leukemias. An investigation of cancer incidence among Los Alamos National Laboratory male employees resulted in standardized incidence ratios of 2.49 (90 per cent confidence limits: 0.98, 5.23) when compared with cancer rates for the state of New Mexico. A study of mortality among 5413 white males employed for at least two years at a plutonium weapons production facility found fewer deaths than expected for all causes of death, all cancers and lung cancer when compared with U.S. death rates. No bone cancer was observed; however, an excess of brain tumors was found for the cohort in general. Comparisons of plutonium burdened with unburdened workers showed elevated rate ratios for all lymphatic and hematopoietic cancers combined (RR = 9.86, 90 per cent confidence limits = 1.26, 94.03) and for all causes of death combined (RR = 1.33, 90 per cent confidence limits = 1.05, 1.68) at 5 years induction time. Rate ratios were also suggestively elevated for esophageal, gastric and prostatic cancers. Although standardized rate ratios for several causes of death at several induction times increased with increasing exposure category, no overall linear dose response trends were found. These findings imply that increased risks for several types of cancers cannot be ruled out at this time for plutonium burdened individuals.

EPIDEMIOLOGICAL FOLLOW-UP OF URANIUM MINERS IN CANADA

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INTRODUCTION

The hazards of exposure of uranium miners to high concentrations of radon and radon daughters have been recognized for several decades. Quantitative estimates of risk based on epidemiological followup of exposed miners are becoming increasingly available in recent years. The present report deals with four studies of Canadian miners; the results are compared with those of two other major studies of U.S. and Czechoslovakian uranium miners.

RISK COEFFICIENTS

Table 1 summarizes the quantitative estimates of risk of induced fatal lung cancers per WLM of exposure to short-lived radon daughters. Lung cancers appearing less than 10 years after first exposure are usually excluded in these calculations in order to ensure better comparability of the data from short and long term followup and to ensure that the risk coefficients are not too low because of the inclusion of years of followup during which no effect of radon daughter exposure is anticipated (2). These data thus represent the average risk coefficients observed between 10 years after first exposure and the end of the followup, at which time most of the miners were still alive.

A major problem in all these studies is the reliability of the exposure estimates. In general, exposure rates were highest during the earliest years of mining and were decreased, primarily by improved ventilation practices, in later years. However, monitoring of radon daughter concentrations was infrequent or absent during the earliest years of mining; the exposure histories of the early miners with the highest exposures and longest followup have necessarily been reconstructed from minimal data. In the study of 15 984 Ontario uranium miners, estimates of radon daughter concentrations were based on results of area monitoring in the mines in the early years or in cases where no reliable measurements were available, on estimates provided by three mining engineers who were familiar with the Ontario uranium mines over the early years of operation (8).

The risk coefficients derived from these studies (Table 1) vary over a range of about 10-fold. This is true both for estimates based on the relative risk model (expressed as % increase in normal incidence of lung cancer per WLM) and on the absolute or attributable risk model (expressed as excess number of lung cancers per 10^6 person-years at risk per WLM). Part of this variation is probably due to uncertainties in estimated exposures. The range of risk coefficients appears to be compatible with the average values of 1% increase in normal incidence of lung cancer per WLM or 10 lung cancers per 10^6 person-years per WLM that were suggested in ICRP Publication 50 (9).

MINIMUM LATENT PERIOD; EFFECT OF TIME AFTER EXPOSURE

Previous studies have in general not shown a significant excess of lung cancers within the first 10 years of followup after first exposure to radon

daughters in uranium mines. It is therefore of some interest to note that the studies of Beaverlodge and Port Radium miners both showed zero excess of lung cancers 0-5 years after first exposure but a statistically significant increase 5-10 years after first exposure (1,7).

The effect of age at first exposure has been examined in 3 of the Canadian studies. No consistent trends in the risk coefficients were observed (1, 5, 7). Three of the 4 Canadian studies showed a trend towards a decrease in relative risk coefficient with increase in age at the time of appearance of lung cancers (1,5,6). A similar trend is probably valid for the followup of U.S. uranium miners (10). A more sophisticated risk model was also tested in a recent reanalysis of the Ontario data by Muller and co-workers (11). In this model, the increase in relative risk was assumed to be 0% during the first 5 years after each increment in radon daughter exposure and best values for relative risk were calculated for subsequent times after each exposure. The available data on Ontario uranium miners suggested an increase in relative risk per WLM of 1.6% at 5-10 years, 3.4% at 10-15 years and 0.3% at 15 or more years after each annual increment in exposure (11) . The raw data from certain other studies can also be fitted by a similar model.

EFFECT OF CIGARETTE SMOKING

Approximately 70% of the miners in all groups listed in Table 1 were cigarette smokers. The Ontario miners' study included large groups of gold miners and of copper-nickel miners as well as uranium miners. An excess of lung cancers was observed in the group of gold miners as well as the uranium miners, but not in the group of copper-nickel miners, who are believed to have a similar smoking history. The major cause of lung cancer among the gold miners was cigarette smoking, as expected, with a relative risk of 7 for smokers compared to non-smokers (12). The excess lung cancers observed in the gold miners group appeared to be correlated with exposure to high ore dust levels in the earlier years of mining before 1945. Smoking also increased the risk of silicosis (12).

Interactions between cigarette smoking and inhalation of radon daughters were examined in the study of Newfoundland fluorspar miners. The data did not permit a clear choice between the relative risk model, indicative of a multiplicative interaction, and the absolute risk model, indicative of an additive interaction (5). Limited data from the Ontario study were compatible with the hypothesis of a multiplicative interaction (11). Other data have suggested that cigarette smoking accelerates the appearance of lung cancers in exposed miners, with the result that a multiplicative interaction between smoking and radon daughter exposure is most evident at the earlier stages of followup but absent or less evident many years after exposure (4, 13).

DOSE-RESPONSE RELATIONSHIP

The data from all 4 Canadian studies are compatible with a linear, non-threshold relationship between accumulated exposure and incidence of induced lung cancer. The fluorspar miners' study did not provide any evidence of a decreased response per unit dose at high accumulated exposures in the region of 2500 WLM (5), such as has been observed in the study of U.S. uranium miners (2, 4, 13).

LIFETIME RISK ESTIMATES

The excess risk of death from lung cancers before age 70 has been calculated for male uranium miners using both relative and absolute risk models, the expected values for non-exposed males being derived from Canadian

vital statistics (5,7). Using the average risk coefficients suggested in ICRP Publication 50 (9), the values are equivalent to a risk of 2.5-2.9 x 10^{-4} per WLM. A total lifetime risk of about 3 x 10^{-4} per WLM would appear to be a reasonable choice, assuming that exposure to radon daughters at any age resulted in an increased risk of lung cancer which continued throughout life. If the relative risk model were correct, the risk to non-smoking miners would of course be appreciably smaller (9). Assuming that increased risk ceased 15 years after each exposure, the lifetime risk would be closer to 1.7 x 10^{-4} per WLM for male miners exposed to 1 WLM per year from age 20 to age 55, based on the time-dependant relative risk coefficients which were derived from the study of Ontario miners. Further study of miners exposed at younger ages only would be required to substantiate this latter model.

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TABLE 1. ESTIMATES OF RISK OF INDUCED LUNG CANCER IN VARIOUS GROUPS OF MINERS

Group of miners	Average estimated exposure in WLM	Excess lung cancers appearing more than 10 years after first exposure		Reference
		% increase per WLM in normal incidence	Excess cancers per 10^6 person-years per WLM	
Eldorado Port Radium uranium	273[a]	0.27	3.1	1
U.S. uranium	1180	0.3-0.45 0.8-1.4[b]	3.5 6[b]	2,3 2,4
Nfld. fluorspar	548	0.9	6.4	5
Ontario uranium	33	1.0 1.3[c]	9.7 7.2[c]	6
Czech uranium, started 1948-52	310	1.8	19.	2
Eldorado Beaverlodge uranium	36[d]	3.3	21.	7

[a] 44% of the person-years at risk in the category with <5 WLM and with no excess lung cancers were excluded for this calculation.
[b] Calculated for groups receiving lower exposures (<360 WLM) only.
[c] Risk coefficients for the group of miners with no prior experience in gold mining. Data based on "special WLM" estimates (6,8) are not included.
[d] 74% of the person-years at risk in the category with <5 WLM and with no excess lung cancers were excluded for this calculation.

LUNG CANCER MORTALITY OF URANIUM MINERS IN FRANCE

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ABSTRACT

An epidemiological study of the uranium miners in France has been planned in order to study any excess of cancer mortality, mainly from bronchogenic lung cancer and to establish a relationship between this excess mortality and the radon daughters exposure. It includes about 2000 miners having entered the mines between 1947 and 1972. This cohort study is actually continuing, the dosimetric data of each miner, collected from different files, having to be verified and completed before being entered in the final analysis.

The vital state of these miners has been studied up to december 1985. We observed a significant excess mortality compared to the national population of the same period, excess of cancer mortality, and of lung cancer deaths. The exposure of the uranium miners having largely changed during the year 1956 when several radioprotection procedures have been introduced such as : ventilation, systematic individual recording of the exposure of each miner ... the excess mortality by lung cancer is discussed by studying separately the group of miners having worked underground before 1956 and those having entered the mine between 1956 and 1972.

INTRODUCTION

The epidemiological study of the French uranium underground miners has been organized in order to study any excess risk of cancer mortality, mainly of bronchogenic lung cancer, in relation to the occupational exposure. The studies on uranium miners in the U.S.A. (1) ,(2), CANADA (3), CSSR (4) showed an excess risk of lung cancer mortality for those miners having cumulated more than 100 WLM* during their working life.The French study gives us the opportunity to study this risk of lung cancer mortality in function of relatively low annual exposures compared to those of the U.S. miners; the mean annual exposure of the French underground miners is comparable to that of the Canadian miners, but the mean duration of underground uranium mining in Canada (less than two years) is shorter than in the French study. Consequently, this study may be able to give an estimation of an excess risk by WLM cumulated during the working period, but may also contribute to the study of an eventual excess risk linked to the annual dose level.

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* 1 WLM (working level month) corresponds to an exposure to a concentration of 1 WL for a period of 170 hours. 1 WL is equivalent to any combination of radon daughters in one liter of air that will result in the emission of $1,3 \times 10^5$ Mev of alpha energy in their complete decay through Polonium 214. This potential alpha energy will occur when 100 pico-curies of Radon-222 in one liter of air is in equilibrium with its daughter products.

The factors implicated in the etiology of lung cancer being numerous, some of them like social class and especially tobacco consumption ought to be taken in account during the study of the lung cancer risk of the uranium miners. Tobacco consumption may be registered after interviewing the uranium miners, if they are still alive, or by collecting this information in the medical files. Information from the family of the deceased may be biased if they know the cause of death. They unconsciouly may enhance information collected about a given factor, known to be carcinogenic.

METHODS

The protocol of this study has been described previously (5) . The cohort includes all the uranium miners having worked underground for more than three months and having entered the mines between 1947 and 1972. This study is part of the large dosimetric survey program of the French uranium miners, collecting the individual annual data of the exposure to short lived daughters of Radon-222, to long-lived radioactive dust, and to gamma radiation. The preliminary analysis of the cancer mortality of this cohort, in relation to occupational exposure, is limited to those miners having experienced more than two years of exposure to the short-lived daughters of Radon-222, and having a complete individual dosimetric survey. (n = 1652).

The mortality of this cohort is compared to that of the French national male population of the same calendar period, standardized on the age distribution by the indirect method of standardization. The observed number of deaths of a given cause is compared to the expected number and is presented in a ratio, called the standardized mortality ratio (SMR). The statistical significance of the observed excess (p - values) is calculated in a one-sided test, based on the hypothesis of a Poisson distribution of the observed values. The number of person-years at risk of dying (PYR) is calculated up to the date of death, or, for those still alive up to the actual endpoint of the survey period : the 31th December 1985.

The search of the causes of death presents some difficulties in France, mainly for those workers dying when retired (after age 55), or when having leaved the mines for already a long period. We have no legal possibility to access to the national files of the mortality data. The medical staff of the uranium mines makes large efforts in order to follow-up all of the miners.

Nevertheless, at the present point of the study, in the cohort of those having worked for more than two years, 14 % of the deceased miners have an unknown cause of death. 65 % of these unknown causes of death are observed in the group of underground miners having worked for less than 5 years, and having entered the mines before 1957.

RESULTS

During the period 1947-1985, the 1652 miners having worked underground for more than 2 years, have been followed up for a mean period of 26 years; their mean number of years of underground exposure is 13 years. At the end of 1985, 79,5 % of these miners where still alive their mean age being 57 years. Table 1 shows the results of the mortality of this cohort, and the cancer deaths significantly in excess in comparison to the national population.

Table 1 :

Total cohort (n = 1652) mean survey : 26 years

Mortality	Obs. Numb.	SMR	Statist. Signif.
All causes	339	1,20	$p < 10^{-3}$
All cancer	99	1,29	$p < 10^{-2}$
Lung cancer	41	2,41	$p < 10^{-4}$

Table 2 :

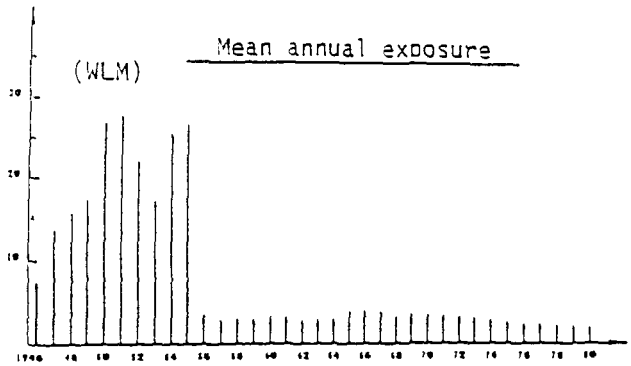

Table 3 :

Cohort having entered the mines before 1956 (n = 767)
mean survey = 28,6 years

Mortality	Obs. Numb.	SMR	Statist. Signif.
All causes	215	1,32	$p < 10^{-4}$
All cancer	62	1,41	$p < 10^{-2}$
Lung cancer	27	2,77	$p < 10^{-4}$

Table 4 :

Cohort having entered the mines between 1956 and 1972
(n = 885) mean survey : 23,7 years.

Mortality	Obs. Numb.	SMR	Statist. Signif.
All causes	124	1,03	N. S.
All cancer	37	1,14	N. S.
Lung cancer	14	1,93	p = 0,02

Table 2 presents the mean annual exposure to the radon daughters (in WLM*). It indicates an important modification during the period 1955 to 1956. In the year 1956 large radioprotection procedures have been introduced in the mines : ventilation, systematic recording of monthly exposure of each individual... During the period 1947-1955, the miners have indeed been exposed to a larger extent than during the following period. The exposure during this first period had to be estimated retrospectively, and it will be difficult to appreciate if this exposure has been overestimated for some individuals or if it is reflecting the real individual exposure of this period. Consequently, we decided to analyze separately the mortality of 2 subcohorts, separated in function of their date of first underground mining. Table 3 is indicating the mortality data of those having entered the mines before 1956; Table 4 includes the same results calculated on those having begun underground mining between 1956-1972.

DISCUSSION

The study of mortality of causes other than cancer having not yet been realized, we cannot conclude if all of the excess mortality is linked to cancer mortality or if part of it can be explained by violent deaths, as seen in most of the studies of underground miners. The excess of cancer mortality seen on the whole cohort, reflects essentially the results of the cohort having been underground before 1956. But this first cohort has a longer survey period than the second one, is older and so has a longer latency period and a better chance to show an excess risk of cancer. In both of the subcohorts, we observe a significant excess of lung cancer deaths. This excess may be linked to the radon daughters exposure and to other dusts present in the underground environment, but it can also be in relation with the tobacco consumption which has not yet been controlled in this study, or with exposure to radon daughters at home, which has to be discussed in this evaluation. Expressing an excess risk by unit of exposure without standardization on tobacco smoking implicates that the tobacco consumption and radon exposure are two independant factors. At the actual point of our study, the verification of the dosimetric data is rather complete and the collection of the tobacco consumption is in progress. The opportunity of studying this second factor in a cohort study or in a nested case-control study is under discussion.

CONCLUSION

At the actual point of the study, we confirm the excess lung cancer mortality of these French underground miners, which is significantly in excess in both cohorts, but the verification of the tobacco factor in these 2 subcohorts is necessary before realizing a dose-response analysis in relation to the radiation exposure or estimating a risk factor.

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EFFECTS OF PARTICLE SIZE DISTRIBUTION IN THE ANALYSIS OF BIOASSAY RESULTS FOR AIRBORNE EXPOSURE INCIDENTS

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ABSTRACT

A study has been conducted of the variations in retention and dose predicted by the ICRP-30 model for whole-body counts and urinalysis. Various isotopes and chemical forms are examined and compared. The variation in retention and dose due to particle size distribution is examined in detail. Calculations are compared against the United States Nuclear Regulatory Commission UNIBIO code.

ONCOGENES IN RADIATION-INDUCED LUNG TUMORS

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ABSTRACT

Oncogenes in Radiation-Induced Lung Tumors. M. E. Frazier, G. L. Stiegler, L. L. Scott, S. R. Peterson and R. J. Rausch. Pacific Northwest Laboratory P. O. Box 999, Richland, Washington 99352, telex 15-2874.

This research examines the role of known oncogenes in lung tumors that develop in animals following the inhalation of ^{238}Pu and ^{239}Pu. The standard NIH 3T3 transfection assay was used to detect dominant-acting transforming oncogenes present in plutonium-induced tumors. The genes responsible belong to the ras family. Cloned viral homologs of oncogenes labeled by nick-translation were used as probes to analyze DNA from tumor cells and from radiation exposed nontumorous cells from the same animals. Novel (tumor-specific) restriction fragment length polymorphisms were detected in the DNA sequences of ras genes from some plutonium-induced tumors. Levels of ras gene transcripts were also higher in the tumors than in normal lung tissue from the same animal. These studies indicate that a ras oncogene is activated in plutonium-induced lung tumors.

Evidence is accumulating that oncogenes activated by certain chemical carcinogens contain characteristic molecular changes in their DNA sequences. The molecular lesions in the activated ras oncogenes of plutonium-induced lung tumors are being characterized in order to determine whether alpha-radiation induces distinct changes in the DNA. If so, the changes may serve as molecular markers for identifying radiation-induced DNA damage in cancer-causing genes. Such markers could provide a means of examining the relationships between risk and causality.

Work supported by the U.S. Department of Energy under Contract #DE-AC06-76RLO 1830.

PLUTONIUM PARTICLE AGGREGATION AND PROMOTION OF PULMONARY CARCINOGENESIS FROM INHALED PLUTONIUM

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Radiation doses delivered to focal areas of lung from aggregations of inhaled $^{239}PuO_2$ particles elicits a sequence of focal changes, including inflammation, fibrosis, bronchiolar hyperplasia and adenomatous bronchiolization that precedes the promotion of malignant lung tumors[1-4,7]. Nonciliated Clara cells in terminal bronchioles may be target cells for most induced carcinomas in the rat lung from inhaled plutonium[1]. These cells differentiate and proliferate during normal, hyperplastic and metaplastic renewal[6]. Promotion represents a clonal amplification of initiated cells, facilitated by increased cell division following tissue damage and repair stimuli, such that they express the neoplastic phenotype[10].

METHODS

Young adult, SPF, Wistar female rats were either sham-exposed (361 rats) or exposed to an aerosol of high-fired ^{169}Yb-$^{239}PuO_2$ with an AMAD, of 1.6 ± 0.11 μm (303 rats). Initial lung burden (ILB) was determined in individual rats by whole-body counting for ^{169}Yb at 14 days post-exposure. Radiation dose to the whole lung was calculated based on ILB determinations and time to death after inhalation using a two-exponent lung clearance function (About 80% ^{239}Pu cleared with a half of 20 days and 20% cleared with a half-life of 200 days) and a lung weight of 1.6 g[5]. Determination of plutonium particle distribution by quantitative autoradiography in the left lobe (using Ilford K-5 emulsion and a four week exposure) was carried out in a 5 μm sections of exposed rats with lung doses from 0.35 Gy to 20 Gy. Lung sections were examined with a light microscope using a 6.3 X objective and a 12.5 X eyepiece containing an ocular micrometer divided into 100 squares using a 10 x 10 matrix, with the side of each square measuring 150 μm. Each star (point source of $\geq$4 α-tracks) was considered to represent one particle. A particle aggregate consisted of more than one star present in one or more adjoining squares of the grid; particles in all adjoining squares were considered part of the same aggregate. The area encompassing each aggregate with $\geq$25 particles was also measured with the ocular micrometer.

RESULTS

Median lifespan ranged from 620-690 days post-exposure for lung doses of 0-10 Gy, and 460-560 days at doses of 12-20 Gy. Plutonium particle aggregation was common in alveolar regions of the lung (Figure). No aggregates with $\geq$25 particles were found at lung doses of about $\leq$1 Gy. Aggregates with $\geq$25 particles were nearly always associated and surrounded by inflammatory and fibrotic tissues. At doses $\geq$1 Gy, the number of aggregates with $\geq$25 particles increased in a linear manner from 0.2% at a dose of 1.4 Gy to 8.2% at 20 Gy; at 8 Gy about 4% of aggregates had $\geq$25 particles (Table). A total of 307 aggregates with $\geq$25 particles were counted in all dose groups (range of 26-515 and a mean ± S.E. of 58 ± 18 particles/aggregate). The mean ± S.E. of tissue volume containing aggregates of $\geq$25 particles was 7.5 + 0.2 x 10^6 μ^3, with a range of 1.4 x 10^6 μ^3 to 2500 x 10^6 μ^3. A mean dose of 1.2 cGy per day was delivered to the mean aggregate tissue volume, containing $\geq$25 particles; each particle was

assumed to be 0.12 μm in diameter. Daily doses ranged from 0.002 cGy/day to 53 cGy/day, depending upon particle number and tissue volume.

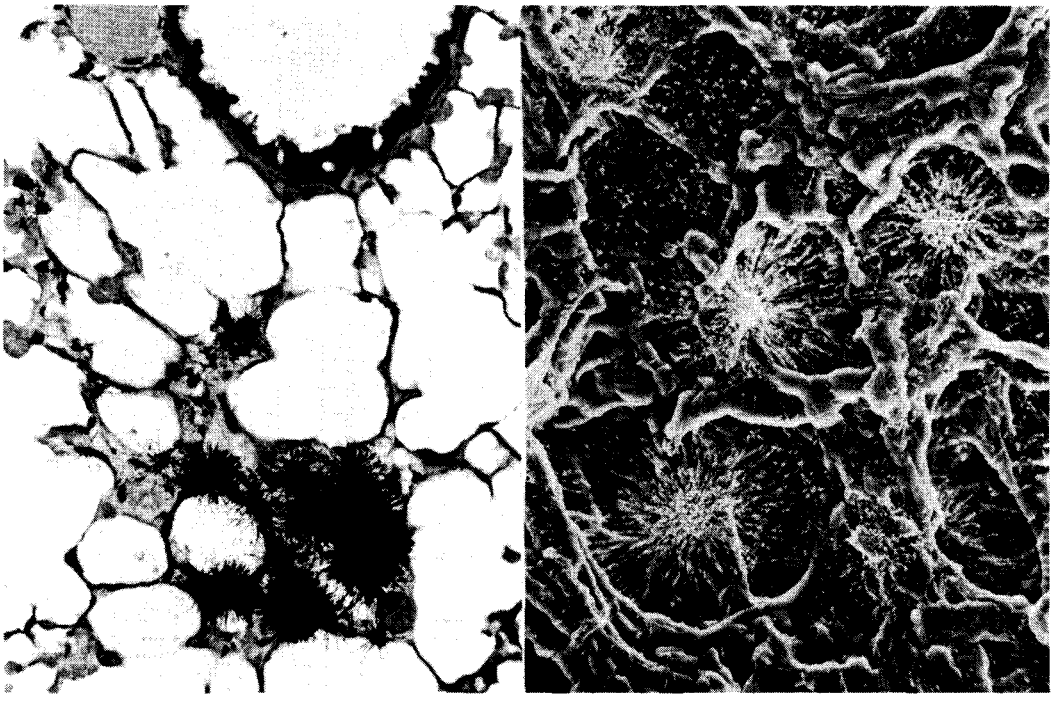

Figure. Aggregation of inhaled $^{239}PuO_2$ particles in rat lung: left, light microscopic autoradiogram; and right, scanning electron microscopic autoradiogram.

The incidence of all lung tumors (about 80% of which were carcinomas) increased rapidly, starting at a lung dose of 1.4 Gy, increasing to a maximum tumor incidence of 83% at 8 Gy (Table). No further increase in lung tumor incidence was seen at lung doses $\geq$8 Gy. The maximum lung tumor incidence at 8 Gy was associated with a plutonium particle concentration of 130 particles/cm^2, an aggregation concentration of 25 aggregates/cm^2, a mean of 4.0 particles/aggregates, and 4.0% of aggregates having $\geq$25 particles.

Table. Relationship of initial lung burden (ILB) to particle aggregation and pulmonary pathology following inhalation of $^{239}PuO_2$; values are means ± standard error.

Number of Rats	IAD kBq	Dose to Lung, Gray	% Aggregates: ≥25 Particles	% Lung Tumors
361	0	0	0	0.6
39	.10 ± .003	.35 ± .01	0	5.1
40	.20 ± .003	.73 ± .02	0	0
35	.38 ± .01	1.4 ± .06	0.2 ± 0.1	5.7
35	.71 ± .02	2.6 ± .4	0.9 ± 0.3	11.4
28	.94 ± .03	3.4 ± .05	1.1 ± 0.4	21.4
17	1.2 ± .02	4.5 ± .06	1.8 ± 0.7	35.3
17	1.7 ± .07	5.9 ± .15	1.6 ± 0.7	58.8
18	2.2 ± .04	7.9 ± .11	4.0 ± 1.4	83.3
13	3.0 ± .29	10 ± .15	3.5 ± 1.1	76.9
20	3.5 ± .18	12 ± .12	4.9 ± 1.6	80.0
22	4.4 ± .17	15 ± .26	5.3 ± 1.1	77.3
19	5.9 ± .17	20 ± .38	8.2 ± 2.1	78.9

DISCUSSION

Inhaled $^{239}PuO_2$ particles are initially deposited in alveoli in a random, uniform fashion, mostly as single particles. Each individual submicron $^{239}PuO_2$ particle is incapable by itself of eliciting an inflammatory or fibrotic response. However, inhaled plutonium particle aggregates formed during the first year after inhalation, particularly in subpleural and peribronchiolar regions of the lung, are capable of inducing these lesions[2,7]. Plutonium-238 microspheres introduced into the lung of rats by intravenous injection have been largely ineffective in inducing lung tumors[8]. However, lung tumors are induced by aggregations of $^{239}PuO_2$ formed in lung tissue following transthoracic injection[9].

About 80% of the total lung dose is delivered during the first 90 days after plutonium inhalation, amounting to a mean daily dose to the whole lung during this period of 7 cGy in rats receiving lifespan lung doses of 8 Gy. Yet, the mean dose-rate to focal aggregate regions at the time of death was only 1.2 cGy per day. Chronic α-radiation damage and cell death in bronchioles adjacent to large plutonium particle aggregates results in a wave of proliferative repair[1]. Aggregates ≥25 particles and lung tumors were infrequent at doses ≤1 Gy. At doses of ≥1 Gy there was a positive association between particle aggregation, of sufficient size to initiate focal inflammatory and fibrotic lesions, and lung tumor induction. The steeply rising portion of the dose-lung tumor curve for inhaled $^{239}PuO_2$ (from 1.4 to 8 Gy) appears to be due to the formation of large plutonium particle aggregates adjacent to bronchioles, that promote the appearance of lung tumors.

Work supported by the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

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COMPARAISON DE LA DISSOLUTION PULMONAIRE DU TETRAFLUORURE D'URANIUM AUX TESTS DE DISSOLUTION IN VITRO

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INTRODUCTION

Le tétrafluorure d'uranium est un intermédiaire important lors de la conversion du concentré en métal ou en hexafluorure. Dans le but d'éviter toute surestimation ou sous-estimation de la charge pulmonaire, en cas d'inhalation accidentelle d'UF_4, la surveillance médicale du personnel, nécessite de connaitre parfaitement son devenir biologique (Métivier et al 1987).

Cependant dans le cas précis d'UF_4 les travaux expérimentaux tentant de déterminer la classe de solubilité de ce composé ont montré d'importantes contradictions. Yuile (1973), Durbin et Wrenn (1976), et Ansoborlo (1983) indiquent qu'UF_4 se comporte comme un composé insoluble, Galibin et Parfenov (1971), Cooke et Holt (1974) et Stradling et al (1985), indiquent qu'UF4 se comporte comme un composé modérement soluble. Toutefois la CIPR (1979) a classé ce composé dans le groupe W des composés moyennement transférables.

Le but de notre étude était de déterminer la classe de solubilité d'un échantillon industriel d'UF_4 après exposition pulmonaire chez le rat et le singe. Nous avons étudié l'élimination urinaire et la clairance pulmonaire de l'uranium chez le rat et le singe, après exposition à de faibles quantités d'UF_4 par inhalation ou instillation intratrachéale. Les résultats ont été comparés avec ceux obtenus par un test de dissolution chimique (Kanapilly et al 1973).

MATERIELS ET METHODES

Deux souches de rats ont été utilisées dans cette étude. Huit rats Wistar et 24 rats Sprague-Dawley pesant 220 g au début de l'expérience ont inhalé par voie sèche de la poudre d'UF_4 en provenance de l'usine de Malvesi (France). Huit rats Sprague-Dawley pesant 220 g au début de l'expérience ont reçus par voie intratrachéale 6 ug d'UF_4. Deux singes babouins (Papio papio) mâle et femelle pesant 7,5 et 8 kg ont reçu par voie intratrachéale 160 ug d'UF_4.

Immédiatement après l'inhalation les rats et les singes sont placés individuellement en cages à métabolisme, les urines recueillies individuellement et quotidiennement. Les rats ayant inhalé UF_4 sont sacrifiés séquentiellement, par groupe de 4; 24, 48, 72, 96, et 168 heures après l'inhalation. Les rats et les singes ayant reçu une instillation unique ont été maintenus en cages à métabolisme pour une durée de 20 jours. Au sacrifice, il a été prélevé le poumon, le foie, les reins et les fémurs.

Les tests de dissolution chimique développés par Kanapilly (1973) ont été utilisés précédemment par Cooke et Holt (1974), Ansoborlo (1983) et Stradling et al (1985). La dissolution dynamique d'UF_4 par un flux traversant (André et al 1987a) d'un milieu simulant le sérum sanguin (Gamble 1967), à 37 °C, oxygèné ou non, a été étudiée pour déterminer l'effet de l'oxydation sur la dissolution de ce composé.

Les concentrations d'uranium dans les urines, les liquides d'élution du test de dissolution chimique ont été dosées par une méthode fluorimétrique

directe. Les organes ou les filtres contenant de l'uranium sont calcinées à 450°C pendant 2 heures avant d'être dosés par fluorimétrie. Le niveau de détection de cette technique est de 5.10^{-3} ug/ml.

RESULTATS

Inhalation - La moyenne des charges pulmonaires initiales des rats (11,75 ug) a été déterminée à partir des dosages effectués après chaque inhalation sur 4 animaux. La rétention pulmonaire est faible. Son évolution au cours du temps peut être décrite par une exponentielle simple d'équation $y = 1,04\ e^{-0,10t}$ ($r = 0,98$) pour laquelle on peut calculer $T_{1/2} = 7,28$ j. Ces résultats indiquent que la dissolution d'UF_4 est rapide.

L'élimination urinaire montre peu de variations entre les 7 premiers jours qui suivent l'inhalation, le taux d'excretion est constant (tableau 1). La moyenne des taux d'excretion urinaire réalisée à partir de ces données et provenant de l'ensemble des rats donne une valeur de $8,18 \pm 4,12\ 10^{-2}$ par jour. La fixation d'uranium dans les reins représente 7% de la charge pulmonaire dès 48 heures après l'inhalation et demeure constante pendant 7 jours. Les charges hépatiques et osseuses se sont révélées inférieures à la limite de détection.

Instillation intratrachéale - Les charges pulmonaires initiales des deux rats sacrifiés après l'instillation étaient respectivement de 7,56 et 5,15 ug. L'instillation de 3,5 ml d'une suspension d'UF_4 à 45 ug/ml a donné une estimation de la charge pulmonaire des singes de 160 ug.

Le taux d'élimination urinaire moyen, mesurée sur 9 jours, chez le rat et le singe est respectivement de $4,8 \pm 0,7.10^{-2}$ et $2,9 \pm 0,3.10^{-2}$. Les cinétiques d'élimination urinaire étudiées chez les deux espèces pendant 21 jours peuvent être décrites par des exponentielles simples d'équations: $y = 7,14\ e^{-0,16t}$ ($r = 0,95$) pour le rat et $y = 1,92\ e^{-0,05t}$ ($r = 0,86$) pour le singe. L'élimination urinaire chez le rat est donc plus rapide que celle observée chez le singe.

Enfin les résultats après instillation intratrachéale sont identiques à ceux obtenus après inhalation (Tableau 2).

Test de dissolution chimique - Les résultats de dissolution chimique indiquent que l'oxydation d'UF_4 tient un rôle prépondérant dans sa dissolution (Tableau 3). En effet, pendant les soixante premières heures d'élution le taux de dissolution pour le "Gamble" additionné d'oxygène est 5 à 10 fois supérieur aux taux observé pour le "Gamble" seul et le "Gamble" additionné d'argon. Quel que soit le milieu d'élution, on observe une augmentation continue de la quantité d'uranium dissout durant les 60 premières heures puis une diminution très nette des quantités d'uranium solubilisées soit après la $60^{ème}$ heure d'élution pour l'argon soit après la $70^{ème}$ pour le "Gamble" seul. L'apparition de ces faibles quantités d'uranium dissoutes coincide avec le moment ou les liquides d'élution ont été remplacés alors que les liquides d'élution d'origine se sont probablement chargés en oxygène à partir de l'air ambiant. A cet instant, la dissolution de l'UF4 par le liquide de "Gamble" additionné d'oxygène est 50 fois supérieure à celle du liquide de "Gamble" non oxygéné et 20 fois supérieure à celle du liquide de "Gamble" additionné d'argon. Ces faibles valeurs de dissolution restent stables pendant quarante heures et ne réaugmentent qu'au-delà de la $100^{ème}$ pour le "Gamble" seul et pour l'argon confirmant ainsi l'oxygénation des milieux d'élution par l'air ambiant. Le taux d'élution moyen du "Gamble" oxygéné de la $10^{ème}$ à la $100^{ème}$ heure est de 9,34 ug/ml/h. Ce qui correspond compte tenu des 40 mg de poudre placée sur le filtre à un taux de dissolution par 24 heures de $3,4.10^{-2}$.

DISCUSSION

La rapidité de la clairance pulmonaire chez le rat est confirmée par le taux journalier d'élimination urinaire de $5,6\ 10^{-2}$. En 7 jours l'élimination

urinaire représente 40% de la charge pulmonaire initiale, compatible avec une valeur d'épuration pulmonaire de 50% pour cette période sachant par ailleurs que la charge rénale à 7j est de 7%.

Le test de dissolution chimique réalisé sans oxygène montre un caractère d'insolubilité d'UF_4 compatible avec les conclusions d'Ansoborlo (1985). A l'inverse, l'oxygénation du milieu d'élution entraine une augmentation d'un facteur 50 de la fraction dissoute. Ceci peut s'expliquer par l'oxydation d'UF_4 en UO_2F_2, composé très soluble. Lorsqu'on compare la dissolution en milieu non oxygéné dans les tests in vitro à celle qui à lieu dans un organe caractérisé par une grande surface continuellement oxygénée il n'est pas surprenant d'observer des résultats si contradictoires. Les résultats obtenus par les tests de dissolution en milieu oxygéné montrent bien que la dissolution de l'UF_4 passe au moins par cette étape de transformation. En effet, le taux de dissolution d'UF_4 par 24 heures en milieu "Gamble" oxygéné (3,4 %), est équivalent aux taux d'élimination urinaire des rats et des singes (4,8 et 2,9 % respectivement).

L'épuration pulmonaire, après inhalation chez le rat, ainsi que l'élimination urinaire chez le rat et le singe, elle même similaire au taux de dissolution de la poudre industrielle d'UF_4, en milieu gamble oxygéné, montrent qu'UF_4 est un composé soluble à classer dans la catégorie D de la classification de la CIPR.

Les auteurs tiennent à remercier Mme I.Gil et Mr D.Renault pour l'aide efficace qu'ils ont apporté tout au long de cette étude.

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Tableau 1. Taux d'excrétion urinaire journalier en fonction de la charge pulmonaire initiale après inhalation de poudre sèche d'UF4 chez le rat.

Temps	Sprague-Dawley		Wistar	
J1	$6,5 \pm 0,9 .10^{-2}$	n = 4	$9,5 \pm 5,4\ 10^{-2}$	n = 6
J2	$6,7 \pm 3,6 .10^{-2}$	n = 7	$7,5 \pm 0,8\ 10^{-2}$	n = 4
J3	$10,3 \pm 2,9 .10^{-2}$	n = 10	$5,6 \pm 1,1\ 10^{-2}$	n = 6
J4	$10,0 \pm 1,5 .10^{-2}$	n = 5	$6,3 \pm 1,3\ 10^{-2}$	n = 4
J7	$7,4 \pm 1,4 .10^{-2}$	n = 3	$8,4 \pm 3,4\ 10^{-2}$	n = 3

Tableau 2. Comparaison des taux d'élimination urinaire journalier entre le rat et le singe après inhalation ou instillation intratrachéale (moyenne sur 10 jours exprimée en pourcentage de la charge pulmonaire initiale).

Inhalation		Instillation	
Rat SD	Rat Wistar	Rat SD	Babouin
$5,5 \pm 2,2$	$7,5 \pm 4,8$	$4,8 \pm 0,7$	$2,9 \pm 0,3$

Tableau 3. Résultats du test de dissolution chimique avec le liquide de "Gamble" (Serum simulant) chauffé à 37°C en présence d'oxygène ou d'argon.

Temps (H)	"Gamble" seul (ug/ml)	"Gamble" + O_2 (ug/ml)	"Gamble" + Argon (ug/ml)
2	0,4	1,3	0,7
5	0,3	3,7	1,8
10	0,4	4,6	1,7
20	1,0	5,3	2,0
30	1,8	6,9	1,7
40	3,0	9,5	1,4
50	4,5	9,3	1,6
60	4,9	10,2	0,5
70	0,2	10,5	0,6
80	0,2	11,6	0,6
90	0,2	12,4	0,6
100	0,4	13,1	0,6

ABSORPTION GASTROINTESTINALE DES ACTINIDES PAR LE SINGE NOUVEAU NE.

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INTRODUCTION

La valeur du transfert gastrointestinal des actinides, f_1, recommandée par la CIPR pour les enfants jusqu'à l'âge de 6 mois est 10 fois plus elevée que la valeur adulte (CIPR 48, 1986). Cette recommendation s'appuie sur des études réalisées avec différents mammifères qui ont montré que l'absorption gastrointestinale des actinides par le nouveau né était environ 100 fois plus élevée que chez l'adulte pour la toute première période de la vie. De plus, contrairement à l'adulte, on observe chez le nouveau-né une rétention importante des actinides dans le tractus gastrointestinal. (Sullivan 1980, Sullivan et Ghoram 1982, Sullivan et al 1985, David et Harrison 1984, Bhattacharyya et al. 1986, Bomford et Harrison 1986, Fritsch et al 1987b).

Cette augmentation de l'absorption gastrointestinale des actinides chez le nouveau né semble être associée à la perméabilité accrue de l'intestin aux macromolécules du lait maternel essentielles à l'acquisition de l'immunité. Mais bien que l'homme acquiert son immunité avant la naissance une augmentation de l'absorption gastrointestinale est probable car même chez les espèces comme le cobaye, ou l'acquisition de l'immunité n'est pas postnatale on observe également une augmentation de l'absorption gastrointestinale du Pu et de Am. Ceci pourrait s'expliquer par le passage de molécules plus petites que les gamma-globulines telle la lactoferrine avec laquelle le plutonium formerait des complexes (Bomford et Harrison 1986).

Bien que toujours largement supérieure à l'adulte, l'absorption gastrointestinale varie considérablement d'une espèce à l'autre chez les animaux agés de 1 ou 2 jours: chiens, 6% (Sullivan 1980), porcelet 13% (Sullivan 1980), rat 1-3% (Sullivan 1980, Bhattacharyya et al 1986), Hamsters 3-4% (David et Harrison 1984), cobaye, 2-3% (Sullivan 1980, Bomford and Harrison 1986).

Chez les rongeurs, hamster (David et Harrison 1984), rat (Bhattacharyya et al. 1986, Fritsch et al 1987a), cobaye (Bomford et Harrison 1986), l'absorption du plutonium décroît progressivement pendant la période de lactation pour atteindre la valeur de l'adulte vers l'âge du sevrage. C'est à partir de cette observation qu'est recommandée la valeur de f_1 pour les enfants.

Devant toutes ces incertitudes, le problème du modèle animal permettant la meilleure extrapolation à l'homme reste posé, c'est pourquoi nous avons étudié l'absorption gastrointestinale du neptunium et du plutonium chez une espèce animale plus proche de l'homme; le singe babouin, nous rapportons ici nos premiers résultats relatifs à la variation de l'absorption gastrointestinale en fonction de l'âge des nouveaux-nés.

MATERIELS ET METHODES

Animaux: Les babouins nouveaux-nés (Papio papio) mâles ou femelles sont nés dans notre animalerie. Ils sont maintenus en cage individuelle avec leur mère qui reçoit une alimentation constituée de fruits et de granulés du

commerce; l'eau de boisson est donnée à volonté; leur âge au gavage est rapporté dans le tableau ci-dessous.

Préparation et administration des radionucléides. Le Neptunium (V) 239 a été séparé de Am-243 par chromatographie comme nous l'avons décrit précédemment (Métivier et al, 1987). Juste avant l'administration, la solution de Neptunium dans HNO_3 1N est diluée au dixième; la masse de Neptunium ingérée a été d'environ 0,002 ug/Kg de poids corporel.

Le plutonium 238 utilisé était en milieu citrate 0.1M, la masse de plutonium ingérée était d'environ 2ug/Kg et l'activité d'environ 35 uCi/Kg. Toutes les solutions ont été filtrées (filtre Millipore 25 nm) avant le gavage et aucune perte de radioactivité n'a été observée.

L'ingestion des solutions (1ml) a été effectuée sur les animaux non anesthésiés à l'aide d'une sonde gastrique (Minerve, Paris, France). Aprés l'ingestion les nouveaux nés sont maintenus avec leur mère en cage à métabolisme. Le problème de la contamination croisée urine, fécès et fécès mère nouveau-né nous a conduit à exprimer nos résultats en terme de rétention et non de facteur de transfert.

Détermination de la quantité de radionucléides retenue. Quatre jours après l'ingestion les animaux sont anesthésiés et sacrifiés. Le sang est prélevé par cathétérisation de la carotide. Le tractus gastrointestinal, le foie, les reins, les fémurs et les humérus sont prélevés et analysés séparément. Les poumons sont également prélevés comme témoin d'une ingestion correcte. Le Neptunium contenu dans les tissus et les excréta a été déterminé par détection des rayons gamma. Le plutonium 238 a été déterminé soit par mesure directe des raies X de son descendant, soit par la technique de Keough et Powers (1970). La rétention totale des actinides a été obtenue en additionnant la quantité d'actinides contenue dans le squelette, estimée à partir des valeurs trouvées pour les fémurs et les humérus multipliées par 5.9 (Métivier et al 1986), du foie, du sang (représentant 7.7% du poids corporel) et des reins.

RESULTATS ET DISCUSSION

L'observation du tableau ci-dessous semble indiquer une diminution progressive de la retention du Neptunium et du plutonium entre l'âge de 1 et 34 jours. Pour le neptuniun la retention passe de 40 à 3 fois la valeur de l'adulte pour les âges de 4 et 26 jours respectivement. (Valeur adulte: 0.042% $\pm$ 0.006 %, Métivier et al 1986). La rétention du Plutonium chez le nouveau-né de 1 jour est 15 fois supérieure à celle de l'adulte (0.015% $\pm$ 0.006 %, Lataillade et al, résultats non publiés), elle est de 11 fois la valeur de l'adulte à 17 jours et est réduite à 7 fois la valeur de l'adulte à l'âge de 34 jours (Tableau 1).

Le neptunium et le plutonium sont retenus dans la paroi intestinale, principalement dans l'ileum, tout comme le décrit Sullivan (1980) et Fritsch et al (1987b) chez le rat le porcelet et le cobaye apres ingestion de plutonium. Toutefois, chez le singe nouveau né, la rétention dans le tractus gastrointestinal reste inférieure à 1% de la dose ingérée alors qu'elle peut atteindre 34% de la dose de plutonium ingérée chez le rat de 1 jour (Sullivan 1980).

Tout comme chez les rongeurs, mais à un degré moindre, la rétention du Pu et du Np après ingestion gastrointestinale chez le singe nouveau-né est supérieure à celle de l'adulte; 15 à 40 fois chez le singe contre 70 à 100 fois chez le rat.

Les valeurs de la rétention du Np à l'age de 26 jours et de celle du Pu à 34 jours (3 et 7 fois la valeur de l'adulte respectivement) laissent supposer que cette rétention atteindra la valeur de l'adulte avant l'âge du sevrage comme c'est observé chez les rongeurs, puisque le sevrage intervient chez le babouin entre 4 et 5 mois.

Nos résultats obtenus chez le singe se démarquent des valeurs obtenues chez le rat et utilisées par Crawford-Brown(1983) en vue d'une extrapolation à l'homme. En admettant que le singe est plus proche de l'homme que les rongeurs de par sa période de sevrage, sa durée de gestation, sa durée de vie, l'extrapolation à partir du modèle de Crawford-Brown peut être remise en cause. Cependant des expériences supplémentaires sont indispensables pour confirmer ou infirmer cette hypothèse.

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Ce travail a été partiellement financé par un contrat de la Commission des Communautés Européennes BI6-0098F(CD).

Tableau: Effet de l'age sur la retention du Neptunium et du Plutonium chez le singe nouveau -né après ingestion de nitrate de Neptunium 239 et de citrate de plutonium238.

	Nitrate de Np						Citrate de Pu		
âge au gavage	4d	4d	6d	8d	14d	26d	24h	17d	34d
dose (uCi/kg)	433	250	160	250	380	620	41	33	31
Tissus	Pourcent de la dose ingérée								
Poumon	0.003	0.002	0.0006	0.0005	0.0005	0.0002	0.0005	0.00092	0.0004
Squelette	1.65	0.183	0.14	0.152	0.443	0.107	0.206	0.1444	0.100
Foie	0.05	0.006	0.0031	0.0037	0.010	0.0020	0.0075	0.021	0.0064
Reins	0.015	0.003	0.0015	0.0016	0.0018	0.0036	0.0023	0.00092	0.0005
Sang	0.004	0.013	0.0011	0.0008	0.0028	0.0006	0.0038	0.0036	nd
Rétention totale	1.71	0.207	0.146	0.159	0.458	0.113	0.2205	0.1699	0.1069
Tractus GI	Pourcent de la dose ingérée								
Estomac	0.010	0.005	0.0066	0.0068	0.0065	0.032	0.002	0.0032	0.036
Duodenum	0.011	0.012	0.011	0.041	0.067	0.036	0.0048	0.0052	0.0031
Jejunum	0.012	0.005	0.012	0.0096	0.042	0.007	0.0034	0.0020	0.0071
Ileum	0.083	0.069	0.092	0.011	0.047	0.019	0.3400	0.2820	0.096
Colon	0.023	0.003	0.013	0.0009	0.0009	0.004	0.0110	0.0036	0.0052
Rétention totale GI	0.139	0.094	0.1346	0.0693	0.1634	0.099	0.3612	0.2960	0.1474

nd:non détectable

PHAGOCYTE ACTIVITY OF MACROPHAGES IN LUNGS OF HUMAN AND RATS EXPOSED TO $^{239}PuO_2$

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ABSTRACT

This paper introduced that the macrophages of human lungs and rat lungs in vitro after exposed to $^{239}PuO_2$ collected by means of washing lung method to study the difference of phagecyte activity between human and rat macrophages. The $^{239}PuO_2$ with MMD 1.3um was introduced into rat's lung by intractracheal injection, observing the correlation for phagocyte action of rats in vivo and in vitro.

The results showed that the phagocyte index of rat lung macrophages in vivo and in vitro increased with the length of exposed time. The phagocyte index was 1.65 and 1.77 for exposed time of 72 hours ($P>0.05$), and the phagocyte percentage was 89% and 85% respectively. In invitro the phagocyte index and percentage were 1.49 and 1.72, 92% and 89% respectively ($P>0.05$). The number of both macrophages in human and in rats were over 85% in whole lung washing liquid and their diameters were 12.53 and 11.35um respectively.

By scanning, there were a lot of folds and microvilii on the surface of control human cells. But after 12 hours exposed, the folds and microvilli of treated cells reduced or disappeared.

Interested in the results suggested that the similar action was appeared in rat macrophages for invovo and invitro model. In invitro, the difference of the phagecyte activity of macrophages between human and rat was not significant. It seems possible to mean that experiment for invitro may represent that for invivo and the results of experiment in rat be able to reflect that in human.

DOES THE GI TRACT MODEL OF THE ICRP PROVIDE RELIABLE DOSE ESTIMATES ?

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INTRODUCTION

For many essential and non-essential elements are the details of the intestinal absorptive pathways still poorly understood. There is increasing evidence that the walls of the small intestine are a more selective tissue than previously thought. As an example, Figure 1 illustrates the phenomenon of a transient intestinal retention component for ingested iron in man. After oral administration, there is a continuing excretion of iron for several weeks, far longer than the gastrointestinal transit time (as evidenced by the whole body retention of ^{51}Cr, which was administered simultaneously as a non-absorbable marker), and also far beyond the lifespan of the enterocytes. This fraction of the ingested iron can not have been transferred to the blood before, since after intravenous administration there is only a marginal excretion during that period (Fig.1). Since this transient iron retention in the gut walls depends on the body iron status and on pathophysiological conditions (1,2), it appears to reflect a hitherto unknown physiological mechanism which regulates iron absorption. Similar fine structures in the absorptive pathways are likely to exist also for other essential and non-essential elements.

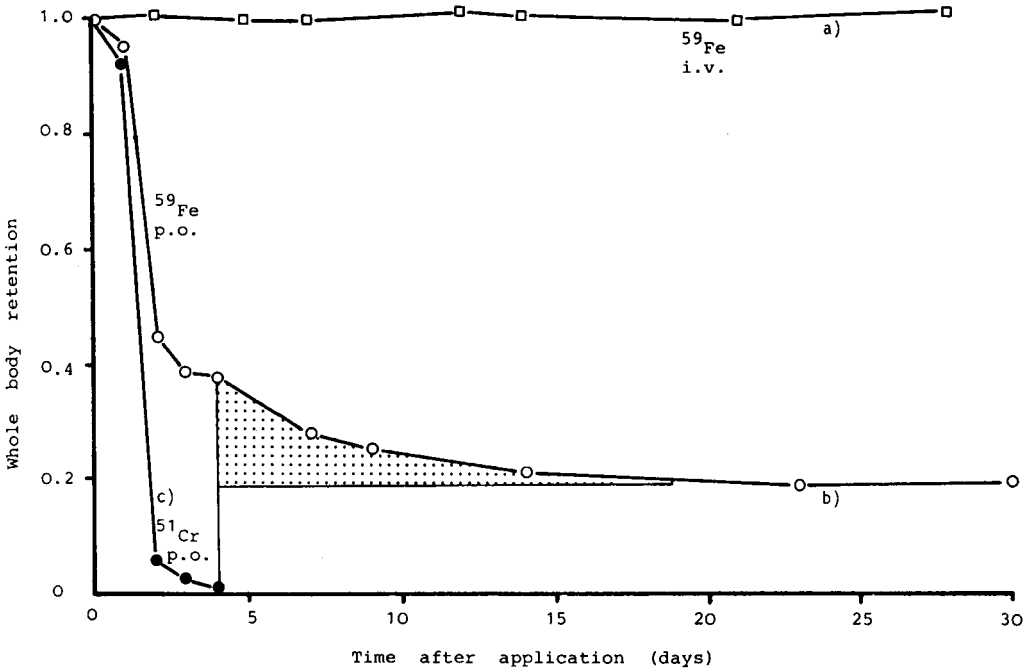

FIGURE 1. Whole body retention of intravenously administered ^{59}Fe (a), orally administered ^{59}Fe (b), and orally administered ^{51}Cr (c) in a healthy male subject.

These findings suggest that the current ICRP model for the gastrointestinal tract (3) might considerably underestimate radiation doses to the intestine, since this model does not account for any details in the absorptive processes. In the present study we therefore used the example of intestinal radioiron absorption in humans to evaluate the dosimetric consequences of a modified GI tract model.

RETENTION OF IRON IN THE GI TRACT

Absorption studies were performed in 23 healthy subjects with normal body iron status. The protocol of the absorption test is outlined in Table 1.

Figure 2 shows our suggested modification of the dosimetric model for the gastrointestinal tract. Instead of a direct transfer of activity from the small intestine to the systemic circulation, as in the ICRP model, the revised version takes account of retention of activity in the walls of the gut. According to this model, the whole body retention of the orally administered radioiron can be described by the following equation:

TABLE 1. Protocol of the radioiron absorption test.

23 healthy subjects with normal body iron status
Test dose: 100 ml deionized water 20 kBq ^{59}Fe-citrate 500 mg ascorbic acid 1mg / 5mg / 10mg / 20mg Fe^{2+} (ferrous sulphate) 0.4 MBq ^{51}Cr (Na_2CrO_4) (non-absorbable marker)
Whole body retention measurements of ^{51}Cr and ^{59}Fe for 3 - 15 weeks.

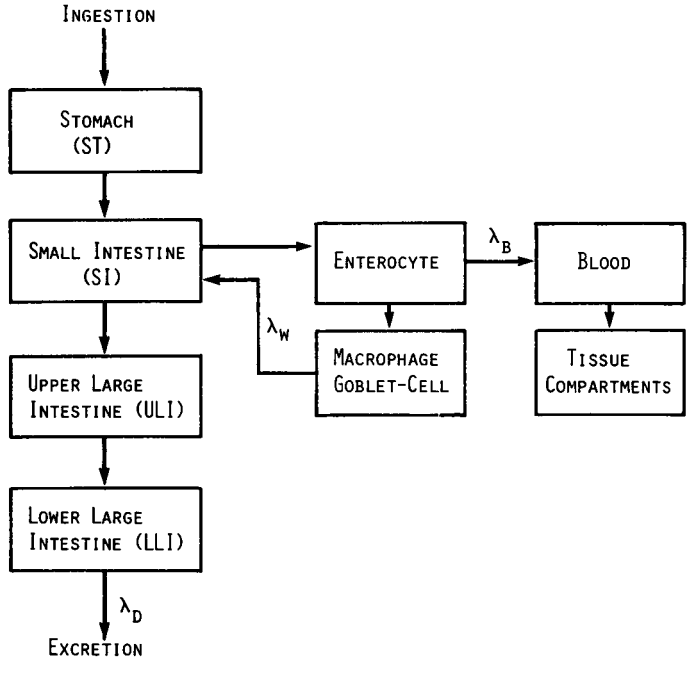

FIGURE 2. Modified dosimetric model for the gastrointestinal tract.

$$R_{WB} = f_1 + f_2 \cdot R_W(t) + (1-f_1-f_2) \cdot R_L(t) + f_2 \cdot \int_o^t R_W(t') \cdot R_L(t-t') \cdot dt'$$

with f_1 : fraction of administered activity transferred to the blood and tissue compartments,

f_2 : fraction of administered activity taken up by the gut walls but not transferred to the circulation,

$R_W(t)$: retention function of f_2; $R_W(t) = f_2 \cdot \exp(-\lambda_W \cdot t)$,

$R_L(t)$: retention of non-absorbed activity (gastro-intestinal passage).

Table 2 shows the values obtained for the absorbed fraction (f_1) and the submucosal retention component (f_2, λ_W). In normal subjects, about 1/3 of the radioiron taken up initially by the absorptive cells is finally transferred to the blood and tissue compartments, whereas 2/3 are temporarily retained in the gut walls and re-excreted into the lumen during the following weeks.

TABLE 2. Absorbed fraction (f_1) and temporarily retained activity component (f_2, λ_W) of orally administered radioiron in healthy subjects (mean values ± SD).

	ICRP - Model	Modified GI-Tract Model
f_1	0.1	0.20 ± 0.09
f_2	-	0.41 ± 0.24
$\lambda_W(d^{-1})$	-	0.24 ± 0.18

(n = 23)

DOSE CALCULATIONS

The calculations of absorbed doses were based on the MIRD concept (4) and were performed according to ICRP Publication 30, with the modifications of the GI tract model as described above.

The doses to the inner organs differ between the two models only with regard to the individual f_1-values (ICRP: f_1 = 0.1). According to the modified GI tract model, the dose to the gut wall has three sources: 1. from activity transferred to the blood and tissues; 2. from the activity retained in the gut walls (and re-excreted into the intestinal lumen); 3. from the activity in the lumen (non-absorbed and re-excreted). It was assumed that 2/3 of the absorption occurs in the duodenum and 1/3 in the jejunum. Iron, temporarily retained in the gut walls was thought to be uniformly distributed in half of the organ masses (duodenum: 30 g; jejunum: 140 g (males), and 125 g (females)). This was based on the assumption that the iron is retained in the gut wall but not in the muscles of the gut.

Table 3 compares the doses to the gut from orally administered ^{55}Fe and ^{59}Fe as calculated according to the two GI tract models. The doses to the gut walls from activity in the lumen and in the blood and tissues ($D_{B+L \rightarrow W}$) are very similar for both models. The additional dose component to the gut wall ($D_{W \rightarrow W}$), which is not considered in the ICRP model, however,

changes the total doses significantly. For ^{59}Fe as well as for ^{55}Fe, the doses to the duodenum increase by a factor of about 20, and to the jejunum by a factor of 3, as compared to the ICRP values.

TABLE 3. Radiation doses to the gut after oral administration of ^{59}Fe and ^{55}Fe. Comparison of ICRP data and values calculated according to the modified GI tract model.

Radio-nuclide	Dosimetric model	$D_{B+L \rightarrow W}$ (nSv/Bq)			$D_{W \rightarrow W}$ (nSv/Bq)	
		SI	ULI	LLI	Duodenum	Jejunum
^{59}Fe	ICRP 30	2.1	3.9	8.4	—	—
	Modified ICRP model	2.6 (+0.6)	4.3 (±0.9)	8.4 (±2.0)	38.5 (±24.4)	4.2 (2.6)
^{55}Fe	ICRP 30	0.12	0.17	0.30	—	—
	Modified ICRP model	0.21 (±0.09)	0.26 (±0.08)	0.37 (±0.08)	1.82 (±1.10)	0.19 (±0.12)

$D_{B+L \rightarrow W}$: Radiation dose to the gut wall from activity in blood and lumen,

$D_{W \rightarrow W}$: Radiation dose to the gut wall from activity retained in the gut wall (and re-excreted later).

CONCLUSIONS

Although radioiron is of limited interest in the field of radiation protection, it serves as a good illustration that the absorptive pathways of radionuclides deserve further investigations, especially in humans. Furthermore, any particular element should be considered in its own peculiarities and dosimetric models used should be based on a more physiological foundation, whenever such information is available.

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GASTROINTESTINAL ABSORPTION OF SOLUBLE URANIUM FROM DRINKING WATER

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This manuscript describes results of an experiment at our laboratory to determine the gastrointestinal absorption (GI) of U from drinking water in twelve healthy adults. Preliminary results obtained in four and eight subjects have already been reported (5,6). The degree of GI absorption of environmental U by humans is not well known. In 1983, Wrenn et al. (1) reviewed the published data on GI absorption and distribution of U in the body. Estimates for the fraction of U absorbed by humans varied tenfold, from 0.8% to 8%, with a consensus "best estimate" of 1.4%. The International Commission on Radiological Protection (ICRP) has recommended the use of 5% for the GI absorption of soluble U by occupationally-exposed workers (2). The GI absorption of U in man may be affected by other factors also. Sullivan et al. (3) and Willis (6) found that the absorption of U is three to seven times higher when given to rats on a fasting stomach than with food.

MATERIALS AND METHODS

Consumption of Water Containing Uranium and Sample Collection: Twelve volunteers, normal in kidney function and free of any indication of complicating pathology, were selected after careful detailed medical, hematological, biochemical and urological examinations at the University of Utah Medical Center. Subjects collected 24 hour total output of both urine and feces each day for seven days so we could establish their normal background of intake and excretion of U. Then during the next day (between 9:30 a.m. and 3:30 p.m.) they drank, at their normal rate of drinking water intake, 900 ml of water containing approximately 90 pCi ^{238}U and 90 pCi ^{234}U. Urine and feces were collected for seven more days.

Determination of U in Urine and Feces: The concentrations of U isotopes in urine and feces were determined by a radiochemical procedure developed in this laboratory (7). Urine samples, spiked with ^{232}U tracer, were wet ashed with HNO_3 and H_2O_2 until all organic materials were decomposed. Uranium was coprecipitated with iron hydroxide using ammonium hydroxide. The precipitate was washed several times with ammoniacal water, dissolved in concentrated HCl and the acidity was adjusted to 10 M. Uranium was extracted into 20% tri-lauryl amine (TLA) solution in xylene twice, back-extracted with 0.1 M HCl and electrodeposited onto a platinum disc. Uranium content and the tracer yield were determined by counting the platinum disc in a solid state alpha-spectrometer. The fecal samples, spiked with ^{232}U tracer, were dry ashed in a muffle furnace at 450°C, raising the temperature slowly in increments of 50°C, followed by wet ashing with HNO_3 and H_2O_2. The coprecipitation, extractions, back-

extraction, electrodeposition and counting were performed as reported for the urine samples.

RESULTS AND DISCUSSION

Most of the U ingested is excreted in feces in the first two days following ingestion of the water. However, in one or two cases we found that fecal excretion was slower and three days or more were required before urinary excretion returned to normal. The daily urinary and fecal excretion of ^{234}U and ^{238}U were determined as well as the volume of urine and mass of feces excreted each day by all subjects. Figure 1 shows a typical pattern.

The gastrointestinal absorption of U was estimated as follows:

$$\text{U absorbed} = \frac{\text{Net urinary excretion of U}}{\text{Amount of U in drinking water}} \times \frac{1}{f}$$

where net urinary excretion is that above background for 3 days, and f is the average cumulative fraction excreted in the urine over 3 days, equal to 0.79 as reported by Spoor and Hursh for 6 human patients injected with uranyl nitrate (8). The amount of U in 0.9 litre of drinking water was calculated from the measured concentrations of ^{234}U and ^{238}U in the water.

The results obtained by this technique are given in Figure 2. The absorption was the same for ^{234}U and ^{238}U for each subject. There was wide variability in absorption among subjects ranging from -0.02% to 2.6%, with a mean of 0.6%. The distribution appears to be bimodal with 60% of the subjects absorbing less than 0.2%. Low absorption may be due to the concurrent ingestion of food, as demonstrated in rat (3,4).

The dietary intake of U can be inferred from the results

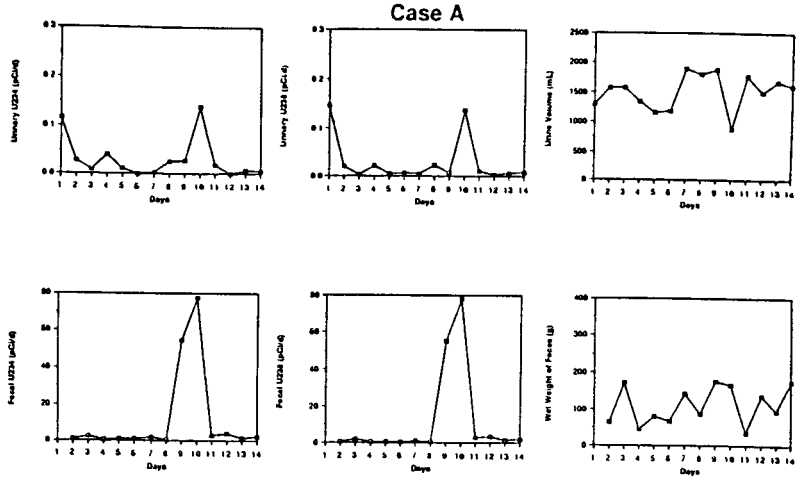
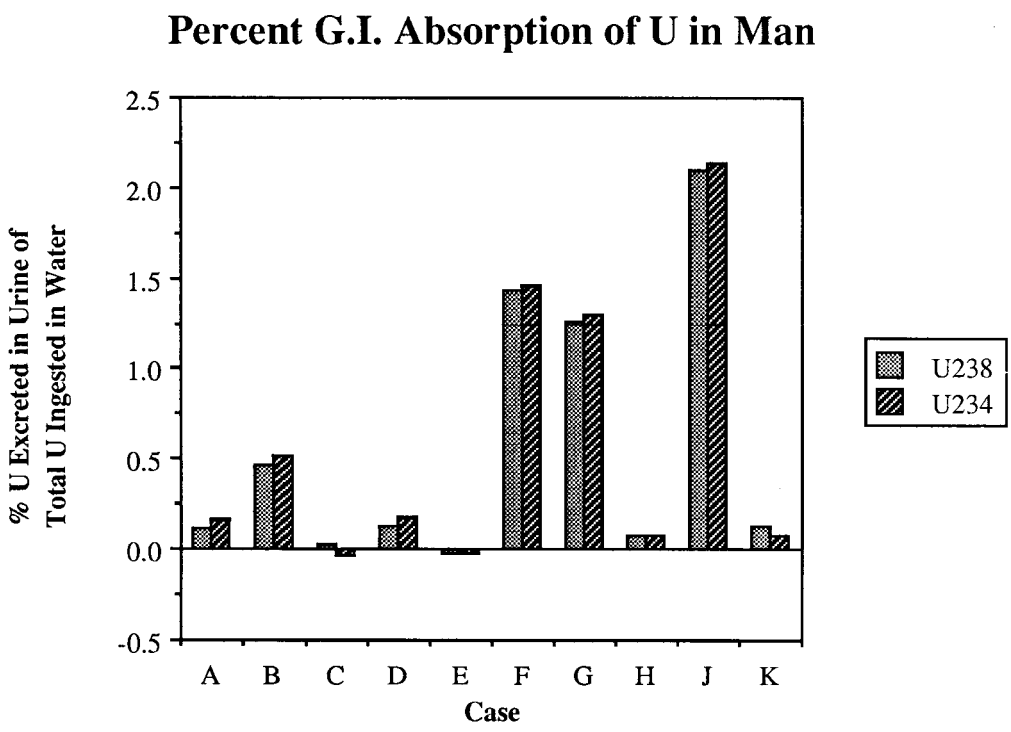


FIGURE 1. URINE VOLUME, FECAL MASS AND URINARY AND FECAL EXCRETION OF ^{238}U AND ^{234}U.



% U Excreted in Urine of Total U Ingested in Water

Case	^{238}U	^{234}U
A	0.11	0.16
B	0.47	0.52
C	0.03	-0.04
D	0.13	0.18
E	-0.02	-0.02
F	1.43	1.46
G	1.26	1.29
H	0.07	0.08
J	2.10	2.13
K	0.12	0.08
Average	0.57	0.58

FIGURE 2. G.I. ABSORPTION OF U

obtained in this study. The background urinary + fecal excretion of ^{238}U over seven days for all ten subjects averaged 5.45 ± 0.59 µg/day. At equilibrium the daily excretion of U is the same as total U intake per day. Since daily intake of U from inhalation is very low (a reference

intake of 0.0042 µg/day) (2), most of the daily intake should be from diet and drinking water. In our study, about 20% of ingested U came from drinking water, assuming 1 litre per day water intake (6). The U in water is high normal for the U.S. (9).

ACKNOWLEDGEMENTS

The work is supported by the U.S. Environmental Protection Agency through contract number CR312681-01-03 and the US Dept. of Energy, Contract DE-AC02-76EV-00119. The author thanks Jim Felt for technical assistance, and Nancy Peterson and Dawn Buster for editing and typing the manuscript.

This document has been reviewed in accordance with U.S. Environmental Protection Agancy policy through Cooperative Agreement CR-812681 to the University of Utah and approved for publication. Mention of trade names or commercial products does no constitute endorsement or recommendation for use.

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SOLUBILITY CLASSIFICATION OF YELLOWCAKE PRODUCED BY A BRAZILIAN URANIUM MILL

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ABSTRACT

Bioassay interpretation is very difficult in case of internal exposure to commercial yellowcake produced by uranium mills. This is because variations among the uranium compounds produced are believed to be process or site specific,and so are their dissolution characteristics in lung fluid.

An in vitro study was performed to determine the solubility of the yellowcake produced in a Brazilian uranium mill to permit classification of the inhaled material and to aid bioassay interpretation. The powder was taken from 6 different lots of yellowcake produced during a 6 month mill operation period, and the amount of uranium was determined by neutron activation analysis. Results can be expressed as:

$$F = f_1 e^{-\lambda_1 t} + f_2 e^{-\lambda_2 t}$$

and $$\lambda = \frac{0,693}{T1/2}$$

where F is the fraction of the uranium undissolved; f_1 and f_2 the dissolution fractions of the more soluble and less soluble portions; and $T_{1/2}(1)$ and $T_{1/2}(2)$,their associated half-lives for dissolution. Values of f_1, $T_{1/2}(1)$, and f_2, $T_{1/2}(2)$ for lots 01 and 02 were equal to 84.2%,10.7 h and 15.8%,12.9 days respectivelly. For lots 03 to 06 results are 70.8%, 8.8 h and 29.2%, 30.9 days respectivelly. The solubility experiment indicates class D and class W compounds.

UK STANDARDS FOR EXPOSURE TO RADON DAUGHTERS IN DWELLINGS

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ABSTRACT

The National Radiological Protection Board has issued formal advice on the standards to be adopted in the UK for control of exposures to radon daughters in existing dwellings and for changes in building procedures for future dwellings[1].

The standards are based on those recommended by the International Commission on Radiological Protection but adapted to circumstances in the United Kingdom. The matters taken into account by the Board when formulating its advice, and which are discussed in the paper, include the conversion from measured concentrations of radon to effective dose equivalent and the implied levels of risk, the comparison of these levels with risks from other causes, the numbers of dwellings in which various annual doses are likely to be exceeded, the geographical distribution of these dwellings, the likely costs and effectiveness of various remedial measures and the degree of domestic disruption.

The action level for existing dwellings selected by the Board was the same, 20 mSv per annum, as that recommended by the ICRP, but the Upper Bound for new dwellings of 5 mSv per annum was lower than that suggested by the ICRP.

(1) Exposure to radon daughters in dwellings. National Radiological Protection Board. ASP 10 (1987).

PUBLIC EXPOSURE TO RADON DAUGHTERS

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INTRODUCTION

Two surveys of indoor exposure to natural radiation in the UK have been carried out, a representative national survey and a selective survey of regions where above average radon concentrations were expected (1). Doses from both gamma rays and radon daughters were determined. Complementary studies were also made of the doses received outdoors. The focus here however is on indoor exposure to radon daughters.

NATIONAL SURVEY

This survey was carried out to determine the general distribution of radiation levels in dwellings, to search for correlations with the factors that might influence them and to improve the estimates of exposure of the population. The dwellings were selected systematically from the UK housing stock. The survey was conducted by post, measurements of radon concentrations in the living area and main bedroom being made with etched-track detectors (CR-39 detector elements) over a whole year. Information on the characteristics of the dwelling and relevant living habits of the occupants were obtained by use of a questionnaire. Of about 5,000 householders invited to take part, 54% agreed to participate. Of these, 88% (more than 2,000) completed the survey. This is a small fraction of the 20 million or so residences in the UK but was regarded as adequate if corrections could be made to any biases in the responses.

The distribution of radon concentrations for each dwelling weighted according to the average occupancy of different rooms (45% of time at home in the living area and 55% in the bedroom) is given in Figure 1. The distribution is approximately log-normal. The arithmetic mean, corrected for bias in the sample, was 20.5 Bq m^{-3} and the median about 13.6 Bq m^{-3}. The highest concentration found was 50 times the mean. The concentrations were strongly related to the local geology. Dwellings on clay had the lowest whereas those on more permeable sedimentary rock were higher. The highest concentrations were found in southwest England generally on or near granite.

The wide spread of results caused problems in analysing the data in relation to dwelling characteristics because the presence of one dwelling with high levels could unduly distort the analysis. To overcome this, the analysis was based on a restricted data set including only those dwellings with radon concentrations within two geometric standard deviations of the median.

About two-thirds of the dwellings surveyed were two-storey houses with the living area on the ground floor and bedroom upstairs. On average, the radon concentrations in such bedrooms were about 65% of those in living areas. About one in eight dwellings were bungalows: in these cases, the mean concentrations in bedrooms were only slightly less, at 90%, than the living areas. Flats and maisonettes above ground-floor level had lower

concentrations than ground-floor rooms. These results confirm the view that the ground is generally the main source of radon in dwellings. There was no correlation between the main building material of the exterior walls and the radon concentrations.

Measures taken to reduce heat losses in dwellings, such as secondary glazing and draught proofing, achieve their success, at least partly, by reducing the ventilation rate. The analysis of the data showed that dwellings with either double glazing or draught-proofing around doors and windows in general had higher radon concentrations than those without. This effect can be clearly seen with the data for 2-storey dwellings where, after subtraction of the contribution from outdoor radon (4 Bq m^{-3}), partial double glazing leads to an increase in radon levels of indoor origin (the ground and building materials) by about 30%; complete double glazing leads to an increase by about 60%. The majority (about 60%) of the dwellings in the survey were of the 2-storey type. For other types of dwellings, such as bungalows, for which smaller sample sizes were available for the analysis, the uncertainties on the data became larger. Nevertheless, a similar, albeit less pronounced trend, was also observed for bungalows.

In most cases, an analysis of the data with respect to window opening habits showed no clear trends. The only detectable trend was in the case of dwellings where the living room window was left open at night. Here the radon concentrations in the living rooms were on average 20% lower than those for the survey as a whole.

Several factors that might be expected to influence radon concentrations showed little or no effect. These included: the type of heating used; the presence of an air brick above floor level; the use of extractor fans; the presence of a chimney or flue and the type of floor (suspended wood or solid concrete).

REGIONAL SURVEY

This survey was carried out to determine the magnitude of individual exposures, the results from which indicated the need for standards in the UK. The regions were selected from a study of the literature on metal mining and uranium mineralisation and in consultation with geologists. No attempt was made to obtain a representative sample for each area but rather to obtain dwellings in locations where above average radon levels might be expected. Over 700 householders, the majority in southwest England, completed the survey.

The distribution of radon concentrations in the dwellings in southwest England, the most interesting of the regions studied, is shown in Figure 2. The values have been weighted as in the analysis of the national survey results by average occupancy of different rooms. The mean radon concentration is about 300 Bq m^{-3}, that is, about 15 times the national average. About 20% of the dwellings surveyed were above the NRPB action level of 400 Bq m^{-3} (2,3). The highest value was about 8,000 Bq m^{-3}. In the Pennine areas of Derbyshire and North Yorkshire, the mean concentration was 100 Bq m^{-3}, with 5% of dwellings exceeding the action level. In the selected areas of Scotland, the mean radon concentration was 70 Bq m^{-3} with only 2% exceeding the action

level. These last two sets of results are of some interest. In the former, it is thought that the substantial fracturing of the limestone which allows the relatively free movement of soil gas into dwellings is largely responsible for the higher than average levels. In the other case, although the regions surveyed were predominantly granite, the contrast to the results obtained in southwest England is marked. The absence of widespread mineralisation and fracturing of the granite is thought to explain the substantially lower levels of radon.

RADIATION DOSE

The national and regional surveys together provide a complementary indication of the exposure of the UK population to radon daughters in dwellings. The conversion coefficient from time-integrated radon concentration to effective dose equivalent now used is about 7 nSv per Bq h m^{-3} (3). This leads to a mean effective dose equivalent of about 1 mSv per annum from exposure to radon daughters in the home. An analysis of the time spent in indoor locations has revealed that, on average, 77% of time is spent indoors in residential accommodation and a further 15% in other buildings (1). On the reasonable assumption that radon concentrations in other buildings are the same as in dwellings, the average person in the UK would receive another 0.2 mSv per annum in these locations. Exposure out of doors (average radon concentration 4 Bq m^{-3}) would contribute a further 0.02 mSv per annum, giving a rounded total of 1.2 mSv per annum. The present evidence suggests that individual doses range from 0.4 to 400 mSv per annum in the UK.

Measurements of the concentrations of thoron decay products were made in 150 or so dwellings in the central uplands of England. A mean value of 0.3 Bq m^{-3} equilibrium equivalent thoron concentration was found. The effective dose equivalent due to indoor exposure was calculated to be 0.09 mSv per annum. Inclusion of a contribution from exposure outdoors leads to a total rounded dose of 0.1 mSv per annum.

These values may be compared with the mean annual effective dose equivalent from terrestrial gamma rays of 0.35 mSv with a range of individual doses from about 0.12 to 1.2 mSv, on the evidence from the surveys. The dose from cosmic rays is about 0.3 mSv per annum with only about a 10% variation throughout the country.

CONCLUSIONS

The surveys of exposure to natural radiation have provided firmer estimates than previously of the exposure of the UK population and have also indicated the range of exposures to radon daughters. These findings have led to an NRPB recommendation and Government agreement that action be taken to reduce high radon concentrations in existing dwellings and that measures be introduced to prevent high levels in new dwellings (2,3). The action level for existing dwellings is 20 mSv per annum (effective dose equivalent) which corresponds to a radon concentration of 400 Bq m^{-3}. The upper bound for new dwellings is 5 mSv per annum corresponding to 100 Bq m^{-3}. It is estimated that about 20,000 dwellings, mostly but not exclusively in

southwest England, exceed the action level and that 3,000 houses are being built each year which are likely to exceed the recommended upper bound, again mostly in the southwest.

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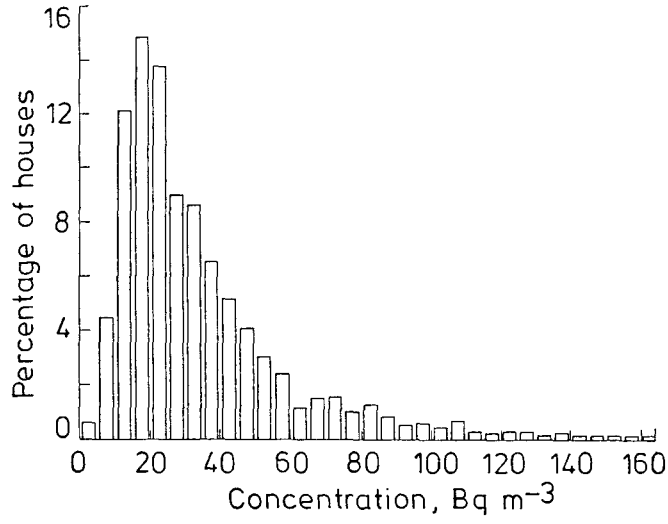

Figure 1: Distribution of radon concentrations in UK dwellings

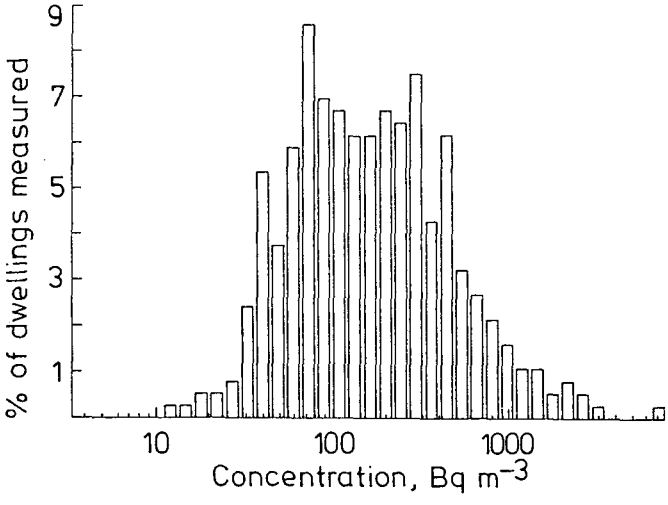

Figure 2: Distribution of radon concentrations in dwellings measured in the south west regional survey

INDOOR EXPOSURE IN THE COUNTRIES OF THE EUROPEAN COMMUNITIES

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I. INTRODUCTION

In the 1980 European Community Council Directive (1) laying down the basic radiation protection safety standards it is explicitly stated that the dose limits contained therein do not apply to exposure resulting from natural background radiation, defined as: "all ionizing radiation from natural terrestrial and cosmic sources, to the extent that the exposure which it causes is not significantly increased by man". By implication, therefore, the dose limits do apply to any exposure to sources of terrestrial and cosmic origin which is "significantly" higher as a result of human action but the word "significantly" is not defined.

In the case of radon in houses it may be reasonably argued that the dose limits should indeed apply to doses arising from soil radon which enters the air space of houses, in that the human activity of building a house irrevocably alters the radon concentration gradient from the soil, both spatially and temporally. Prior to the construction exposure to radon diffusing from a particular piece of land may be insignificant due to atmospheric dilution but, trapped in house air as a consequence of building, it may give rise to significantly enhanced exposure. In the case of radon from building materials a similar line of argument may be developed. It is therefore reasonable, to propose some form of dose limitation for radon in houses, whether from building materials or the soil.

In many E.C. Member States there is in fact increasing awareness of the significance of indoor exposure of the population to radon and its daughters. Several countries already have or are drawing up dose control policies. Action at E.C. level to establish such policies on a Community-wide basis is, therefore, urgently required to avoid a diversity of limits in the different Member States creating confusion leading to a negative psychological impact on the population as regards hasards from ionising radiation in general and giving rise to distortion of trade in construction materials between Member States. These reasons prompted the E.C. Commission to initiate a series of actions in both the regulatory and research fields.

II. ACTIONS IN THE REGULATORY FIELD

The Commission asked the Group of Experts, set up under the terms of Article 31 of the Euratom Treaty to advise on radiological protection matters, to review the problems of indoor exposure and in particular:

- to make an inventory of surveys of exposure of the public from natural radiation in dwellings that have been completed or are being carried out in E.C. Member States,
- to review the recommendations of international radiation protection organisations to limit natural radiation exposure,
- to carry out a survey of activities in different countries for limiting exposure in dwellings.

An ad hoc Working Party, established by the Article 31 Group of Experts with the above terms of reference, produced a report which was subsequently adopted by the Group and published in 1987 (2). Apart from considering all of the above, the report extends technological countermeasures dosimetry risk assessment and surveys conducted outside the Community.

E.C. SURVEY RESULTS

In recent years an impressive number of national and regional surveys of exposure from natural radiation have been carried out in the E.C. Member States with particular regard to indoor exposure and especially that from radon and its daughters. Some of these surveys were carried out under contract to the European Commission within the framework of the Radiation Protection Research Programme (3,4,5).

Temporal average values for indoor radon concentrations in the populated areas surveyed mostly ranged from about 20 to 50 Bq/m^3. However, the log normal distribution extends values to include a small percentage of houses with concentrations in excess of 400 Bq/m^3 and some individual dwellings with concentrations a further order of magnitude higher. In the case of high indoor concentrations (> 400 Bq/m^3) available evidence generally indicates the underlying soil as the main source.

Assuming an exposure to dose conversion coefficient of 20 Bq/m^3 per mSv/y, average indoor radon concentrations in the E.C. give rise to effective doses ranging from 1 to 2,5 mSv/year, but some members of the public receive doses in excess of 20 mSv/year.

CURRENT NATIONAL REGULATORY APPROACHES WITHIN THE E.C.

Many countries are considering a natural radiation control policy and have set up working groups to draw up proposals. In some the work is well advanced and recommendations on remedial action levels and limitations for future housing are being prepared. In the U.K. the National Radiological Protection Board (NRPB) issued recommendations in 1987 on exposure control of radon daughters in dwellings (6) and the Government has already adopted measures to deal with the problem in some parts of the country (7).

The results of the Article 31 experts' report (1) prompted the E.C. Commission to ask the experts to draw up a proposal for an E.C. policy on exposure to radon daughters in dwellings. Draft recommendations are in an advanced state of preparation and should be presented to the Commission shortly. They follow closely the principles given in I.C.R.P. Publication 39 (8) in stressing the importance of the controllability of the exposure; for this reason in recommending limiting values they distinguish between existing and future housing. They further stress the need to use the principles of optimisation in the application of either remedial or preventive measures

Finally, they emphasise the need to develop criteria for identifying regions, sites and construction characteristics which are likely to result in significantly high indoor radon levels.

III. ACTIONS IN THE RESEARCH FIELD

In the current (1985-1989) Radiation Protection Research Programme several items relate to natural radiation exposure (4,9)

- the physico-chemical properties of radon daughter aerosols; studies are being carried out to evaluate the electrical charge and size of radon daughter particulates and their dependence on environmental conditions such as humidity, trace gases and aerosol loading;

- the evaluation of risks due to inhalation of radon daughters; an overall model is under development, relating effective dose to radon gas concentration, taking into account ventilation rate, aerosol concentration and size distribution and plate out velocities;

- measurement of radon exhalation from building materials and soils, one study deals with the development of an instrument to measure "in situ" radon exhalation, another with the development of a model relating actual exhalation rates to three dimensional measurements;

- development of countermeasures; for this purpose test structures have been erected on a soil with artificially enhanced radium concentrations. The effects of various types of floors are being tested.

IV. CONCLUSIONS

Within the European Community, several Member States are working on the development of radon control policies. To avoid public confusion and distrust, the Commission, in collaboration with the Article 31 Group of Experts, defined the principles of a harmonised approach and formulated several recommendations. The most important of these is the acceptance of the guidance given in ICRP publication no 39 and the indication of the need for the development of effective remedial or preventative techniques based on the principles of optimisation. This will require research in the fields of dosimetry, risk assessment, epidemiology and control technology.

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NATIONWIDE DISTRIBUTION OF INDOOR RADON MEASUREMENTS

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Recently there has been increasing concern about the health hazard of lung cancer associated with indoor radon in the United States.[1-5] Because of this concern, Track Etch radon detectors are being used in indoor radon surveys in the United States and elsewhere.[6-8] The Track Etch method for measuring radon has been described previously.[9-12] Until quite recently only a small number of long-term indoor radon measurements had been made in the United States. The results of some of these earlier studies have been presented elsewhere.[8,13-20]

The recent widespread use of the detectors has led to the accumulation of a data base consisting of over 60,000 indoor radon measurements made as of mid-1987 throughout the United States in individual residences with long-term Track Etch detectors. This collection of indoor radon measurements and some of the conclusions that may be drawn from them are the subject of this paper.

The set of indoor radon measurements consists only of long-term measurements in homes not associated with man-made sources of radium or uranium contamination. Furthermore, only measurements in usual living areas are included. No short-term (less than a week) measurements are included and measurements in crawl spaces, sumps and drains have been eliminated. This data base is a compilation of measurements done by hundreds of clients and is not the result of a statistically designed survey of radon concentration in homes in the United States.

The number of measurements is not the same as the number of houses because sometimes more than one detector is used in the same house or at more than one time of year. The number of dwellings measured can be estimated approximately by dividing by two. Thus, the data base applies to about 30,000 homes.

At mid-1987 the data base file contained 62,000 records, each of which contain the following information: 1.) Mail zip code (when available), 2.) Average radon concentration, 3.) Serial number of the detector, 4.) Type of Track Etch detector, 5.) State or the region; 6.) Room where measured (when available), 7.) Dates of installation and removal, and 8.) Measurement period (days).

The data base, as a whole, is not representative of the whole United States. Large numbers of measurements have been conducted in known "hot-spots" in Pennsylvania, New Jersey and Maine. In some instances large numbers of measurements have been made in low population states. Large numbers of measurements have also not been made in several large states, for example Texas and California. Such measurements can wrongly dominate the overall statistics. Six states have over one thousand measurements but

many states have far fewer. Small magnitude trends in statistics can be observed after each updating of the data base and this compilation is the fifth major update.

Because the majority of the 50 states have less than one thousand measurements per state, we have also combined the data from certain groups of states into seven regional sets to obtain statistics more representative of the region. These data sets are the New England, Mid-Atlantic, Midwest, Northwest, Southeast, Mountain States, and Pacific States. State, regional and entire U.S. data bases are lognormally distributed. This is illustrated in Fig. 1, a histogram of the entire data base. Fig. 2 shows the distribution by arithmetic mean for the states and Fig. 3 likewise for the various regions. Data on the median, the geometric mean and standard deviation, the arithmetic mean and the percent of readings exceeding 4, 8 and 20 pCi/l have also been compiled for each state and region. (A value of 0.02 WL, or 4 pCi/l, has been given by the U.S. EPA as an action guideline.[22-23])

The seven regions have quite different statistics and can be related to regional geology. Both the Mid-Atlantic and Mountain State regions show elevated statistics and it is noteworthy that these regions contain granitic mountain ranges.[17]

Some 36 states have more than five percent of radon readings greater than 8 pCi/l which approximates an exposure of 2 working level months per year, the NCRP-recommended guideline.[1-2]

The radon concentration statistics for the states are quite different from one another and can also be correlated with general geology.

To make the data base statistics somewhat more representative of the United States, each state statistic can be weighted by population. For such a population weighted data base the geometric mean is 1.76 pCi/l, and the arithmetic mean is 3.61 pCi/l with 19 percent of the measurements exceeding the 4 pCi/l guideline.

There exists a growing data base of long term indoor radon measurements consisting of over 60,000 measurements. The primary conclusion that can be drawn at this time is that there are some homes with unacceptably high levels of radon in nearly every state. Furthermore, there exist 23 states and three regions where greater than 20% of the measurements exceed the 4 pCi/l guideline. A population-weighted treatment of the data base shows an arithmetic mean of 3.6 pCi/l, representing, in our opinion, the upper limit on the U.S. national average. Nevertheless, the data so far imply a national exposure in excess of the 0.8 - 1.5 pCi/l cited by the NCRP and other workers.[2,20] This would lead to a greater number of radon-related lung cancers than the 9000 cases per year also estimated by the NCRP.

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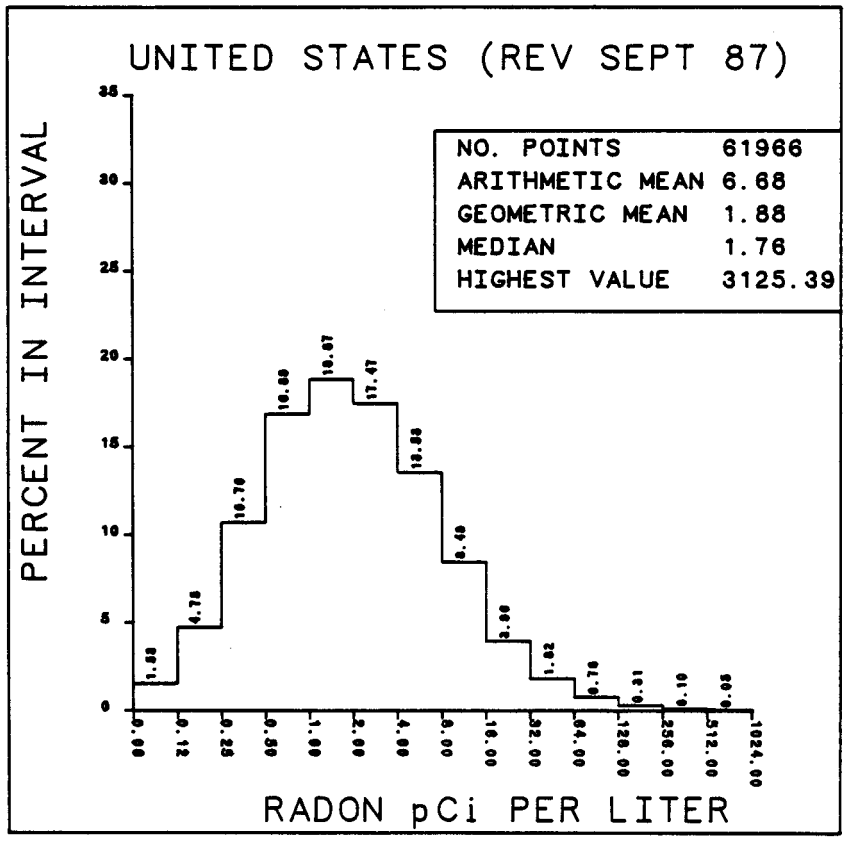

Fig. 1 Log-normal Distribution of Entire Data Base

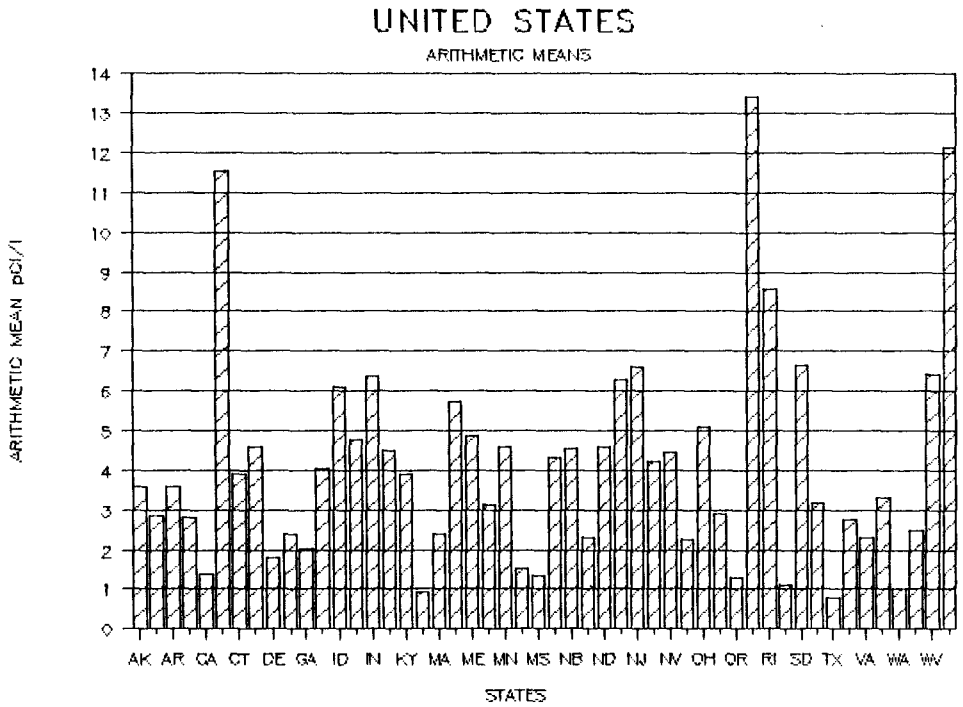

Fig. 2 Distribution of Arithmetic Means by State

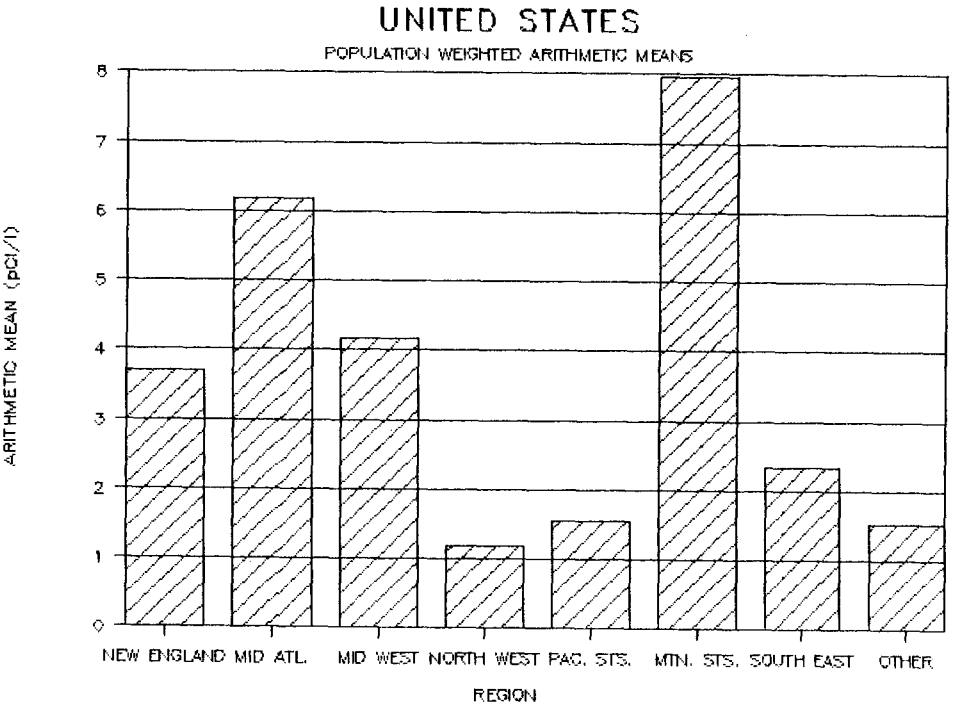

Fig. 3 Distribution of Arithmetic Means by Region

SOURCES OF "HIGHER" RADON LEVELS IN HOUSES, THE RESULTING RADIATION EXPOSURE AND ESTIMATE OF RISK

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ABSTRACT

The result of the "German study on radon" shows a median radon level of about 40 Bq/m³ in dwellings with a measured mean equilibrium factor of F = 0,3. The resulting median equilibrium equivalent concentration amounts to about 13 Bq/m³ corresponding to an annual exposure of 0,14 WLM for the population. The frequency distribution of the measured radon concentrations indicates that about more than five percent (~13000 dwellings) of the buildings investigated show concentrations exceeding 300 Bq/m³. This means that in the Federal Republic of Germany about 40000 - 50000 persons live in dwellings with radon concentrations exceeding this value. The maximum value of radon concentrations was found to be about 1250 Bq/m³. Special investigations in dwellings with comparable high radon levels showed that the exhalation from the soil is the decisive indoor radon source. It was established that the way of entry is mainly determined by specific construction parameters. The resulting mean annual radiation exposure by inhalation of short lived radon daughters for persons living in dwellings with radon concentrations of more than 300 Bq/m³ amounts, therefore, to more than 1 WLM. This means an enhancement of the radiation exposure of about a factor of ten compared to the corresponding mean exposure of the whole population. This results in an effective dose equivalent of about 7 mSv/a for this small group of the population. Applying a mean risk factor for lung cancer of 0,01 per WLM/a (modified relative risk model) a life time risk for this subgroup of the population of about 1 percent could be estimated.

Based on our investigations on diffusion and exhalation of radon and the results of the experiments in dwellings with high radon levels, ways and measures will be shown, to decrease the radon concentrations in existing houses and to prevent excessive exhalation from the soil into living rooms in "critical" regions.

RADON LEVELS IN AUSTRALIAN HOMES

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INTRODUCTION

There is currently little data published on the magnitude and geographical distribution of radon exposure for dwellings within Australia (1,2). To address this lack of data, the Australian Radiation Laboratory (ARL) is investigating radon levels within domestic dwellings in Australia. These measurements form part of an on-going programme by ARL to establish representative values for radiation exposure to the Australian public from natural sources, including radon. ARL has underway a large scale survey of Australian homes. A total of 12,000 persons, randomly selected from the electoral roll, have been invited to participate in a one year survey of radiation levels in their homes. The expected rate of positive reply is 30%, leading to an expected 3600 dwellings for study. Radon levels in these dwellings will be measured using a passive monitor containing nuclear track material, and the external gamma-ray radiation will be measured using a thermoluminescent dosimeter (TLD). This survey will be completed mid 1989. ARL has carried out a preliminary study of radon in domestic dwellings, using a charcoal-based, passive radon monitor developed at the Laboratory. These measurements were limited to homes in three cities and two towns, and were for an exposure period of 7 days. This paper reports the methodology and results of these initial surveys and examines the likely trends in average radon levels for houses in Australia.

RADON SURVEY

Charcoal-based passive radon monitors have been used by many workers to assess integrated radon exposure for periods of up to 10 days (3,4,5). These monitors collect radon by adsorption onto activated charcoal and gamma-ray counting determines the quantity adsorbed. A number of these designs incorporate some form of diffusion barrier to increase the integration time and provide a better measure of integrated radon for changing radon concentrations (3,4). The monitor developed at ARL is a low-cost variant of this type of monitor. Its main features and its operating characteristic are described elsewhere (6). The monitors were used to survey homes in three cities and two towns in a number of states

around Australia. The distribution method was chosen for convenience and was not designed to provide a random sampling of a particular location. The exposures were restricted to periods of 6 to 8 days and may be affected by seasonal variations. The derived radon concentrations should be indicative of general trends, but they may not be truly representative of the Australia-wide yearly average radon exposure.

Volunteers were given a questionnaire by a central contact in a government laboratory or university department in the city or area under study. This questionnaire covered details on the construction of the dwelling to be surveyed. Following the return of these forms, each participant was provided with a radon monitor. The monitor was exposed in the participant's home and the exposure details recorded on a label on the cup. The monitors were returned to the appropriate central contact, who then returned all the monitors to Melbourne for counting. This survey method ensured rapid distribution and collection of the monitors and minimised possible delays. Preparation and counting of the monitors was carried out at ARL, providing consistency in measurements from different locations.

Measurements were made in the cities of Melbourne (67 homes) and Sydney (45 homes) on the east coast, and Perth (45 homes) on the west coast of Australia. Two small towns in areas known to have higher than average uranium concentrations in the ground were also surveyed. Jabiru (43 homes) is situated close-by to the Ranger uranium mine and Armidale (55 homes) is situated in an area of granitic rock, containing uranium and thorium in higher than average levels. The mean and median radon concentrations in these homes for the periods sampled are summarised in Table 1. The highest radon concentration measured was 395 Bq/m^3. The combined radon data for all five locations (255 homes) approximates a log-normal distribution, with a derived geometric mean and geometric standard deviation of 20 Bq/m^3 and 2.6, respectively. The derived inter-quartile range was 10 to 46 Bq/m^3.

Table 1. Average radon concentrations for Australian homes

Location	No. of Homes	Month	Rn Concentration (Bq/m^3) Geometric mean	Inter-quartile range
Melbourne	47	May	53	39 - 70
Melbourne	20	July	19	6 - 24
Sydney	45	July	21	5 - 45
Perth	45	May	15	7 - 32
Jabiru	43	May	19	9 - 40
Armidale	55	August	10	4 - 31

SEASONAL RADON VARIATION

In general, Australian homes are well ventilated. For the 255 homes surveyed, 43% were built on a concrete slab and 51% had wooden floors with a ventilated area underneath. Less than 4% of homes had some form of basement. In the absence of a strong localised radon source, domestic radon levels could be expected to follow the open-air radon levels. Some support for this was found in measurements at Jabiru, where no significant difference was found between indoor and outdoor radon exposures. Short-term measurements of radon exposure in homes may suffer from sampling bias due to seasonal variations in ambient radon. The two sets of radon measurements in Melbourne during May and July gave mean radon concentations of 53 and 19 Bq/m^3, respectively. Further charcoal cup measurements of homes in Armidale (NSW) during the period July to November gave mean monthly radon concentrations of 20, 10, 5, 26 and 23 Bq/m^3, respectively, showing a minimum in the mean radon exposure during the months of August and September.

This seasonal variation is also present in the results of an extended measurement of ambient radon levels, using a continuously sampling electronic radon monitor, located at ARL in Melbourne. Table 2 summarises the average values, derived from the monitor, for ambient radon concentration, over a one year period at Yallambie, Victoria. The monthly average values were derived from the alpha count-rate on the second filter of a 200 litre two-filter tube, operated continuously over this period. This data shows an autumn maximum and a spring minimum, with a 30% range around the yearly average of 12.7 Bq/m^3.

Table 2. Open-air average monthly radon concentrations (Bq/m^3) for period November 86 to November 87, derived from two-filter tube measurements (at 3 meters) at ARL, Melbourne.

	Nov	Dec	Jan	Feb	Mar	Apr	May	June	July	Aug	Sep	Oct
Mean	11.3	11.4	14.3	16.3	16.5	18.8	16.1	10.1	12.6	7.9	9.7	8.0
1 SD	6.9	7.3	8.8	8.6	10.7	10.4	12.0	7.5	6.5	3.1	5.5	5.0
SEOM	0.3	0.3	0.3	0.4	0.3	0.4	0.4	0.3	0.2	0.1	0.3	0.2

DISCUSSION

Within the limitations of this preliminary survey, a number of general trends in the measured radon levels can be observed . The locations surveyed were geographically widely separated, in some cases by many thousands of kilometres.

Climate ranged from tropical (Jabiru) to temperate (Melbourne), with widely varying local geology. Despite these differences, the derived mean radon levels for each city and town were remarkably similar. Small scale surveys with short-term exposure periods may be biased due to the monitor distribution methodology and possible seasonal variations. The measurements of seasonal variations indicate that, relative to the yearly average radon concentrations, the measurements made in Melbourne during May may be seasonally enhanced, while the Armidale results for August may be seasonally reduced. Both locations produced a mean radon level of approximately 20 Bq/m^3 during July, corresponding to the derived mean value for the five cities and towns sampled.

ACKNOWLEDGEMENTS

The assistance of Bill Toussaint of the Western Australian State Health Department, Dr. Frank Cattell of Macquarie University, Dr. John Kleeman of the University of New England, Dr. Jiri Kvasnika of the Northern Territory Department of Mines and Energy and Alan Murray is gratefully acknowledge in the distribution of the monitors.

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ASSESSMENT AND CONTROL OF EXPOSURES TO NATURAL RADIATION IN DENMARK

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ABSTRACT

Assessment of the exposures to the Danish population from different natural radiation sources including building materials, drinking water, fly ash etc. has been performed from 1975 and up till now.

In 1987 a comprehensive nationwide investigation of the gamma exposures and radon levels in 500 randomly selected Danish dwellings will be concluded by the National Institute of Radiation Hygiene. At the same time the Danish authorities will publish a control strategy for limiting the exposure of the Danish population from natural sources, especially from radon daughter exposure in dwellings.

The presentation will outline the main results of the nationwide survey in Danish dwellings together with the main principles behind and the consequences of the initiated control strategy for limiting the exposures from natural radioactive sources.

RADON MEASUREMENTS IN INDOOR AIR

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INTRODUCTION

Exposures to radon and daughters due to natural background in indoor air have been estimated by a number of authors (Unscear, 1977 and 1982). Measurements in Australian buildings were reported by Wong (1983) and Kennedy et al (1984). Elevated activity concentrations were found in various types of buildings including terratecture and lithotecture (Baggs and Wong 1987; Hines, Wong and Taylor 1987). In this paper, an attempt to assess the effects of these activities is made.

EXPERIMENTAL

Calibrated nuclear track detectors were used for the survey of radon in indoor air. They were positioned at various exposure sites in selected buildings for periods up to one year. The detectors were processed and the average radon concentrations were determined as described elsewhere (Hines, Wong and Taylor 1987). Further calibrations were made with the radon chamber at the Australian Radiation Laboratory (ARL). In order to assess the effects of these radon activities, the working levels were calculated using the following equation (Straden 1979)

$$mWL = F \times \langle Rn \rangle / 3.7 \qquad ..(1)$$

where $\langle Rn \rangle$ = radon concentration in Bq m^{-3}, and
F = equilibrium factor for radon daughters.

The equilibrium factor for radon daughters may vary considerably with the atmospheric conditions. The mean value value obtained in the survey of private dwellings (Unscean 1977 nd 1982) was 0.5 and the deviation from the mean was found to be within 40%. In this paper the mean value was adopted for the calculation of the working levels.

RESULTS

The results of ten areas which have been surveyed (Hines, Wong and Taylor, 1987) were given in Table 1. These results of the working levels have been compared to those obtained by the air-sampling method. The system was calibrated in the radon chamber at ARL. Reasonably good agreements between these data suggested that the validity of the methods used in the present survey.

TABLE 1. WORKING LEVEL AND ANNUAL EFFECTIVE DOSE EQUIVALENT

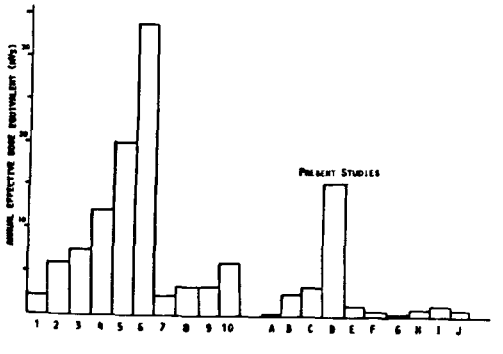
Area*	Mean Activity Concentration (Bq m^{-3})	Mean Working Level (mWL)	Mean Annual Effective Dose Equivalent (mSv)
Wooden Home	17	2	0.23
Sewerage Room	174	24	2.3
Area adjacent to Radon Laboratory	243	33	3.3
Radon Laboratory Radium Safe and Radon Plant Radon Laboratory	1230	166	16
Radon Laboratory - Measuring Area	122	17	1.6
Office 1	34	5	0.6
Office 2	21	3	0.23
Control Room for LINAC	64	9	0.92
Material Store	110	15	1.4
Geology Store	81	11	1.2

*See ref. (Hines, Wong and Taylor 1987) for details.

The lung dose due to the inhalation of airborne alpha emitting nuclides has been calculated by several authors (Jacobi, 1964; Harley, 1982; Hofmann, 1982). An important consideration in the dose-assessment is the mix of attached and unattached radon progeny. The unattached radon daughters, which amount to about 7% in indoor air, deposit all their energies in the respiratory tract, while the attached particles only deposit a fraction of their energies. The dosimetric calculations yielded an average value of 2 Sv J^{-1} for effective dose equivalent per unit of inhaled potential alpha energy intake of radon daughters (UNSCEAR, 1982). Using this figure, the annual effective dose equivalent for exposure in a building (ICRP, 32, 1981) was estimated to be 0.056 mSv per Bq m^{-3}. The values given in the third column of Table 3 were calculated using this conversion factor taking into account the fraction of the time used for occupational exposures.

DISCUSSION AND CONCLUSION

It would be difficult to assess hazards caused by airborne alpha-emitting nuclides, radon and thoron, because of the problems associated with accurate determination of some important parameters such as the equilibrium ratio and the unattached fraction during the prolonged periods of exposure. The results presented in Table 1 were calculated using the mean values of published data as explained previously. If we believe in these figures then the data given in Table 1 may serve as a guideline for the assessment of health hazards from these sources. There is a good reason to adopt these values as the estimates used in this paper are close to lower limits of published results, and consequently they would not overstate the effets. Consistency of track density measured in various detectors exposed at the same site and the comparison of the results to that obtained by the air sampling method supported these assumptions. The cancer risk for inhalation of radon and thoron daughters had been estimated by several international organizations (UNSCEAR, 1977; NAS, 1980; ICRP 31 1981; Cross, 1985) using uranium-mining data. The lifetime risk estimates for lung cancer attributable to radon progeny expsoure range from 1×10^{-4} per WLM to 14×10^{-4} per WLM, where WLM is the working level month. Using the lowest estimate, the highest lifetime risk in the area surveyed was found to be about 2×10^{-4} (Table 1), provided that the required condition of life time exposure is met. It should be pointed out that the estimation of the mean annual effective dose equivalent in areas such as the radon laboratory may not be realistic because the actual occupancy in this area could be lower than 170 hours for a working month used in the estimation. A comparison (Figure 1) of the mean annual effective dose equivalent with that obtained for other occupational exposures could be useful. It is interesting to note that the activity concentration in some of the areas surveyed could deliver a dose which is even higher than those obtained for the coal mines and private dwellings but is lower than those for the uranium mines. In view of the results obtained from this survey, it is apparent that the survey of radon in indoor air could give interesting and useful results for radiation protection. A passive dosimeter would be most suitable for this purpose.



1 Mean in private dwellings
2 Canadian uranium mines (1979)
3 Highest in private dwellings
4 French uranium mines (1979)
5 Indian uranium mines, underground (1979)
6 U.S. uranium mine (1977)
7 U.K. coal mines (1981)
8 Finland coal mines (1977)
9 Norway coal mines (1980)
10 Swedish coal mines (1980)

Fig. 1. Mean Annual Effective Dose Equivalent

Acknowledgement: The authors are grateful to Dr. J.R. Peggie of ARL for his help in calibration and to Mr. V. Leach of the Health Dept., Queensland for his assistance in the air-sampling experiments.

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ITALIAN SURVEY TO EVALUATE THE AVERAGE EFFECTIVE DOSE EQUIVALENT DUE TO RADON INDOORS

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1. Introduction

A more stringent control of the radioactivity either from artificial and natural sources has been requested in Italy by the public opinion after the Chernobyl accident. The competent Authorities and the radiation protection Community are making continuous efforts to fulfill such a request in the most suitable and appropriate ways. The person related environmental monitoring system - existing since the middle fifties - is being improved to face large scale contamination. Environmental monitoring at regional level ("Regione" is an administrative district), is being set up (1).

The control of the radioactivity in the environment was mainly focused on artificial sources; as far as natural radioactivity is concerned no regular surveys were carried out, except in some cases of occupational exposures. Extensive research work was, however, performed to develop suitable radon monitoring devices and the range of radon levels indoors and outdoors have been investigated throughout the country and locally (2,3,4). Considering the radiological relevance of the radon problem (5,6) it was deemed that the exposure of the population due to natural sources of radiation indoors deserves particular attention.

2. Project of the Italian survey on radon indoors.

The knowledge of the average Equilibrium Equivalent Radon concentration existing in dwellings is the first parameter needed to plan whatever action aimed at preventing and/or reducing the exposure to radon. A two step survey has been therefore planned in our country with two different purposes: a) to evaluate the average effective dose equivalent due to radon indoors and b) to identify areas where the radon concentration levels indoors are considerably higher than the average. In these places measurements shall be carried out in a systematic way.

The first step of the survey will permit the evaluation of the average risk associated with radon exposure and the comparison of the radon risk with other sources of domestic risks. In such a way the competent Authorities will be enabled to choose their policy in the field. A comparative study of all the pollutants existing indoors (organic and inorganic compounds, mineral fibres and particles) is, in principle, the correct policy approach. The study should allow to evaluate the social and economical impact due to the indoors pollutants in its entirety. The degree of knowledge in the case of radon is better than in the case of other pollutants, from the point of view either of the detection techniques and of the evaluation of the health effects. Therefore it is considered advisable to limit the survey to the radon indoors.

The second step will allow to select hot spots of radon concentrations and will be the starting point for future actions. The environmental laboratories at regional level will play an important role in this type of survey as far as the measurements are concerned and will operate according to guidelines and procedures established at national level.

3. Statistical criteria to select a sample of dwellings at national level

Keeping in mind the above considerations a sampling technique has been implemented for evaluating the average effective dose equivalent due to radon indoors.

The selection of a sample of dwellings which can be considered representative of the Italian situation is limited by the availability of manpower and economical resources. A sample of 5000 dwellings distributed in about 200 different towns (i.e., the smallest administrative units) was considered a reasonable compromise between cost and representativeness. A two-stage sampling technique is used. The parameters examined to stratify the samples are:

1) geolithology of soil, rather well known and showing concentration of natural radioactive materials in a wide range of values;
2) building materials, which in some cases play an important role for the radon concentration;
3) climate conditions, which influence the way of life indoors;
4) floor, where the flat is located in a multifamily house, being the radon concentration indoors dependent on the height from the ground.

The Italian territory is divided into seven areas homogeneous in the respect to the geolithology. The cooperation of the Rome University, Department of Earth Science, is acknowledged in performing such division. Interference between geological areas and the administrative units created operational difficulties preventing the use of geolithology as stratification parameter. Correlations "a posteriori" will be performed between the geolithogical areas and the radon indoors concentration. For building materials no satisfactory information exist at national level.

Starting from these considerations, in the first stage the administrative units are chosen and in the second stage the dwellings are singled out. The first stage of the sample is stratified according to a geographycal criterion (five zones are selected) and to the population density (six classes are considered appropriately weighted). In the second stage the strata are chosen according to the age of construction (before 1919; 1919-1945, after 1945) and to the floor where the flat is located (ground floor, first floor, over).The choice of the dwellings within each stratum is at random.

The selection of the sample is made with the help of the Istituto Centrale di Statistica using the data of a general census carried out in 1981.

4. Operational problems

To carry out the survey several economical, political and organizational problems have to be solved. First of all, to avoid undue and dangerous alarm, a public information campaign shall be carried out directed both to the general public and to health and administrative authorities involved. Transparency should be mantained in performing the survey and it is also important to avoid any type of economical speculation - the spread of private firms performing radon measurements do not give any sound contribution to the knowledge of the problem.

The training of the personnel responsible for the distribution of the dosimeters is matter of great importance. No postal distribution is foreseen as the door to door approach is considered having the largest probability of success. The personnel shall be able to satisfy the questions posed by the inhabitants on the scope of the survey, its significance, etc. Particular attention will be required in preparing a questionnaire to be filled when the dosimeters will be distributed. The questionnaire has to contain the information useful to correlate the experimental measurements with those parameters related with the indoor radon concentrations, such as heating system, source of waters, etc. To evaluate the average effective dose equivalent, other information on the inhabitants, their habits, distribution of ages, shall be collected.

5. Experimental set up

A national large survey as that just outlined requires the development of a radon monitor device simple, sensitive, compact and inexpensive. The detector consists of a heat-sealed polyethylene bag enclosing two track etching detectors (cfr.4). The inside of the device is completely transparent and this is particularly important for psycological reasons.

One detector is a piece of LR-115: the tracks can be counted by the spark counter after the chemical etching. The response of the detector saturates at about 3500 Bq/m^3 in 6 months. The other detectors is CR-39 and is used as back up dosimeter. It is chemically etched and can be counted by image analysis or by a microscope. The dosimeters have been extensively used for indoor and outdoor measurements.

TLD-dosimeters will be used to measure gamma ray components. Both the radon and gamma ray dosimeters were calibrated at international level.

Intercomparison excercises will be also planned between the laboratories which will take part into the survey.

6.Conclusion

The experience gained in the preliminary surveys (2, 3) allows to say that the average radon indoor concentrations will range around 20 - 50 Bq/m^3, of the same order of magnitude of those detec-

ted in other European Community Countries. Nevertheless there are sound reasons to presume there will be locations where these levels can be exceeded by an order of magnitude.

It is clear that before undertaking whatever action, the extent of the problem shall be evaluated with a certain degree of confidence. It is also clear that the principles given by ICRP in its Publication n.39 for limiting the exposure of the general public should followed; as far as the possible actions are concerned, the distinction between existing and future housing is of paramount importance. Attempts should be made to set up a range of action levels in the first case and of upper bounds in the second one. In order to prevent confusion and negative effects on the public opinion it should be avoided that at least in the European Community countries different limits be set in Member States.

Acknowledgements

The authors are very grateful to dr. G.Coccia and dr. G.Di Traglia of the Central Institute of Statistics for the collaboration in the sample selection and to prof. G.Lombardi of the Rome University, Department of Earth Science, for useful discussions and cooperation.

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THE ORIGIN OF INDOOR RADON

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INTRODUCTION

The major source of radon in single-family dwellings is the soil underneath (1). The principal route of entry into dwellings built on concrete slabs is through cracks and other openings. Two processes operate: the diffusion of radon through air-filled pores of soil and the pressure-driven flow of soil air. Small but persistent pressure gradients are created in the soil by the buoyancy of warm indoor air (the stack effect) and the effects of wind across open chimneys (the venturi effect) and across the fabric of the dwelling. Empirical and semi-empirical information indicates that pressure-driven flow is often the dominant process, particularly in the case of high radon levels (2,3,4). More recently, the radon transport equation describing the flow of radon from soil into dwellings through cracks in the concrete floor slabs has been solved numerically using the finite difference method (5). The solutions have been used to indicate the circumstances under which the pressure-driven flow mechanism might be expected to dominate and assist in the quantification of the parameters that determine high indoor radon levels.

MODELS OF RADON MOVEMENT

The steady state transport equation for the flow of radon in soil under concentration and pressure gradients is given by:

$$\frac{D_e}{p} \nabla^2 C_{Rn} - \frac{1}{p} \underset{\sim}{v}.\nabla C_{Rn} - \lambda (C_{Rn} - C_o) = 0 \qquad \ldots\ldots (1)$$

where D_e is the effective diffusion coefficient of radon in soil, p is the soil porosity, C_{Rn} is the soil radon concentration and λ is the radon decay constant. The model assumes a uniform volume source of radon and C_o is the concentration deep below the surface. $\underset{\sim}{v}$ is the linear flow velocity and is given by:

$$\underset{\sim}{v} = - \frac{k}{\mu} \nabla P \qquad \ldots\ldots (2)$$

where k is the air permeability of soil, μ the viscosity of soil gas and ∇P is the pressure gradient.

This model has been applied to a concrete raft 2r metres wide with a 2w cm wide crack in the centre. The length of the crack was assumed to be large in comparison with its width and height. The volume of soil beneath the raft was taken to be 5 m deep and extended 4 m beyond the walls. The concrete slab was assumed to be impermeable. This geometry can be repeated to represent periodic cracks.

From the transport equation, the concentration gradient of radon and hence radon flux at the top of the crack were determined (5). In this paper we calculate the radon concentrations that would exist in a dwelling with a ventilation rate of 0.5 air changes per hour, a volume of 250 m^3 and a floor area

of 50 m^2. The following values were also used: $\mu = 1.83 \times 10^{-5}$ kg m^{-1} s^{-1}, $D_e = 9 \times 10^{-7}$ m^2 s^{-1} and p = 0.35.

THE EFFECT OF CRACKS

Figure 1 shows the radon concentration in the dwelling plotted against air permeability of soil for different crack widths and periodicities. The radon concentration at depth was taken as 10^4 Bq m^{-3} and the pressure difference at floor level between outside and inside air was taken as 5 Pa. This would arise, for example, in a dwelling with a temperature difference between inside and outside of about 20°C. The radon concentration in a dwelling with a bare floor is also shown.

The following conclusions can be drawn from the figure:

a) The crack width and periodicity have a relatively small effect on the indoor radon concentration. This would imply that reducing high radon ingress in dwellings by caulking of cracks in a concrete floor will only be effective if all cracks are sealed. Even minor cracks may still be significant sources of radon.

b) At permeabilities below 10^{-10} m^2, the radon concentration in a dwelling with a cracked floor is not substantially lower than would arise from a bare floor. The reduction in radon concentration from a cracked floor at high permeabilities is due to fresh air being drawn from outside through the soil adjacent to the walls. The model used for the bare floor did not however take account of the effect of fresh air being drawn from soil adjacent to the walls.

THE EFFECT OF PRESSURE DIFFERENCE

Figure 2 shows the indoor radon concentration plotted against air permeability for different pressure differences at floor level between outside and inside air (+ denotes that the room is at lower pressure than atmospheric; - denotes a higher pressure). The crack width was taken as 0.5 cm with a 1 m period. C_o was again taken as 10^4 Bq m^{-3}. The concentration due to diffusion only is also shown ($\Delta P = 0$).

At permeabilities below about 10^{-12} m^2, the increasing pressure difference has relatively little effect on indoor radon concentrations. Between 10^{-12} and about 10^{-10} m^2, the concentration increases to a peak value which is about 15 times that due to diffusion alone. This peak is shifted to lower permeabilities with increasing pressure difference but the magnitude of the peak value is unaffected. Beyond the peak, the concentration decreases with increasing permeability for the reason given above. Extending the sealing of the soil immediately adjacent to the wall would reduce the depletion of the radon in the soil and hence increase the radon flux into dwellings.

For soil with a permeability in the region of 10^{-11} to 10^{-10} m^2, increasing the pressure difference from 2 to 10 Pa, causes an increase in the radon concentration by a factor of about 3. Further increase in the pressure difference would not increase the radon concentration beyond a maximum level owing to the fact that the position of the peak is progressively moving towards the lower permeabilities.

The reduction in radon concentration due to a negative pressure difference is marked. The implication is that over-pressurisation of a dwelling may well be successful in reducing high radon levels. However, over-pressurisation of the dwelling itself may well result in condensation being produced in the walls. Over-pressurisation of the soil under a fairly intact concrete slab may however be an effective means of reducing high levels and avoid this problem. Such an effect has been demonstrated (6).

CLASSIFICATION OF SOILS

Figure 3 shows the bands of radon concentration C_o and air permeability of soil which result in radon concentrations in dwellings of 100 and 400 Bq m^{-3} for a range of pressure differences. These values correspond respectively to the NRPB recommended upper bound for new dwellings and action level for existing dwellings (7). The figure therefore defines the circumstances in terms of permeability and radon in soil gas concentrations in which these levels might be exceeded. If assumptions are made regarding the radon emanating fraction of soils (here, an average value of 20% is used), this figure leads to a relatively simple classification of soils in terms of both radium-226 activity and radon in soil gas concentration. This is shown in Table 1 along with examples of possible soil types. Similar classifications based on more empirical considerations have been proposed by other workers (2,8,9) and this work provides some theoretical support.

It is noted that the model discussed here does not deal with two matters, both of which may significantly influence radon ingress into dwellings. These are tunnel flow caused by cracks and fissures in the underlying rock and the presence of a water table. In the first situation, the ingress of radon will not be dependent on permeability, and it is possible that soil gas could be transported over large distances and lead to high indoor radon levels. Some evidence for this mechanism is provided by the relatively high levels observed in limestone regions (1). The presence of a high water table will on the other hand lead to reduced soil gas flow, and this also has been observed (2).

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Table 1: Combination of radiological parameters and air permeabilities of soils that might lead to radon concentrations above the Upper Bound of 100 Bq m^{-3}

^{226}Ra (Bq kg^{-1})	C_o (Bq m^{-3})	k (m^2)	Example
25	10^4	10^{-10}-10^{-9}	Gravel
100	5 10^4	10^{-11}-10^{-10}	Sand
500	2 10^5	10^{-12}-10^{-11}	Silt
2000	2 10^6	10^{-13}-10^{-12}	Till

Figure 1: Indoor radon concentration versus soil permeability for different crack widths and periodicities

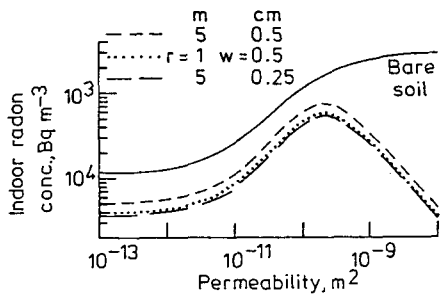


Figure 2: Indoor radon concentration versus soil permeability for different pressure differences

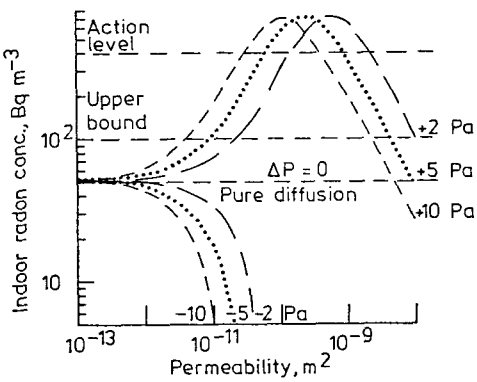
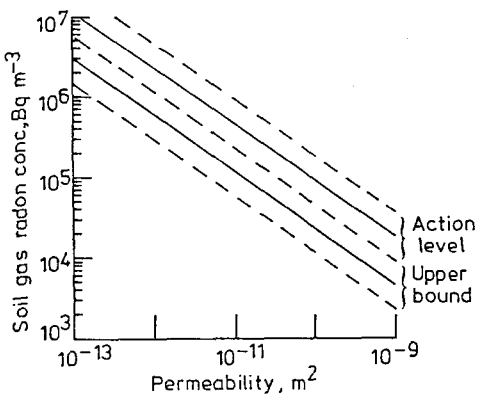


Figure 3: Soil gas radon concentration, C_o, versus soil permeability showing bands of values defining the Upper Bound and Action level



AN EXPERIMENTAL STUDY ON EXHALATION OF ^{222}Rn AND ^{220}Rn FROM THE SOIL AND INFLUENCING PARAMETERS

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ABSTRACT

Radon exhalation from building material is the main source of normal radon activity concentration in dwellings in the Federal Republic of Germany. On the other hand exhalation of radon from the soil is the determining factor for the highest measured indoor radon concentrations.
A special measuring device has been developed for our investigations in order to determine directly the exhalation rates of radon and thoron from the ground. Using electrostatic deposition of the radon daughter products Polonium-218 and Polonium -216 on a surface barrier detector and subsequent analysis of the measured alpha spectra the daily course of the radon exhalation from the soil has been recorded. For a sampling and counting time of 2 hours the lower limit of detection of our equipment has been established to be $0,3 \cdot 10^{-3}$ $Bq/m^2 \cdot s$ for Radon-222 and $3 \cdot 10^{-3}$ $Bq/m^2 \cdot s$ for Radon-220. During summer and autumn of 1986 the radon exhalation rates have been determined for various types of soil and different atmospheric conditions. It was found that radon exhalation rates vary between $2 \cdot 10^{-3}$ and $100 \cdot 10^{-3}$ $Bq/m^2 \cdot s$ whereas the maximum values for Radon-220 mounted up to 1 $Bq/m^2 \cdot s$. Moreover the temperature of the soil and the air and such parameters as porosity and water content of the soil under investigation have been correlated to the measured radon exhalation rates. The movement of the water through the pores of the ground and the temperature gradients between the upper layers of the soil and the atmosphere are the decisive factors which dominate the transport of radon through the soil. A cartographic analysis of the results obtained showed regions with relatively high radon exhalation rates. In order to avoid high indoor radon concentrations caused by the exhalation from the soil in these regions special measures for the construction of buildings are recommended.

ASSESSMENT OF INDOOR RADON LEVELS IN SINGLE FAMILY HOMES IN NEW MEXICO, U.S.A.

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ABSTRACT

The object of this study is a statewide assessment of levels of indoor radon in single family homes in the State of New Mexico, U.S.A. Data from the northeast portion of the state are reported in this abstract. Monitoring data were collected over a 24 hr. period with two continuous working level monitors per home using protocols recommended by Environmental Protection Agency.

Data collected in northeast New Mexico during the winter of 1986-1987 showed the following percentages of homes were above the recommended guidelines of 0.02 WL established by E.P.A.: 1) 20% of 56 homes in the greater Santa Fe area, 2) 50% of 36 homes in the Pojaque Valley, 3) 34.8% of 46 homes in the Espanola area, 4) 38% of 13 homes in Los Alamos and White Rock area, and 5) 75% of 12 homes in the Taos area.

The respective means and ranges for these five areas were : 1) Santa Fe, mean = 0.015 WL (n=56), range = 0.003 WL to 0.078 WL; 2) Pojaque Valley area, mean = 0.025 WL (n=36), range = 0.004 to 0.074 WL; 3) Espanola area, mean = 0.019 WL (n=46), range = 0.0025 WL to 0.067 WL; 4) Los Alamos and White Rock area, 0.020 WL (n=13), range = 0.004 WL to 0.051 WL; 5) Taos area, mean= 0.039 WL (n=12), range =0.007 WL to 0.092 WL.

Since adobe brick construction materials are fairly common in New Mexico, the hypothesis was tested that radon/radon daughter levels are increased in homes with this type of construction. The respective means for areas in northern New Mexico are as follows: 1) Santa Fe area, mean of adobe house construction = 0.0140 WL (n=18), non-adobe construction mean = 0.0143 WL (n=37); 2) Pojaque Valley area, mean of adobe houses = 0.0276 WL (n=24), mean of non-adobe houses = 0.0190 WL (n=10); 3) Espanola area, mean of adobe houses = 0.0215 WL (n=23), mean of non-adobe houses = 0.0166 WL (n =20). The means of these groups were not found to be significantly different.

Data will also be reported from other areas of the state, this data being collected using the charcoal cannister method (two per home) over a 48 hr. period.

HEALTH RISKS OF INDOOR RADON GAS

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A WHO Regional Office for Europe organized Working Group discussed radon at a meeting in Dubrovnik, Yugoslavia, in August 1985. In the light of the anticipated and observed health problems caused by radon, the Group was asked to review and evaluate the risks involved, and to recommend appropriate guidelines.

Much of the natural background radiation to which the general public is exposed comes from the decay of radium-226 which produces radon gas and other products. Because radium is a trace element in most rock and soil, indoor concentrations of radon can come from a wide variety of substances, such as building materials and the soil under building foundations. Tap water taken from wells or underground springs may be an additional source. Tests indicate that indoor concentrations of radon and its decay products are often higher than those outside. High indoor concentrations of radon are of concern due to the potential carcinogenicity of its decay products. (The US EPA has estimated that current exposure to radon gas could account for as much as 10% of all lung cancer deaths in the USA.)

OBSERVED LEVELS OF RADON AND RADON DAUGHTERS IN INDOOR AIR

The radon daughter concentration in indoor air is a function of the rate of entry into the building from the various sources, the effective ventilation rate removing radon from the building and the rate at which radon daughters settle or plate onto surfaces. A sizable number of buildings have been monitored, and the reports show that the range of mean concentrations is 10-50 Bq/m^3. However, in a number of countries some dwellings exceed even 400 Bq/m^3 equilibrium equivalent radon (EER) concentrations. It is also clear that a small fraction of dwellings in each country has concentrations exceeding 10 times the national average of that country. In general, the average level of radon daughters in buildings can be expected to be higher in a region with high terrestrial gamma flux as has been found for Finland [1], and as has also been found for Cornwall as compared to the whole of the United Kingdom [2].

EXPOSURE AND EFFECTS

The critical organ for exposure to radon daughters is the lung. The uptake of radon gas by all other routes is minor. Therefore, the air route is of the major concern. Upon being inhaled, the radon daughters will be carried into the bronchial region of the lungs and a substantial fraction will be deposited and remain there. The radioactive half-life of the radon daughters is of the order of 30 min, so that once deposited the alpha emissions will impact on the immediate region [3] and deliver a very densely

concentrated ionization in that tissue. The amount of radon daughter intake depends on the breathing rate which affects both the total quantity inhaled per unit time and the pattern of deposition in the respiratory tract. The volume of air inhaled at rest is about 0.5 m^3/h; during physical activity this volume can reach 1.5 m^3/h or higher. The fraction of attached radon daughters inhaled which are deposited depends on the size of the particles and the breathing patterns but has been estimated as 30%.

A minimum and irreducible exposure to radon daughters of about 1.5 Bq/m^3 EER occurs outdoors. But most people spend 80-90% a day in residential and nonradiation-connected occupational environments, which constitute their major source of radon exposure. The concentrations of radon daughters in these environments are 10 to $10^4 Bq/m^3$ EER, and sometimes even higher.

UNSCEAR estimated a mean indoor concentration of 15 Bq/m^3 EER as an average for the total population of the world's temperate regions [4]. Hence, a mean dose equivalent of about 15 mSv/a to the bronchial and 2 mSv/a to the pulmonary region from indoor exposure to radon daughters should be expected, corresponding to an effective dose equivalent of 1 mSv/a. The indoor exposure at home accounts for 70-80 % of this dose. In population groups living in houses with strongly enhanced radon levels, considerably higher dose levels to the target tissues in the lung will occur.

Underground mine workers, and especially uranium mine workers, are exposed to substantial concentrations of radon daughters, equivalent to 5-days-a-week and 8-hours-a-day exposures of 1500 Bq/m^3 EER. For residential and non-occupational indoor exposures, this would correspond to about 430 Bq/m^3 for an exposure factor of 80% indoor occupancy. The increased lung cancer risk among miners, and especially uranium miners, has been well documented in several large epidemiological studies. Results from the follow-up of more than 25 000 radon-exposed miners have been summarized and reviewed [4,5,6,7]. The results are consistent with the conclusion that at least for the lower dose range, the relationship between cumulative exposure to radon daughters and excess frequency of lung cancer is linear. This conclusion is also supported by the results of animal exposure studies and human studies on the effects of densely ionizing radiation. Assuming an 80% occupancy of indoor environments, an annual incidence of 10-40 cases per million persons should be expected that is attributable to exposure to radon daughters. The model of relative risk leads to the conclusion that under these conditions, about 5-15 % of the observed frequency of lung cancer or of the lifetime risk may be attributable to radon daughters in the indoor environment. This relative risk is nearly equal for males and females, and for smokers and nonsmokers. The combined effect of exposure to radon daughters and tobacco smoke can thus approximately be described as multiplicative, and people exposed to radon daughters can especially reduce their risk of lung cancer by avoiding tobacco smoking.

CONCLUSIONS

1. Radon daughter concentrations in indoor environments are

considerably higher than outdoor concentrations which are of the order of 2-5 Bq/m^3 EER. Mean population averaged concentrations in houses have been found to range to 50 Bq/m^3 EER.

2. In any region, the range of indoor radon daughter concentrations is substantial, with an approximately log-normal distribution; a small number of buildings have values of 10 times the median concentrations or more.

3. The estimated risk of lung cancer attributable to inhaled radon daughter concentrations indoors is a significant fraction of the total lung cancer risk. It is estimated that at the observed mean levels indoors, about 10% of all lung cancer cases might be caused by radon daughters. Corresponding higher risk values should be expected in members of the public who are chronically exposed to higher indoor levels. At the high end of the concentration distribution, the risk is of the order of that caused by cigarette smoking.

4. Although radon daughter concentrations cannot be reduced to zero, it is possible to reduce them, especially the higher concentrations. In most existing buildings, the radon concentration can be reduced by a variety of measures. In new construction, this can be done even more effectively.

5. Reducing exposure to radon daughters is an effective approach to reduce lung cancer risks.

RECOMMENDATIONS

1. In general, buildings with radon daughter concentrations of more than 100 Bq/m^3 EER as an annual average should be considered for remedial actions to lower such concentrations, if simple measures are possible.

2. Remedial actions in buildings with a radon daughter concentration higher than 400 Bq/m^3 EER as an annual average should be considered without long delays (the total dose before remedial action should not be allowed to exceed 2000 Bq.a/m^3 EER).

3. Building codes should have sections designed to avoid levels exceeding 100 Bq/m^3 EER in new buildings and should prescribe appropriate practices.

4. All appropriate authorities should consider representative surveys of radon daughter concentrations in their building stock to identify areas that may have excessive levels.

5. Appropriate epidemiological studies should be conducted in regions of high and varied radon daughter concentrations, where cancer case registries can be used to estimate the relative and absolute risks of lung cancer related to different exposures. Specifically, countries in those regions, such as northern Europe, can separately or jointly perform case-control studies, with appropriate controls for age, sex, occupational exposure, site of exposure and tobacco smoking.

6. Appropriate animal and epidemiological studies are still required and should be conducted to develop exposure-response relationships.

7. Further studies should be conducted to investigate the dependence of indoor radon daughter concentrations on sources, ventilation rates and other factors to improve the basis for criteria to identify areas or houses with excessive concentrations and for measures to limit such concentrations.

8. Comparison and intercalibration exercises between organizations involved in the measurement of radon and its daughters should be organized at regular intervals.

ACKNOWLEDGEMENT

The final, complete report was issued as a document in Environmental Health 13 by the WHO Regional Office for Europe, Copenhagen, 1986. This paper is based on that report. The views expressed are those of the author and the WHO Group of Experts and do not necessarily represent the decision or stated policy of WHO.

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VENTILATION TECHNIQUES AND RADON IN SMALL HOUSES

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INTRODUCTION

Indoor radon is the main cause of radiation exposure in Finland. The National Board of Health set the recommended concentration limits in 1986: an action level of 800 Bq/m^3 and a planning value of 200 Bq/m^3 for new buildings. According to Castren et al. (2), the 800 Bq/m^3 concentration is estimated to be exceeded in 1.4 % of the housing. This rather high number has motivated a number of studies concerning countermeasures against radon in existing houses.

In general, the primary types of countermeasures reported have been: locating and sealing of radon penetrations, sub-floor ventilation (or depressurizing) systems and increased ventilation (e.g.3,4,5,11,12,13). In cost-benefit studies best applicable to the Finnish circumstances, the sub-floor measures have been found to be normally superior to the ventilation measures (3). The use of ventilation measures may, however, be reasonable when the existing ventilation rate is very small. An important point often neglected is the beneficial effect of increased ventilation on the general indoor air quality.

Our approach to the radon problem is based on the foundings of a large national indoor air survey (7,8), which clearly indicated poorly designed or adjusted ventilation systems to be common. A large variation was observed in the air exchange rates: although the ventilation rates in average were acceptable, the air exchange rates in a large number of small houses were found to be very small and well below the limits set by the Construction Code. One reason for the low flow rates are the poor (non-existing) supply air systems which were found to be common. The increased underpressure due to the lack of supply air inlets often contributes to high radon concentrations even if the exhaust system is capable of producing an adequate ventilation rate. This is caused by the pressure-driven radon flow through the sub-surface structures (1,9,10). It is evident that a number of houses exist where problems are encountered with indoor air impurities, whenever the impurity source is enhanced. In addition to radon, high concentrations of CO_2 in bedrooms have been observed.

The purpose of this study was to find out possible remedial actions against radon using standard ventilation techniques. The ventilation rates were not increased over 0.7 1/h in order to have a realistic view about the possibilities of the state-of-the-art techniques. Special attention was given to methods which would be generally applicable to a large number of dwellings already existing. Results are reported of a pilot study with six small houses with established high radon concentrations.

EXPERIMENTAL

The six measurement sites were selected from a larger sample of dwellings, where high radon concentrations had been measured. The studied dwellings represent different types of construction and ventilation techniques. Table 1 shows some basic data about the houses. The summer and winter radon concentrations were measured by the Finnish Centre for Radiation and Nuclear Safety using solid state nuclear dosimeters.

Table 1. Measuring sites.

Site	Rn, Bq/m^3 Summer/Winter	Construction year	Total floor area, m^2	Stores	Ventilation
1	470/1840	83	173	2	exhaust
2	1280/1960	85	216	2	supply/exhaust
3	1265	84	216	2	exhaust
4	1430/1960	29	103	3	natural
5	250/1810	83	115	1	natural
6	280/430	80	104	1	supply/exhaust

Continuous monitoring of radon was performed using meters constructed in our own laboratory (6). Air exchange rates, air flow rates and pressure differences were also measured. Meteorological conditions — wind direction and speed, temperature and atmospheric pressure were obtained from the data registers of a nearby weather station.

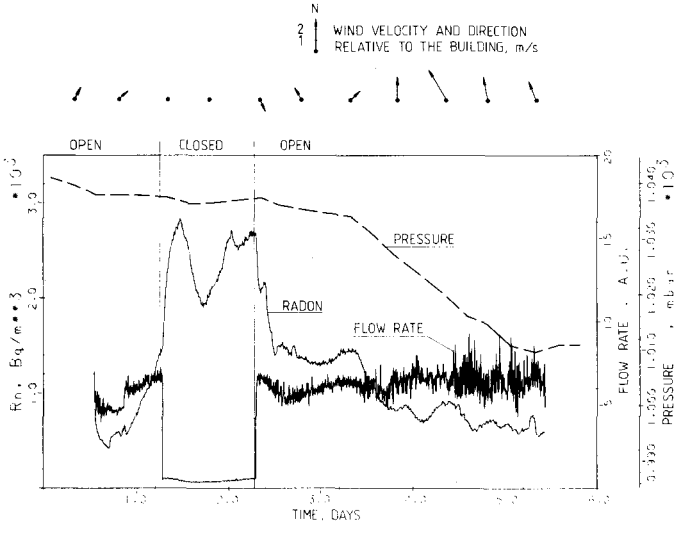

Figure 1. The effect of the valve of the supply inlet installed in house 1 on the radon concentration. The effect of atmospheric conditions on the concentration and on the flow rate is also evident.

The remedial actions ranged from opening an obstructed air inlet to installation of complete ventilation systems. Two major types of commercial systems were used, both constructed to minimize the underpressure in the building: balanced mechanical ventilation with heat recovery and passive supply inlets with electric heating. The supply inlets produce a relatively large flow rate at a low pressure drop. The heating of the supply air is essential during the wintertime. No measures were performed in site 6, whereas the others were subject to following:

Site 1. The ventilation was re-adjusted. A supply air heater was installed downstairs (air ducted in to the two bedrooms). The effect of the inlet on the radon concentration is shown in Figure 1, as is the effect of atmospheric conditions on the flow rate through the device.

Site 2. The ventilation was adjusted and the obstructed major supply air inlet opened.

Site 3. A mechanical supply/exhaust ventilation system was installed, utilizing the existing exhaust ducting.

Site 4. Routes of radon entry from the soil sealed, an exhaust fan and a supply air intake with heater installed.

Site 5. A mechanical supply/exhaust system installed. Figure 2 shows the effect of the system on the radon concentration.

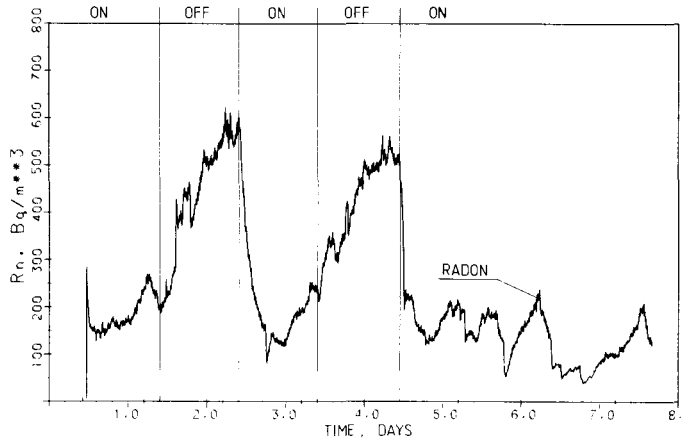

Figure 2. The effect of the installed supply/exhaust ventilation system on the radon concentration in site 5. The ventilation rate, while system OFF, is approximately the same as before the installation.

RESULTS AND DISCUSSION

Table 2 shows the obtained concentration decrease in the five dwellings and the approximated maximum costs including planning, equipment, installation and adjusting. The running costs are not presented, but were at maximum in site 5: approximately 500 FIM/year.

Table 2. Obtained decreases in the radon concentration and approximated total costs.

Site	1	2	3	4	5
Concentration decrease %	70*	40*	50	70	75
Total cost (FIM)	4700	1000	12000	8500	14000

*in bedrooms

The concentration decrease is a good approximation for wintertime. During summer the decrease is expected to be smaller. Observed large momentary deviations in the radon concentration set demands for the measurement of the effect of the measures. The on-off method used here combined with a long-term integrating measurement is believed to give reliable results. Because of the sometimes large yearly variations measurements should also be done in different seasons. The best results were obtained in sites 1, 2 and 5, where the decrease in the radon concentration was equal or better than the increase in the ventilation rate. In multi-stored houses 3 and 4 (partly below the surface level) complete control of the air could not be achieved with the simple systems used.

CONCLUSIONS

Although not extensive, the preliminary study shows that the use of standard, commercially available (balanced) ventilation equipment may be a cost-effective alternative as a remedy against high radon concentration. Apart from the reduction in the radiation exposure, these methods have other beneficial effects: decrease in the concentration of all indoor-source impurities as well as increased living comfort. It should also be remembered that the decrease in the potential alpha energy concentration obtained by increased ventilation is normally larger than the decrease in the radon concentration. The example of site 2 shows that the existing ventilation system should always be inspected to ensure proper functioning before other countermeasures take place.

It is evident that increased ventilation is a cost-effective measure only in houses where the existing air exchange rate is small and/or the increase can be easily obtained. It is equally evident that there is a considerable amount of houses in Finland fulfilling at least the first condition. The next step is to find out if it is possible to characterize these dwellings so that they may be identified without extensive studies on the basis of basic data about the construction. Obtaining simple model solutions with installation instructions for these houses is the final object of the extension study.

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INDOOR RADON LEVEL MEASUREMENTS IN IRAN USING AEOI PASSIVE DOSIMETERS

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ABSTRACT

A passive radon diffusion dosimeter was developed at the RPD of AEOI for nationwide indoor radon level measurements. Several parameters of the dosimeter were studied. Radon levels were determined in about 250 houses in Ramsar (a high natural radiation area), Tehran, Babolsar and Gonabad. In this paper, the results of some dosimeter parameters as well as radon levels in indoor air are reported.

INTRODUCTION

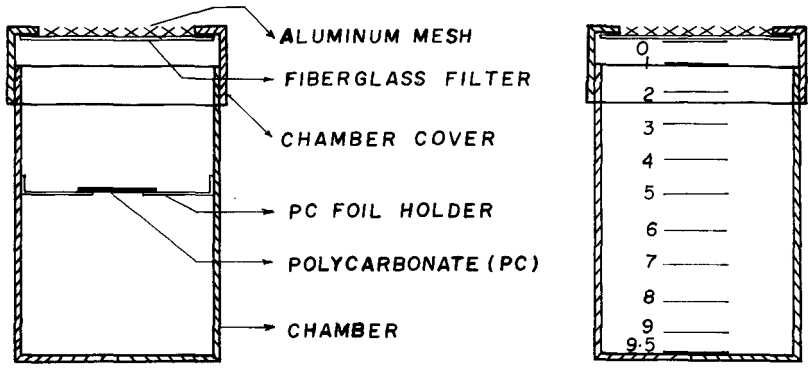
Some epidemiological studies on uderground miners exposed to ^{222}Rn and its daughters have shown an excess lung cancer risk (1).It is also known that the inhalation of radon and its daughters is the major source of natural radiation exposure of general polulation. On the average, about half of the annual effective dose equivalent(0.95 mSv) of public living in normal radiation areas is due to Rn and its daughters, as reported by UNSCEAR (2). The urgence for radon control indoors has enforced limitation schemes in many countries (3). Therefore, research and development on radon measurements have become of some concern in Iran.

At first, active CEA-type monitors using LR-115 and Lucas chambers were applied (4,5). Later, due to some new findings on efficient registration of alpha tracks in polycarbonate (PC) by electrochemical etching (6,7,8), this polymer was used inside a cup for radon detection; the same principal originally proposed and used by Alter and Price (9) and Alter and Fleischer (10) using cellulose nitrate films. Therefore, PC foils were used inside open cups for uranium exploration studies in Saghand and Arousan areas of Yazd province of Iran (11), or inside diffusion chambers for radon dosimetry in indoor air of houses in Iran.

SYSTEMS AND METHODS

A passive radon diffusion dosimeter was developed at the RPD of AEOI as shown in Fig.1(a,b). It consists of a plastic cup, a top cover, a 0.025 cm PC foil, a fiberglass filter under a circular opening of the top cover with an aluminum mesh support. The cup and the cover are made tight together by a thick masking tape. The radon gas can diffuse through the filter while stopping the daughters. The foil position was optimized inside the chamber at different distances from the filter (Fig.1b). Chambers with different heights and different openings were constructed. About 1000 dosimeters were made by hand at optimized conditions.

The PC foils were etched using our new multi-chamber electrochemical etching system (13), at the conditions stated below. After ECE, the tracks were counted under a light microscope. The lower detection limit was found at 50% relative standard deviation versus track density to be 52 Bq/m^3 for 3 months detection period.

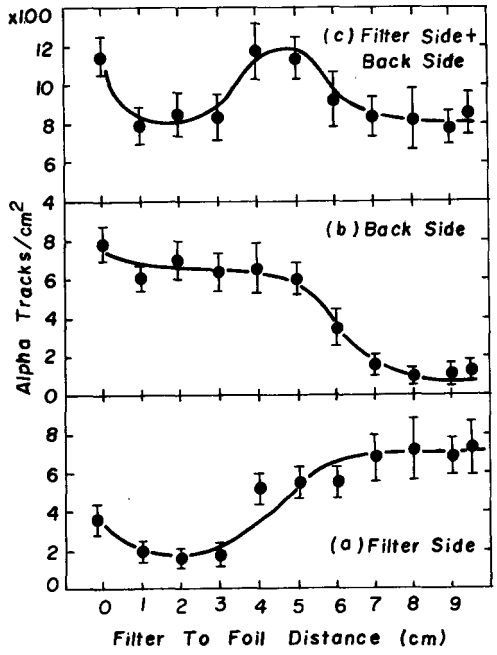


Figs.1(a,b)-Cross Section of AEOI Passive Radon Diffusion Dosimeter.

The radon dosimeters were exposed and calibrated inside a glovebox with some uranium ore inside to provide a natural radon field. The Rn field was calibrated by Lucas scintillation cells calibrated in turn by using a Pylon Standard Source type RN-150. Based on this calibration, a sensitivity of 0.05 ± 0.007 track.cm^{-2}/Bq.m^{-3}.day or 1.86 ± 0.28 tracks.cm^{-2}/pCi.l^{-1}.day was obtained by etching PC foils under 32 kV/cm, 2kHz with 1 hr pre-etch followed by 2.5 hrs ECE in a mixture of 15g KOH + 40g C_2H_5OH + 45g H_2O at 25 °C.

RESULTS AND DISCUSSION

Details on the dosimeter parametric studies and ECE optimization have been reported recently (12), some results of which are reported here. Figs.2(a,b,c) show the effects of 11 foil positions inside 11 different chambers with 296 cm^3 volume and 5 cm filter diamter.Since PC has an upper cut off energy of 1.8 MeV (14), there should be enough length inside the chamber for attenuation of 5.49, 6.00 and 7.68 MeV alphas due to ^{222}Rn, ^{218}Po and ^{214}Po respectively with 3.8, 4.3, and 6.3 cm ranges in air to be registered efficiently. The side of the foil facing the filter (0 cm, Fig.1b), will register only the alphas attenuated in the filter itself while its backside register all alphas properly attenuated in the chamber length, thus showing the highest sensitivity (Fig.2b). On the other hand, the side of the foil facing the chamber's buttom (9.5 cm, Fig. 1b) shows the lowest sensitivity (Fig.2b) while its filer side shows a high sensitivity which stays constant for distances down to 5 cm from the filter. Thus the foil in the middle shows an optimum and equal sensitivity on both sides, as shown in Figs. 2(a,b) with a sum showing the highest sensitivity(Fig.2c). Thus the position with 5 cm distance from the filter was selected optimum.



Figs. 2(a,b,c). Effects of foil position on the sensitivity of a)filter side, b)back, c) both.

Fig.3 shows the effect of changing the chamber height and thus its volume on sensitivity. As volume increases, the sensitivity increases due to proper attenuation of alphas in air to an energy registerable in PC. Therefore, as shown in Fig.3, 296 cm^3 chamber shows about five times higher sensitivity than that of 140 cm^3;i.e. five times higher sensitivity for almost doubling the volume.

Fig.4 shows the effect of filter diameter on sensitivity. When there is no opening in the top cover, the dosimeter still shows some sensitivity. By increasing the filter size from 0 to 1 cm diameter, the sensitivity has been increased rapidly to about 2.5 times which has stayed constant up to 6 cm studies. Thus, 2 to 3 cm filter diameter was determined as optimum filter size.

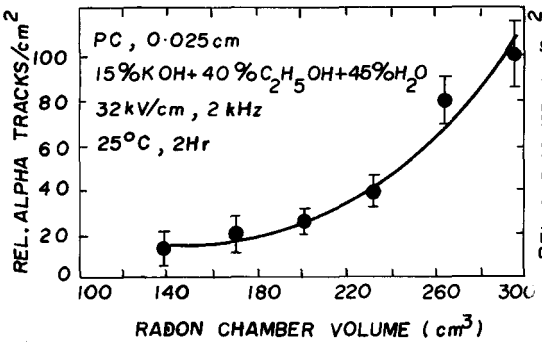


Fig.3 The Effect of Chamber Volume on the Dosimeter Sensitivity.

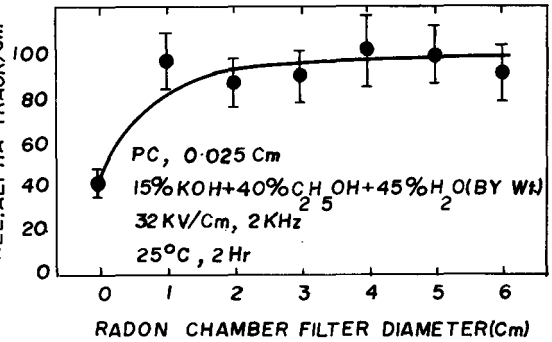


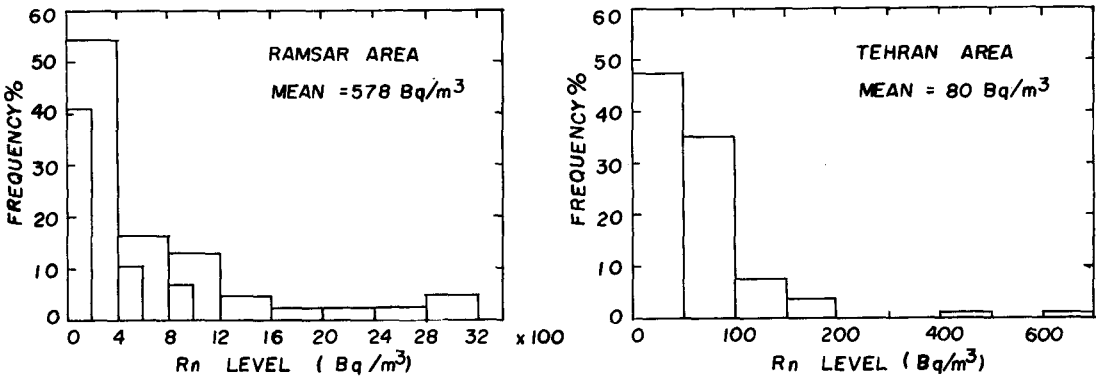
Fig.4 Effect of Filter Diameter on Dosimeter Sensitivity.

Radon level measurements were carried out in about 250 randomly selected houses in Ramsar, Tehran, Babolsar and Gonabad cities of Iran. Table 1 shows the brief results of radon levels in about 250 houses in these cities. Figs. 5 and 6 also show percent frequency of radon levels respectively in Ramsar and Tehran. The highest levels were found in some suburbs of Ramsar which have high natural radiation areas. In Ramsar and its suburbs, about 55% of the houses have radon levels below 400 Bq/m^3 and 45% between 400 to 3200 Bq/m^3,

Table 1. Summarized Results of Radon Levels in fours Cities of Iran.

TOWN	PERIOD	NO.OF HOUSES	EXTREME LEVELS (Bq/m^3)	MEAN LEVEL (Bq/m^3)	S.D. (Bq/m^3)	WL (F=0.5)	$\frac{WLM}{q^*}$	$H_e(\frac{mSv}{q^*})$
RAMSAR	Aug-Sep.87 (50 days)	85	low 25%<95 high 3071	578	677	0.078	0.79	4.3
TEHRAN	Feb-May 86 (100 days)	80	low 48%<50 high 635	80	84	0.011	0.11	0.60
BABOLSAR	Feb-May 86 (90 days)	14	low 21%<50 high 163	88	35	0.012	0.12	0.66
GONABAD	Aug-Sep.86 (100 days)	27	low 19%<50 high 156	84	31	0.011	0.11	0.60

q*= quarter



Figs.5 and 6 Frequency (%) of Rn Levels in Houses of Ramsar and Tehran.

while in Tehran about 85% of levels are below 100 Bq/m^3. Although based on this preliminary study, determination of accurate doses can not be achieved, crude estimates of mean effective dose equivalent in the above regions, based on using an equilibrium factor of 0.5, 80% occupancy factor and a dose equivalent conversion factor of 5.5 mSv/WLM (2), were found to be respectively 4.4,0.61, 0.67 and 0.64 mSv/quarter. The results of Ramsar and Gonabad are from summer measurements with houses usually having natural conversion which is expected to be at least two times higher in winter than that of summer.

Gamma dosimetry was also carried out in houses of Ramsar by TLDs using CaF_2:Dy bulbs and film badges; both fixed together with radon dosimeters on the wall. The gamma doses showed extreme values of 0.09 to 90 mGy/q with a mean value of 6.57 mGy/q which showed a good correlation between the two gamma dosimetry systems. A crude estimate of over-all effective dose equivalent for gamma and Rn showed a mean value of 10.97 mSv/q. Especially some houses in Ramsar have very high exposures. For example, one house showed respectively 72 mSv/q (80% occupancy) and 46.8 mSv/q respectively for gamma and Rn with a sum of 118.8 mSv/q or 11.9 rem/q resulting to 475.2 mSv/y or 47.5 rem/y. This is at least 10 times higher than the annual dose limit for radiation workers. Such preliminary results have stimulated further detailed dosimetric studies which is in progress.

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MEASUREMENT AND DOSIMETRY OF INDOOR RADON CONCENTRTIONS IN KUWAIT

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INTRODUCTION

Radon-222 is a radioactive gas that occurs naturally in the earth's surface. Public concern about radon gas and its decay products is growing fast. For most countries, 5-10 percent of the total observed lung cancer incidence is due to radon (1).

Risk from radon and its daughters could be quantified by determining the level of radon concentrations indoors and the factors that influence them. From the measured concentration values, the annual mean radon concentration and its seasonal levels could be established. Population doses and related risk estimates from such doses could be calculated.

Kuwait is situated north-east of the Arabian Peninsula and on the northern coastal area of the Arabian Gulf. All rocks in Kuwait are due to Sediments, containing no trace of volcanic activities, and deficient in their mineral contents. Therefore, the emanation of radon from the soil is expected to be relatively low.

The houses monitored in this study are 5 to 10 years old. Most of them are insulated against the dust and outside heat. In this article we present results of a survey of indoor radon concentrations undertaken in different parts of Kuwait over a period of 15 months. More than 60 houses were monitored with Terradex Track - Etch SF type monitors from Terradex (USA). Seasonal changes were measured by replacing the detectors every 3 months.

METHOD OF MEASUREMENT

Most of the measurements were carried out in residential areas, in addition to some measurements performed in offices and occasionally occupied areas. Within the measuring site, the detectors were mounted at points far enough (about 1.5 meters) from strong air flows. However, due to practical difficulties, many detectors were fixed immediately to the wall without observing the 0.5 meter distance from room boundaries, which would provide even distribution of air flow within the space.

At least one room was monitored in each dwelling. In others, more than one room, situated at different floors within the same dwelling, were monitored. Monitors were placed for four consecutive intervals which lasted 3 months each. Such periods reflect the seasonal changes in Kuwait which is marked by dry and hot summer-autumn. Thus, more time is spent indoors during this period with the use of mechanical ventilation and air - conditioning systems.

About 45% of the monitored locations were ground floor, 29% first floor, 13% second floor, another 13% were third and fourth floors, and only

3% basement. The houses were not particularly energy efficient. They represent conventional living houses.

RESULTS

Table 1 presents the distribution of average radon concentrations obtained for the entire year of the study. The results in Figure 1 are plotted against the commulative percentage of all readings that are below a given concentration. The arithmetic mean of the distribution is 41.3% Bq/m^3 with a standard deviation of 14.9 Bq/m^3 . The largest mean concentration is 103 Bq/m^3 which is less than the 110 Bq/m^3 limit adopted by the U.S. Nuclear Regulatory Commission for continuous exposure in controlled areas.

Table 1: Summary of seasonal changes of radon concentrations taken at various sites

	Autumn	Winter	Spring	Summer
Total number of readings	31	69	69	31
Mean (Bq/m^3)	41	43.5	42.5	41.3
S.D (% of mean)	14.7(36)	18(41)	18.6 (44)	14.5(35)
Range (Bq/m^3)	82 - 18	103 - 16	102 - 8	80 - 12
Percentage deviation from all readings mean (40.8 Bq/m^3)	0.5	6.6	3.7	1.2

Figure 1 shows a deficiency in the low values as compared to those in the high range. The median is 40 Bq/m^3 and the distribution calculated skeweness is + 0.29. Consequently the propability histogram of the values in Fig. 1 will not be symmetrical about the mean, and shall have longer "tail" to the right.

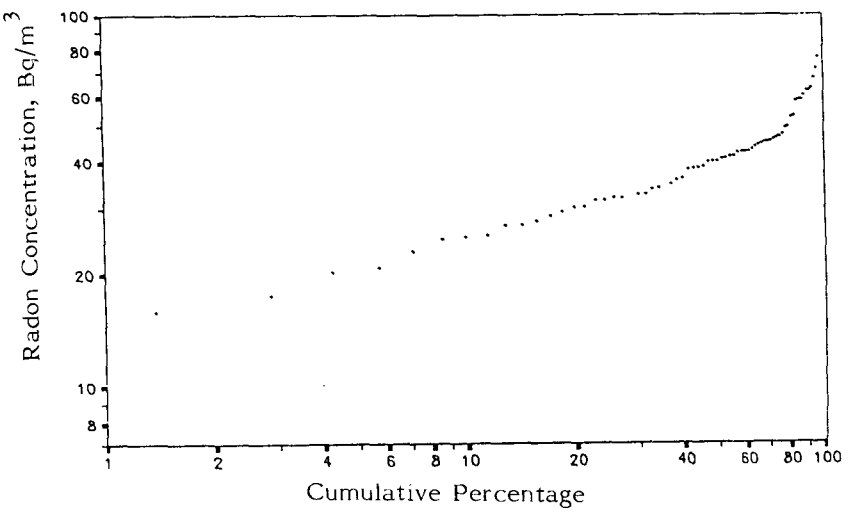

Figure 1 presents the distribution of average radon concentrations during one year. The abscissa shows the cumulative percentage of all readings that are below a given concentration.

SEASONAL RADON CONCENTRATION VARIATIONS

To appreciate the significance of seasonal radon concentraitons seperate calculations performed are summarized in Table 2. Winter and spring mean values appear to be higher than those from summer and autumn. Due to the hot summer-autumn period, dwellings are kept tightly sealed for most of the day. However, houses monitored use forced-air ventilation systems which allow exchange of dwelling air with outside air.

The large spread and variations in the winter readings had influenced the overall mean and standard deviation in the total data distribution.

VARAITION OF RADON CONCENTRATION WITH FLOOR LEVEL

Table 2 presents a summary of the radon concentrations at different floor levels. Most of the readings were in the ground, first, and second floors, with very few basement measurements.

Table 2: Variation of Radon concentrations (Bq/m^3) with floor level

	Basement	Ground Floor	First Floor	Second Floor	Third Floor
Total (*) number of readings in each floor (**)	2, 4	20, 50	14, 30	12, 18	4, 8
Mean (Bq/m^3)					
Summer-Autumn	53.4	47.8	41.9	37.5	26.2
Winter-Spring	55.4	48.9	46.0	44.3	36.5
Percentage deviation from basement mean(***)					
Summer-Autumn	0	-10	-21	-30	-51
Winter-Spring	0	-12	-17	-20	-34
Range (Bq/m^3)					
Summer-Autumn	58-49	82-26	80-28	46-21	34-18
Winter-Spring	64-48	103-21	102-24	68-8	54-17

* Some readings were disregarded for being associated with high standard deviations.

** The two numbers given for each floor are the Summer-Autumn followed by the Winter-Spring reading.

*** The negative sign indicates reduction from the mean.

The results obtained in the basement were the highest with mean concentrations of 53.4% and 55.4% Bq/m^3 for the Summer-Autumn and Winter-Spring seasons, respectively. These levels decrease on upper floors. It is worth noting that the percentage deviations obtained are increasing steadily, with almost the some rate for the floors from ground to second. This rate suddenly becomes higher for the third floor readings obtained for all seasons which possibly indicates additional ventilation due to increased draft at high-rise buildings.

HEALTH EFFECTS FROM EXPOSURE TO INDOOR RADON CONCENTRATIONS

Dosimetric models for inhalation of radon daughters, with different assumptions used, consider the activity distribution in the respiratory tract in steady - state equilibrium taking into account the lung geometry, and the deposition and retention functions (3, 4-6).

The BEIR III report (6) gives an elaborate age dependent risk factors of 10×10^{-6} per WLM.y for age 35-45, 20×10^{-6} per WLM.y for ages 45-65 and 50×10^{-6} per WLM.y for ages above 65. We used the above model to calculate the distribution of lung cancers in Kuwait between the different age groups from a radon exposure of 0.228 WLM per year. The results are tabulated in Table 3 and gives 45 cases per year.

Table 3 : Distribution of lung cancer cases due to indoor radon exposure between the different age groups

Group	Effective Exposure duration, (years)	Effective Rn Exposure, (WLM)	Lung Cancer Risk/yrx10^{-5}	Population** x 10^5	Cases/Year
35-44	29.5	6.73	6.73	2.378	16
45-54	39.5	9.00	18.00	1.197	21
55-64	49.5	11.29	22.58	0.428	10
Above 64*	62	14.13	28.26	0.21	6

*If we consider life expectency at age 65 is about 14 years, the average of this group will be 62.

**Obtained from (10)

Small differences in radon daughter induced cancer appear to exist after long-term follow-up studies conducted between smokers and non-smokers (3,7,8). Earlier studies showed a decrease in the risk ratio between the smoking and non-smoking population (9). Considering the estimated risk in this study, using the UNSCEAR model of $1.2\text{-}2.7 \times 10^{-4}$ per WLM for smokers and the decreased risk ratio (9), the estimated risk for non-smokers in Kuwait would be in the range $0.6\text{-}1.3 \times 10^{-4}$ per WLM. Assuming 20% of the population are smokers yields a reduced total population risk of 1.76×10^{-4} per WLM (30-63 cases). If we take the 1982 observed population risk of 1.76×10^{-4}, the relative risk (Rn daughters/observed) is in the range 0.09-0.2. This means 9% to 20% of the observed lung cancer would propably be due to exposure to indoor radon daughters.

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AN ESTIMATE OF THE NATURAL AND ENHANCED NATURAL RADIATION DOSE IN BELGIUM

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ABSTRACT

The ICRP report 39 has reconsidered in 1983 the dose from natural sources of radiation from 1mSv whole body dose to 2mSv effective dose equivalent.

A Belgian and European research programme has been established in order to analyse the main exposure contribution, due to radon and its decay products, by numerous field investigations.

The annual average radon concentration in Belgian dwellings, mainly due to Rn infiltration from the soil, is 53 Bq/m3. The study of the indoor aerosol and of the behaviour of Rn decay products in indoor air resulted in a dose conversion factor proportional to the radon concentration of 50 (μSv/y)/(Bq/m3). The effective dose equivalent due to radon, 2.1mSv/y, is higher than the ICRP 39 total dose estimate.

The Belgian mean value is of the same order as recent USA estimates.

The other components of natural and enhanced natural radiation exposure are analysed referred to measurements, calculations and to comparisons with other European estimates.

Interest is given to a correction of the cosmic and terrestrial free field measurements taking into account the interaction of roads and buildings.

The different medical components still largely uncertain are discussed, pointing out the importance of considering both organ dose (particularly thyroid) and effective dose concepts. Some population averaged small dose contributions were reconsidered, concerning as well occupational doses as consumer product exposure.

The total average effective dose to the population for the two geologically characteristic regions in Belgium is estimated at 3.8 and 5. mSv.

INDOOR RADON MEASUREMENTS IN THE SHENZHEN REGION, THE PEOPLE'S REPUBLIC OF CHINA

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INTRODUCTION

In recent years there has been a growing concern about radon level indoors. Measurements of indoor ^{222}Rn concentrations of residential buildings in the People's Republic of China are organized. The indoor ^{222}Rn survey program in the Shenzhen region was completed in 1986. The purpose of this paper is to present the results and to discuss the indoor radon variations. The dose estimation is also presented.

METHODOLOGY

Shenzhen is a special economic region of China. It is located between 113°46'E and 114°37'E, 22°27'N and 22°57'N. It faces the Daya Bay on the east and the Pearl River estuary on the west. Hong Kong is on its south. The climate is typically maritime with average annual rainfall of 1948mm. The temperature is between 1.4°C and 36.6°C, no significant seasonal variation.

In order to obtain representative data, many measurements have been made in different types of dwellings and at widely distributed locations. Dwellings investigated were typical in each location. The multi-family dwellings selected were constructed with brick and fabricated concrete blocks. In each apartment there was water supply but no mechanical ventilation. Most of the high-rise buildings were morden hotels, or commercial buildings built with concrete, and morden building materials with central or separate mechanical ventilation system. Most of investigated detached houses were in the old area of the city or in the rural area. They were mainly built of brick or stone with brick or concrete floors. Almost all dwellings measured were less than 5 years old except for some detached houses.

Most of the measurements were conducted with a dual-filter environmental radon monitor, with decay volume of 15 lit.. The lower limits of detection (LLD) were 1.2 Bq m^{-3} for ^{222}Rn. Another grab sampling technique used was scintillation flask radon monitor with a volume of 0.7 lit., the background was 0.04 counts per minute and the LLD was 1.2 Bq m^{-3}. It mainly used for checking and for rural area measurement. We also employed integrated activated carbon detectors for comparison. Its LLD was 5.9 Bq m^{-3} for exposure period of 3 or 4 days (Re86). All these instruments were calibrated with liquid radium sources. The in situ intercomparison showed that the results of these three methods were agreeable within ±24%. By the end of 1986, 222 dwellings were investigated. Since there is no significant reasonal variation, the investigation covers half a year, from March to September.

RESULTS

Radon-222 concentration

Radon-222 concentration have been obtained from 19 multi-family apartments, 50 detached houses and 147 high-rise apartments. The frequency and cumulative frequency distributions of indoor ^{222}Rn are approximately log-normal, though the existence of high ^{222}Rn levels in some dwellings makes the distribution skewed.

The measured ^{222}Rn concentrations, extremes, arithmetic and geometric means in several types of buildings and outdoor are given in Table 1.

Table 1 Measured ^{222}Rn concentration in different types of dwellings in Shenzhen (Bq m^{-3})

Types of dwellings	Number of dwellings	Extreme value	Arithmetic mean	Standard deviation	Geometric mean	Geometric standard deviation
High-rise buildings	147	1.5 - 14.2	31.1	22.9	25.2	1.96
Multi-family dwellings	19	3.33- 52.5	15.5	11.8	11.8	2.16
Detached houses	50	4.07- 54.0	17.8	11.8	14.4	1.93
Basement	6	77.7 -142	108	26.3	106	1.27
Outdoors	10	2.78- 35.9	13.7	11.1	20.4	2.24

Statistical analyses indicate that ^{222}Rn concentrations in detached houses and multi-family apartments are no difference, with arithmetic means of 15.5 Bq m^{-3} and 17.8 Bq m^{-3}, respectively. The average concentration in high-rise buildings is 31.1 Bq m^{-3}, significantly higher than that in the other two types of buildings. One of the probably explanation for the fact is that the air exchange rate is much lower in high-rise buildings as mentioned above. In basement the mean is 108 Bq m^{-3} about 2.5, 5.1 and 6 times higher than in high-rise, detached and multi-family dwellings, respectively.

For comparison, the outdoor ^{222}Rn concentration is also listed in Table 1. The arithmetic mean is 13.7 Bq m^{-3}, much higher than the world average, 5 Bq m^{-3}, given by UNSCEAR (Un82). The measured specific activity of ^{226}Ra in soil of Shanzhen region is, on average, 85 Bq kg^{-1}, much higher than the world average, 25 Bq kg^{-1} (Un82). This may explain the higher outdoor ^{222}Rn level.

Radon-222 daughter concentration

Alpha potential energy concentrations of ^{222}Rn daughters have been measured in 181 dwellings. Their frequency and cumulative frequency distributions are similar to the distributions of indoor ^{222}Rn. Statistical analyses shows that the daughters' concentrations

in detached houses and multi-family apartments are statistically the same with arithmetic mean of 1.02 mWL and 0.81 mWL, respectively. The concentration, however, in high-rise buildings is significantly higher than in detached and multi-family dwellings.

Equilibrium factor

From the measured activity concentrations of polonium-218, lead-214, polonium-214 and ^{222}Rn, equilibrium factors F were calculated. Table 2 listed the equilibrium factors in different types of dwellings as well as outdoors. It indicates that there are no significant differences and 0.20 is representative for all types of dwellings. It is much lower than the UNSCEAR estimated values, 0.5 for indoors and 0.6 for outdoors (Un82). The main reason for the low value might attribute to the high air exchange rate and low aerosol concentration in this subtropical city.

Table 2 Equilibrium factors in different types of dwellings

Types of dwellings	Number of dwellings	Extreme values	Arithmetic mean	Standard deviation
High-rise buildings	105	0.04-0.79	0.22	0.13
Multi-family apartments	18	0.02-0.46	0.16	0.10
Detached houses	21	0.07-0.47	0.22	0.09
Basement	5	0.11-0.19	0.14	0.03
Outdoors	4	0.06-0.25	0.17	0.08

DISCUSSION

Variations of ^{222}Rn concentration with heights above ground: in naturally ventilated buildings, ^{222}Rn concentration is higher on lower floors, then it lows gradually as the altitute goes up. However, in centrally or separately ventilated buildings, this trend is not observed, even in a 53 story building, where the correlation coefficient between radon concentration and floor height is -0.43, it means no significant correlation exists. The variation of ^{222}Rn daughters concentration with height is about the same.

The influence of ventilation: The influence of ventilation on concentrations of indoor ^{222}Rn and its daughters is demonstrated. The concentrations in centrally and separately ventilated buildings are significantly higher than in naturally ventilated. The main reason may probably be the difference in air exchange rate. In fact if we use the measured air exchange rate, ^{226}Ra specific activities in building materials, and an appropriate model (Un86), the radon concentrations predicted are reasonably agreeable to the means

measured. It means that air conditioning may raise indoor ^{222}Rn and its daughters concentrations and the inhancement of indoor air exchange rate is only way to reduce ^{222}Rn concentration indoors.

Dose estimation: If the assumption is made that the radon equilibrium factor is 0.2 and that the UNSCEAR's occupancy factor and effective dose equivalent conversion factors, 0.061 and 0.031 mSv $(Bq\ m^{-3})^{-1}a^{-1}$ for indoor and outdoor are used (Un82), then the absorbed dose to T-B region and P region and effective dose equivalent for different types of dwellings caused by ^{222}Rn daughters can be calculated, as given in Table 3. Calculation indicates that effective dose equivalent caused by ^{222}Rn is about 12 times less than that listed in Table 3.

Table 3 Estimation of annual dose caused by ^{222}Rn daughters in different types of dwellings*

Types of dwelling	EEC concentration ($Bq\ m^{-3}$)	Absorbed dose (uGy)		Effective dose equivalent (mSv)	
		T-B region	P region	Indoor	Indoor plus outdoor
High-rise buildings	6.53	255	32.7	0.40	0.49
Multi-family apartment	3.25	127	16.3	0.2	0.29
Detached houses	3.74	146	18.7	0.23	0.32
Basement	22.7	885	114	1.39	1.48

*The doses are estimated based on arithmetic mean.

It is evident that due to low radon and its daughter concentrations, the population weighted average annual effective dose equivalent is about 0.3 mSv in this subtropical region. It is about one third of the world average estimated by UNSCEAR. For comparison, the effective dose equivalent from external radiation, including cosmic ray, is about 1.1 mSv a^{-1} in this region, roughly 3.7 times that caused by ^{222}Rn and its daughters.

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IS CR39 WORTH THE EFFORT?

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INTRODUCTION

CR39 proton sensitive track detectors were greeted by the radiation protection community at the end of the last decade as a major breakthrough for personnel neutron dosimetry. A number of laboratories eagerly began research on application of CR39 to their dosimetry needs. However, in the last two or three years the enthusiasm has subsided, and many health physicists have stopped working with the material. The number of participants using CR39 in the Oak Ridge National Laboratory Personnel Intercomparison Studies dropped from six in 1985 to three in 1986 [1,2]. On a national level, the Federal Republic of Germany with researchers active in CR39 research recently adopted an albedo system as their national standard [3]. In contrast, the United States Department of Energy (DOE) is supporting development of a CR39 based combination dosimeter to meet Department wide dosimetry needs [4]. The English National Radiological Protection Board (NRPB) now features the use of CR39 in the NRPB PADC(CR39) [5]. There has obviously been a range of experiences with CR39 in the dosimetry community. Why has this been the case, and what is the proper role for CR39 in personnel neutron dosimetry?

DISCUSSION

First, it is essential to understand that the simple term "CR39" does not specify the dosimeter any more than saying "neutron dosimeter." There are a large number of variables involved in using and processing CR39 track detectors, and three of the primary performance characteristics of CR39 based dosimeters (sensitivity, neutron energy response, and background) are highly dependent on the specific values of these variables. The basic detector processing can be done by using either chemical or electrochemical etching (CE or ECE). Although initially performed at room temperature (LTECE) [6,7], ECE can now be done at elevated temperatures (≥50°C, ETECE) [8,9], with the result being a dose equivalent response that is much flatter in the range from 0.1 to 5 MeV. Figure 1 illustrates generic differences in energy response between CE, LTECE and ETECE using typical processing conditions. Even within a particular etching class (ETECE for example), the energy response depends a great deal on the particular parameters used (Fig. 2).

CR39 background and neutron sensitivity depend on the source of track detector material, age, and the full range of processing conditions--etchant type and concentration, etch time, temperature, pre-etch conditions parameters, post-etch parameters, frequency, field strength, and detector thickness. The effect of these conditions can sometimes be profound. In a recent EURADOS-CENDOS intercomparison program, 11 national laboratories submitted CR39 dosimeters for irradiation using 6 monoenergetic sources plus ^{252}Cf [10]. Three laboratories used CE, three used LTECE, four used ETECE and one used two sets of dosimeters--one with CE and the other with ETECE. The background and neutron sensitivity ranges shown in Table 1 clearly reflect the effect of the particular processing protocol on detector performance.

TABLE 1

Backgrounds and Cf Fission Neutron Sensitivities for Participants in the Joint European/USA/Canadian Irradiation Program [10]

Processing Technique Used	Number of Participants	Background Range mSv	^{252}Cf Sensitivity Range $cm^{-2} \cdot mSv^{-1}$
CE	4	0.16 - 1.89	297 - 969
LTECE	3	0.19 - 0.64	43 - 160
ETECE	5	0.05 - 3.03	424 - 721

In addition to the problem of parameter dependent dosimeter response, CR39 development has been inhibited by the lack of high quality, dosimetry-grade material capable of providing truly reproducible results. Issues of manufacturer differences and batch-to-batch variations in surface quality, background and sensitivity are well known. These problems are understandable, however, considering that initially the sources of CR39 were plastics companies that use the material for optical applications which do not require the quality control necessary for dosimetry. It should be remembered that significant materials' problems also plagued dosimetrists during the development of TLD.

A final problem that caused a number of users to become skeptical of CR39 performance was the poor performance exhibited by key commercial suppliers of dosimetry services. Users of these services often found that the results provided by the services were anomolous and inconsistent. In inter-comparisons, data provided by these services may not have been of acceptable quality. Unfortunately, for whatever reason--poor quality material, poor quality control, lack of appreciation for the details of the application, etc.--these experiences gave CR39 a bad image, but were not representative of the true potential of CR39 dosimetry. This potential is illustrated by the quality of results obtained in the European/USA/Canadian Joint Irradiation Program.

Although a number of users have experienced problems with the development and use of CR39, a lot of progress has been made. Improved detector material capable of providing good quality results is gradually becoming available, although the material quality problems are by no means solved. Processing techniques necessary to obtain reproducible results have been identified. Even though different protocols yield different dosimeter characteristics, careful adherence to a given procedure will produce satisfactory results [10]. Commercial suppliers are beginning to take advantage of the experience of national laboratories through technology transfer programs.

An advantage to CR39 is that the track formation properties of CR39 can be used to provide valuable information about the neutron field. For example, the etching parameter dependent variability in detector response can be used to perform simple spectrometry [11]. In fact, recent work suggests that electrochemical etch pit size distributions can be used to characterize neutron spectra for more accurate calibration factor assignment [12].

CLOSING REMARKS

The development problems associated with implementation of CR39 dosimetry have been magnified by the small numbers needed for neutron dosimetry compared with the large number of people monitored world-wide for photons. The cost/benefit ratio for development of CR39 is relatively high. At the same time, development has been slow because the financial support available for research is low. However, in spite of its deficiencies, CR39 is the best fast neutron detector currently available at a reasonable cost for routine personnel dosimetry. Yes, CR39 is worth the effort.

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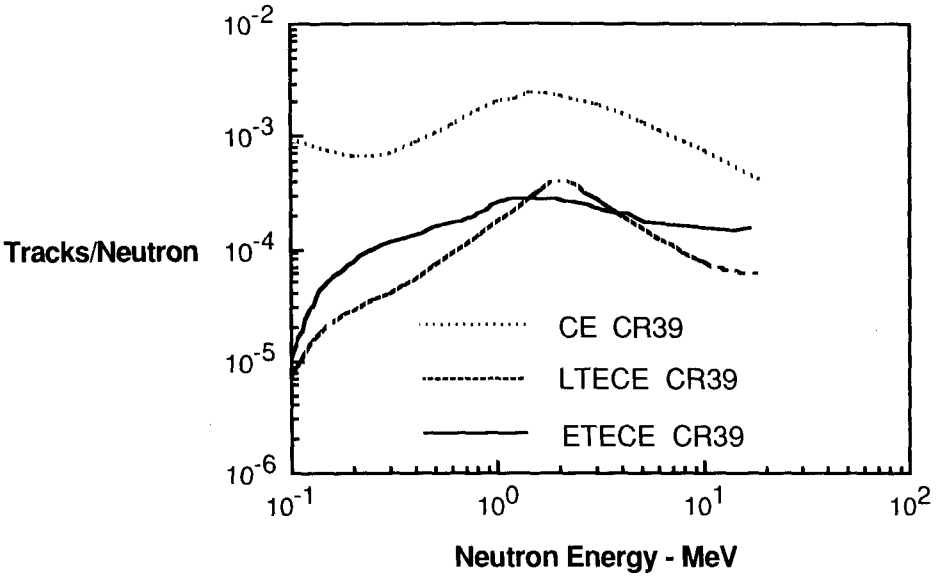
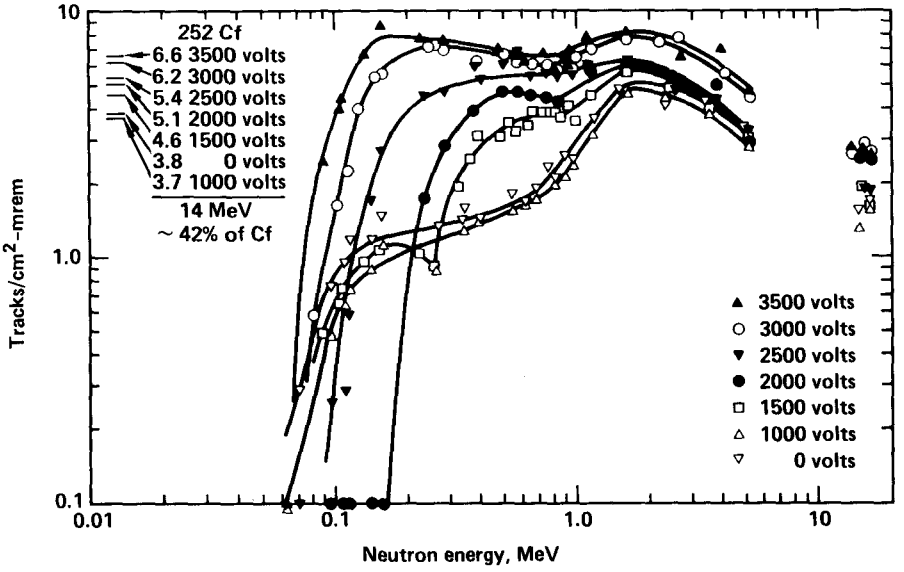


1. Comparison of typical CR39 neutron energy responses using chemical etching (CE), room temperature electrochemical etching (LTECE), and elevated temperature (60°C) electrochemical etching (ETECE).



2. Illustration of the voltage dependent ETECE CR39 neutron energy response for a 5 hour, 60 Hz etch, using a fixed 3000 volt, 2 kHz ECE post etch (approximately 30 minutes).

TWELVE YEARS OF NEUTRON PERSONNEL DOSIMETRY INTERCOMPARISON STUDIES AT OAK RIDGE NATIONAL LABORATORY: WHAT HAVE WE LEARNED?

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INTRODUCTION

To provide an opportunity for dosimetrists to test and calibrate their personnel neutron monitoring systems in a variety of incident radiation fields, the staff of the Dosimetry Applications Research (DOSAR) Facility at the Oak Ridge National Laboratory (ORNL) has conducted personnel dosimetry intercomparison studies (PDIS) periodically since 1974 and annually since 1976 (Si82, Sw87). During these studies, personnel dosimeters are mailed to ORNL, mounted on phantoms and exposed to low-level (less than 15 mSv) dose equivalents in mixed-radiation fields mainly produced using the Health Physics Research Reactor (HPRR) at ORNL (Au65), and then returned to the participants for evaluation. Reported dose equivalents are compared to reference values provided by the DOSAR staff and to results reported by individual organizations which made measurements under identical conditions. These intercomparisons, which require no fee and are open to any organization interested in external personnel dosimetry, have provided more data concerning neutron dosimeter performance characteristics in mixed-radiation fields than any other periodic open test program conducted to date. The following text presents a summary and analysis of neutron dose equivalent measurements reported for the Seventh through Twelfth intercomparisons (1981-1986) using the HPRR as the source of radiation. Particular factors examined include low dose equivalent sensitivity and measurement accuracy for the basic types of neutron personnel dosimeters.

INTERCOMPARISON DATA

A total of 116 different organizations (78 from the United States and 38 from other countries) has participated in the ORNL intercomparisons. These organizations include industrial and government laboratories (40%), nuclear utilities (20%), universities (14%), vendor services (13%), military and regulatory agencies (12%), and hospitals (1%). Participants submitted a total of 5750 personnel dosimetry badges - 4700 were exposed and 1050 were controls - for PDIS 7-12, and 3451 measured neutron dose equivalents were reported for HPRR-only irradiations. Most (about 85%) of the PDIS exposures have used the HPRR with and without spectrum-modifying shields as the source of radiation. About 90% of the HPRR irradiations have been conducted for four different shield conditions: unshielded (bare reactor) and the reactor shielded with 13-cm of steel, 20-cm of concrete, and 12-cm of Lucite. These fields range from a hard, almost U-235 fission neutron spectrum with relatively low thermal fluence and a small gamma component (unshielded HPRR) to a soft, hydrogen-moderated neutron spectrum with a high thermal fluence and

strong gamma component (Lucite-shielded spectrum). The steel- and concrete-shielded spectra in that order are between the unshielded and Lucite-shielded HPRR spectra in terms of increasing thermal component. The analysis presented in the following text is based on data obtained for these four radiation fields since they provided the most measured results and should give the best indication of dosimeter performance.

Although few of the badge designs submitted by different organizations are the same, the basic neutron detection mechanisms can be classified into six categories: direct-interaction thermoluminescent (TLD), TLD-albedo, film, recoil track, fission track, and combination albedo plus recoil track. Direct-interaction TLD's differ from albedos in that direct systems are affected by all incident neutrons including thermals while albedo systems either discriminate against incident thermals by using a thermal neutron absorber or by measuring the incident thermal component separately. Recoil track dosimeters considered in this study consisted of allyl diglycol carbonate (CR-39) plastic and were electrochemically etched to enhance track sizes. Results presented in this report are for those CR-39 systems that were commercially available between 1981 and 1986 and do not include any data for recently developed dosimetry-grade CR-39 and improved etching techniques (Ha87).

LOW DOSE EQUIVALENT SENSITIVITY

To determine the sensitivity of the basic neutron personnel dosimetry systems at low dose equivalents, irradiations were conducted for the four primary HPRR spectra at neutron dose equivalents of about 0.5 mSv (50 mrem). Results of these studies showed that participants who used TLD-based systems had fewer problems obtaining measurable indication (results above zero or the minimum detectable) of neutron exposure than did those who used track-based systems. Only approximately 4% of all results for albedo systems were reported as zero or below minimum detectable for dose equivalents of about 0.5 mSv. Combination albedo-track and direct-interaction dosimeters exhibited slightly greater difficulty providing measurable indication of neutron exposure at this level with about 9% and 17%, respectively, of the results reported as zero. Track-based systems had much more difficulty providing indication of neutron exposure in that approximately 25% of all fission track results, 29% of all reported film results, and 47% of all CR-39 results were reported as zero or below minimum detectable. The next lowest neutron dose equivalent level considered in the ORNL intercomparisons was about 1.5 mSv (150 mrem) which is the lowest limit specified for neutron dosimetry accreditation testing (AN83). None of the basic dosimeter types exhibited any difficulty providing measurable indication of neutron exposure at dose equivalent levels above this value.

MEASUREMENT ACCURACY

The quantity of most concern to those involved in applied dosimetry or accreditation program testing is measurement accuracy. In this analysis, accuracy is indicated by the mean normalized dose equivalent

which is the average of all measured-divided-by-reference dose equivalents for a particular incident spectrum reported for PDIS 7-12. Measurement accuracy for albedo, direct-interaction TLD, film, and recoil track neutron dosimeters as a function of the four most used HPRR spectra (unshielded, steel-shielded, concrete-shielded, and Lucite-shielded) are shown in the figure below for neutron dose equivalents above 1.5 mSv. The spectra shown on the horizontal axis are ordered such that the neutron energies get increasingly softer (e.g., higher thermal neutron component) going from left to right. Points shown in the figure are accuracy values and represent the average normalized results reported for PDIS 7-12 by all participants who used a particular dosimeter type. Fission track systems are not included in the figure since only a few measurements were reported for this dosimeter type. Combination albedo-track dosimeters are also not included because the variation in accuracy as a function of incident spectrum is almost identical with that observed for recoil track systems.

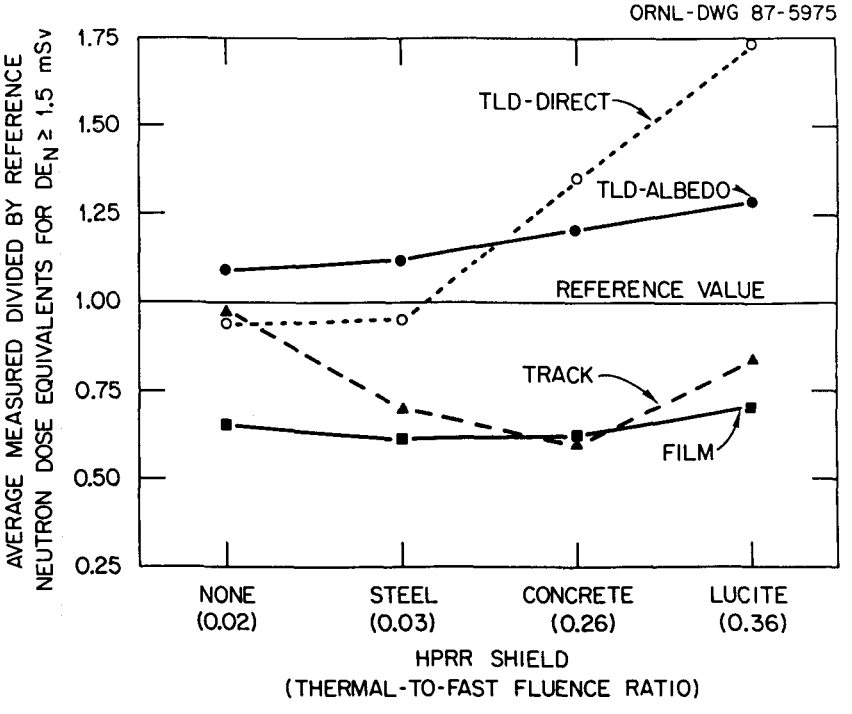

Figure 1. Neutron dosimeter accuracy as a function of incident spectrum for dose equivalents greater than 1.5 mSv.

The figure shows that direct-interaction TLD's, albedo systems, and recoil track dosimeters provide average results within about 10% of reference values for the hardest (unshielded) HPRR spectrum. Since most of these dosimeters were calibrated with hard energy spectra (e.g., unmoderated californium or PuBe), this performance is expected. As the spectra become softer, TLD-based systems tend to overestimate reference dose equivalents with direct-interaction systems overresponding more

than albedos with increasing spectrum softness. Dosimeter types with threshold detection energies such as recoil track and film underestimate reference dose equivalents for moderated spectra by more than 20%. Film neutron dosimeters, which were generally calibrated in spectra much harder than the unshielded HPRR, also underestimate reference values for the unmoderated reactor spectrum. These observed variations are qualitatively the same as would be expected if the dosimeters were calibrated to hard energy spectra on the order of the unshielded HPRR and no corrections were made to dosimeter responses to account for energy response characteristics and differences between incident and calibration spectra. This suggests that many PDIS participants are not making corrections to dosimeter responses or that they are using inadequate corrections. Despite this inferred problem, average reported results for all basic dosimeter types shown in the figure are within 40% of reference values with the exception of the direct-interaction TLD data for the softest incident spectrum.

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ROUTINE NEUTRON MONITORING IN THE FEDERAL REPUBLIC OF GERMANY USING A TWO-COMPONENT ALBEDO DOSEMETER SYSTEM

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INTRODUCTION

In personnel monitoring the NTA film was the only applied neutron detector for more than 25 years (1). However, the energy threshold at about 1 MeV, the relatively high fading and the gamma sensitivity allow only a useful application in neutron fields near accelerators. When used in personnel monitoring at reactors and in the fuel element cycle, the NTA film never indicated neutron exposures. This „zero reading" of the NTA film were sometimes taken as an alibi of negligible neutron dose contributions.

Since ten years CR 39 track etch detectors have been improved in the etching technique and dosimetric properties. Commercially available materials, however, have not yet reached the state that materials of constant quality and uniformity in response and background are available for a large scale routine application (2). The energy threshold at about 100 keV, on the other hand, calls for a combination with an albedo detector.

Energy dependence prevented a large scale routine application of albedo dosemeters in the past. This has been overcome by using location and facility dependent calibration factors. In the stray neutron fields of interest a large-scale routine monitoring (3, 4) has been recently realized with a two-component albedo dosemeter by classifying the variety of neutron fields in four types of application areas. Well-known field calibration factors for each type of application area may be applied without the need of knowing more details about the local situation such as the neutron energy spectrum, the shielding and the distance to the source. The calibration factors which may significantly vary with neutron energy in the four areas of application are transferable to other types of TLD systems using the two-component dosemeter capsule. In routine monitoring therefore is in general no need to perform field calibrations at the actual neutron facility of interest.

DOSEMETER SYSTEM

The two-component albedo dosemeter consists of a boron-loaded plastic encapsulation with an albedo neutron detector (i) facing the body of the wearer and an additional thermal neutron detector (a) for field neutrons. For special applications only track etch detectors may be incorporated. The universal albedo dosemeter is applicable for automatic TLD read-out systems of the manufacturers Alnor, Harshaw, Panasonic and Vinten (Fig. 1).

The TLD systems applied make use of TLD cards with four detectors, for instance two pairs of TLD600/700 detectors. The TLD700 detector reading is used to subtract the gamma dose component from the TLD600 reading. An empirical calibration factor N_N corrects for neutron energy dependence of the albedo detector reading $M_n(i)$. The neutron calibration factor $N_N = H_R/M_n(i)$ has been derived from field calibrations using a 30 cm polyethylene sphere as a phantom and reference dosemeter for the measurement of H_R using similar TL detectors in the center of the sphere. In detail, N_N is the product of the relative neutron calibration factor n_N normalized to the calibration factor in a Cf-252 neutron reference field, the relative neutron calibration factor n_{nr} in the neutron reference field normalized to a photon calibration, and the gamma calibration factor N_γ for the TLD reader given by the equation

$$N_N = n_N \cdot n_{nr} \cdot N_\gamma = \left(\frac{N_N}{N_{nr}}\right) \cdot \left(\frac{N_{nr}}{N_\gamma}\right) \cdot N_\gamma = \frac{1}{R_N} \qquad (1)$$

The results of field calibrations in all available types of neutron fields (3) have shown that the variety of neutron fields can be reduced to four typical types of neutron fields in the application areas N1-N4 shown in Fig. 3. Within one area the relative neutron response R_n/R_{nr} can be estimated with a sufficiently low scatter of about a factor of two which in general is acceptable in personnel dosimetry. At least in N1 and N2, constant response values may be used . In N3 and N4, the relative albedo response R_N/R_{nr} increases with the reading ratio of the albedo detector (i) and the thermal neutron detector (a). This ratio and the calibration curve may be used to estimate the actual albedo response in personnel monitoring. Behind heavy shieldings fission neutrons as well as 14 MeV neutrons may result in the same albedo response (area N1). There is the strong decision in neutron monitoring to use the albedo dosemeter in one field of application only.

DOSIMETRIC ASPECTS OF ALBEDO NEUTRON DOSIMETRY

In routine monitoring the two-component albedo dosemeter turned out to be of sufficiently low energy and angular dependence. On the basis of calibration factors derived from Fig. 3, routine application does not need further investigation of workplace dependent calibration factors. The calibration factors are applicable in similar stray neutron fields and also transferable to other TLD systems as seen in Tab. 1 for different neutron fields. Only in cases of higher neutron exposures it may be useful to improve the dose estimation by an actual field calibration.

Within personnel monitoring the albedo dosemeter should indicate the angular dependent dose quantity H'(10), in particular for an radiation incidence from the front half space. Also in the extreme case of lateral exposures the albedo detector still overestimates H'(10). On the basis of field calibrations, the uncertainty of the albedo dosemeter reading due to energy dependence is comparable with that of the reference dosemeter. With respect to other „rem-meters", the sphere of 30 cm diam. offers the lowest energy dependence, i.e. overestimation in stray neutron fields (4). Field calibration factors are conservative with respect to the dose estimation within personnel monitoring, because the reference dosemeter overestimates the dose equivalent H'(10) for frontal exposures below 1 MeV due to its energy dependence and for other directions of the radiation incidence due to its isotropic response.

However, the application of albedo dosemeters in routine personnel dosimetry requires
(a) the use of the personal dosemeter in only one field of application,
(b) the need to apply once established field calibration factors for each type of application area N1 - N4, which are applicable for TLD systems from different manufacturers,
(c) the side correct attachment of the dosemeter encapsulation on the body.

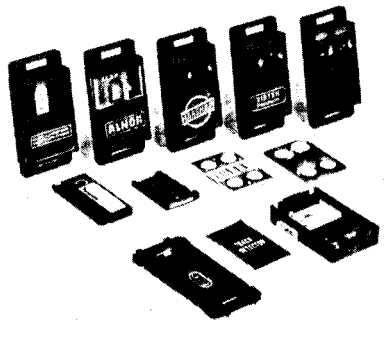

Fig. 1 Universal albedo dosemeter type Alnor, Harshaw, Panasonic, Vinten

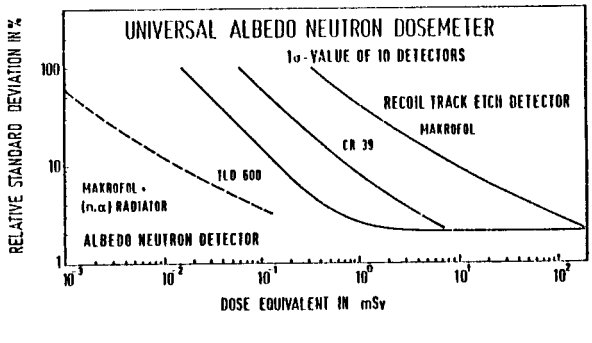

Fig. 2 Random uncertainty of albedo TL and track etch detectors

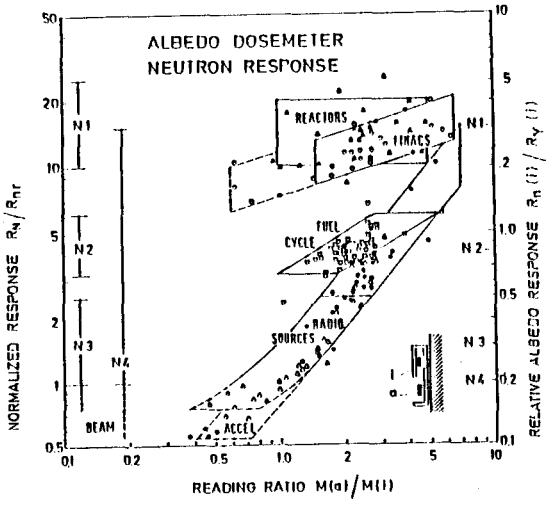

Fig. 3 Relative neutron response R_N of the albedo TLD system Alnor normalized to R_{nr} of a Cf-252 scatter free reference field vs. the reading ratio $M_n(a)/M_n(i)$ for different areas N1 to N4

DOSIMETRIC PROPERTIES AND APPLICATION

As shown in Tab. 2, albedo dosemeters covers a dose range which is sufficient for routine and accident exposures in mixed photon/neutron fields. In the case of criticality, the calibration factors are well-known, the component of thermal neutrons are identified separately and the reading ratio M(a)/M(i) indicates changes in the shielding/scattering conditions during the exposure. These results have to be completed by other measurements such as Na-24 in blood, P-32 in hair, by the chromosome aberration technique, for instance

The effect of body-to-detector distance is comparable with other photon dosemeters. With respect to gamma discrimination the dose equivalent of neutrons and gamma rays in total may be estimated accurately enough.

Tab. 1 Relative neutron response of different albedo TLD systems in stray neutron fields

RELATIVE NEUTRON RESPONSE $R_n(i)/R_n(i,Cf\text{-}252)$					
	ALNOR (KFK)	HARSHAW (MPA)	PANASONIC (GSF)	VINTEN (KFK)	$R_{mean} \pm \Delta R$
Cf-252[1]	1	1	1	1	1
Reactor beam	4.0	3.94	4.3	3.55	3.83 ± 12%
Power plant	13.3	14.8	14.9	14.3	14.0 ± 13%
Fuel element storage[2]	3.2 ± 18%	4.25 ± 21 %	3.1 ± 21 %	-	3.52 ± 15 %
RELATIVE ANGULAR RESPONSE Rn(i,90°)/Rn(i,0°)					
Cf-252[1]	0.62	0.63	0.59	0.55	0.60 ± 6%
Power plant	0.56	0.52	0.53	0.54	0.54 ± 4%

1) Stray neutron field of source in 2.5 m distance and 1.25 m above floor. Field calibration includes room scattered neutrons

2) Mean value and maximum scatter for 12 to 20 calibration results

In special cases, the albedo dosemeter allows the combination with track etch detectors in order to detect separately neutrons above 5 MeV. The TL detectors, on the other hand, may be replaced by gamma insensitive $^{10}B(n,\alpha)$ track etch detectors resulting in a lower detection limit (Fig. 2).

Tab. 2 Dosimetric properties of albedo dosemeters in stray neutron fields N1 to N4

Dose range photons/neutrons (N1-N4)	0.03 mSv to 10 Sv/(0.02-0.10) mSv to (3-50) Sv
Random uncertainty/gamma discrimination	$\sigma_n < 5$ % for 10 mSv/$\sigma_{n+\gamma} < 20$ % for $H_\gamma/H_n \leq 10$
Indication of H'(10)	conservative in front half space (including 90°)
Energy and angular response R_n (front half space)	within factor 2 N1, N2: constant R_n values N3, N4: calibration curve (see Fig. 3)
Systematical uncertainty of field calibration	conservative albedo response $R_n(i)$, overestimation ≤ 50 % due to energy dependence of 30 cm sphere
Detector-body distance effect	within ± 10 % for distance up to 4 cm
Scatter in the response curves (Table 1)	within $\pm$ 15 % for TLD systems Alnor/Harshaw/Panasonic/Vinten

Within a governmental supported research program four official dosemeter services make use of different automatic TLD systems of the manufacturers Alnor, Harshaw, Panasonic and Vinten. As the result of the joint calibration program and the long-term application in routine monitoring (4), the albedo dosimetry systems have been found to be adequate dosemeters for the estimation of the neutron and photon dose equivalent. Within routine monitoring the lower detection limit and the uncertainty of neutron dose measurement have been found to be comparable with those of photon dosemeters. Since 1986, the universal two-component albedo neutron dosemeter is the official neutron dosemeter in the Federal Republic of Germany which replaces the neutron film dosemeter (NTA film) used so far. Neutron albedo dosemeters are applied routinely if the neutron dose rate at workplaces exceeds 20 % of the photon dose rate and/or if there is a risk for an overexposure due to neutrons.

ACKNOWLEDGEMENT

The authors would like to thank Mr. David, Mr. Nahrstedt and Dr. Schraube, GSF Neuherberg, Mr. Morfeld, Strahlenmeßstelle Berlin, and Mr. Ritzenhoff, MPA Dortmund-Aplerbeck, Dr. Alberts and Mr. Kluge, PTB Braunschweig, and Mr. Schwarz, KfK Karlsruhe, who participated in the joint program of the governmental services.

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APPLICATION OF A TISSUE EQUIVALENT PROPORTIONAL COUNTER (TEPC) SYSTEM IN DIFFERENT RADIATION FIELDS

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INTRODUCTION

The TEPC system used at EIR was originally a copy of the Jülich counter developed by Booz, Schmitz et al. [1]. However, calculations by Morstin et al. [2], concerning tissue equivalence at E_n > 20 MeV led us to modify the counter, adding a few secondary improvements in the process.
The counter is now made up of the following: Starting at the center, we have the anode wire of 0.1 mm dia. stainless steel, surrounded by the tissue-equivalent gas volume of 70 mm dia. and 70 mm length (a square cylinder) at 13 1/2 mbar, contained in turn in the 1 mm thick A-150 (Skonka plastic) cathode. Around this there is the modified outer shell: Instead of the 20 mm P.E. shell of the Jülich design we now have an 8 mm Plexiglass shell, and, in order to have a metallic vacuum barrier (since all plastics allow oxygen and water, both very detrimental to T.E. gas mixtures, to diffuse to some extent), an Al skin of 0.8 mm thickness.
Attached to the base-plate of the counter (to keep the leads short and noise low) there is the preamplifier which feeds into the main, logarithmic amplifier. The log. amplifier was purchased from KFA Jülich. The pulses, now of comfortable signal strength, can be routed through coaxial cables of nearly arbitrary length to the multichannel analyzer, where they are stored and processed to yield dose and D.E. distribution, total dose and D.E. as well as the parts thereof pertaining to gamma and neutron radiation, and, finally, the total, γ and neutron quality factors.
The reader wishing to see the system described in more detail will find this in [3].
Since the system was put into operation at the beginning of 1987, we have first spent some time calibrating it in Cs-137 gamma and Am-Be neutron fields of known strength (as well as getting acquainted with it and perfecting the evaluation), then proceeded to measurements in nuclear power stations in Switzerland, both PWR's (Beznau, Gösgen) and BWR's (Leibstadt, Mühleberg). The results from Gösgen and Leibstadt as well as a measurement performed at SIN are described in [3].
In the present paper, however, we are going to describe a second measurement at SIN and several measurements taken at the radiation therapy facilities of the University Hospital in Basle, Switzerland.

SIN

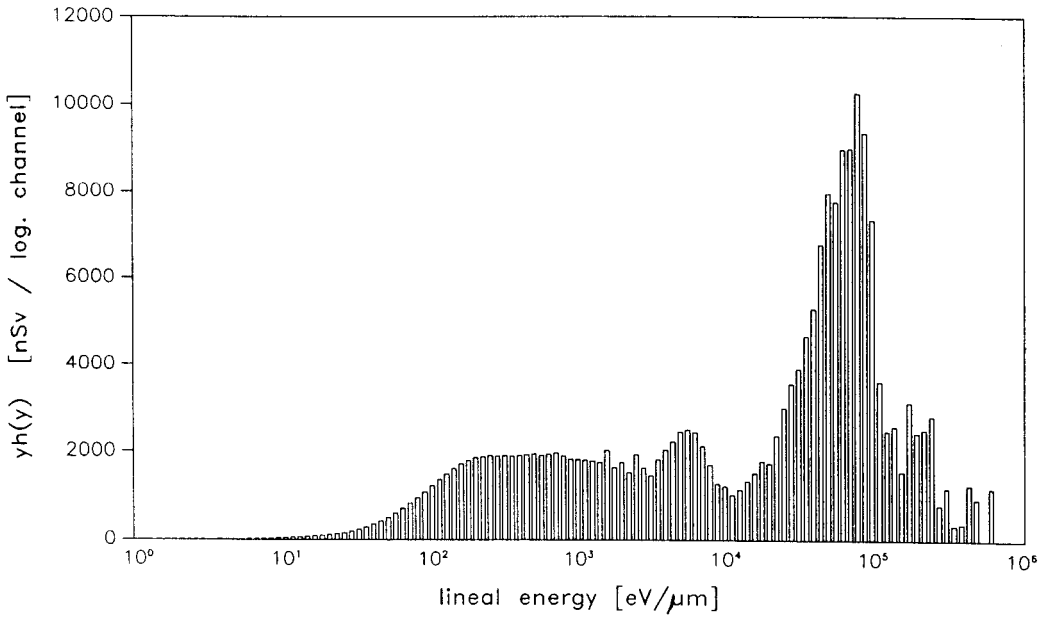
SIN, the Swiss Institute for Nuclear Research, is a 600 MeV proton accelerator located at Villigen (AG), Switzerland. It harbours the so-called Piotron, a radiation therapy facility using, as the name implies, pions. These are produced by a 16 μA beam of 600 MeV protons impinging on a Be target. Thus, all sorts of particles are produced, and those of the right mass and momentum are selected by two sets of superconducting magnets and focussed on the tumour to be treated, the rest being absorbed by the very massive shielding (4.5 m of concrete and shot concrete). Also, the beam tunnel under the shielding has to

be ventilated to evacuate the A-41 formed by n capture in Argon lest it accumulate in intolerable concentrations in the building. This ventilation shaft is, of course, a breach in the shielding, and it is on top of the elbow of that ventilation shaft, quite close to its emergence from the shielding, that we have put our counter. There, one expects a reflection or labyrinth spectrum of neutrons, together with a few penetration neutrons and gammas arising either directly from the target or from inelastic scattering and capture of neutrons. The measured spectrum is shown in Fig. 1 for dose equivalent distribution. The γ, n, and total dose and D.E., the derived quality factors and $\dot{H}_n$ / $\dot{H}_\gamma$ ratio are listed in Table 1.

Table 1: Results of the measurements at SIN and Basle University Hospital. At pos. A and C the detector was covered by 5 cm of lead (see text).

Location	$\dot{D}_\gamma$ µGy/h	$\dot{H}_\gamma$ µSv/h	$\bar{Q}_\gamma$	$\dot{D}_n$ µGy/h	$\dot{H}_n$ µSv/h	$\bar{Q}_n$	$\dot{D}_T$ µGy/h	$\dot{H}_T$ µSv/h	$\bar{Q}_T$	$\dot{H}_n / \dot{H}_\gamma$	Rem-counter $\dot{H}_n$ µSv/h
SIN, Piotron Airshaft	32.7	37.4	1.14	4.93	53.6	10.88	37.6	91.0	2.41	1.43	-
Hospital Basel A	78.8	109.1	1.39	56.5	563.4	9.98	135.2	672.5	4.97	5.16	1600
B	87.9	105.0	1.19	23.9	247.4	9.24	111.8	352.4	3.15	2.36	900
C	61.4	82.5	1.34	46.8	461.0	9.84	108.2	543.5	5.02	5.59	1200
D	27.2	32.2	1.18	5.06	48.5	9.60	32.2	80.7	2.50	1.51	100-200
E	4.8	5.7	1.18	0.95	9.1	9.54	5.7	14.8	2.57	1.60	10- 20
F	0.15	0.23	1.50	<0.03	<0.3	-	-	-	-	-	<1
G	1.17	1.33	1.14	<0.20	<2	-	-	-	-	-	<1

Fig. 1: Dose equivalent spectrum on Piotron ventilation shaft



UNIVERSITY HOSPITAL BASLE

As the central hospital serving nearly half a million inhabitants, the University Hospital of Basle has several radiation treatment machines, the most important of these being the Asklepitron, a 45 MeV Betatron manufactured by Brown Boveri Co., intended for high energy photon therapy and capable of delivering 0.7 Gy/min at 1.10 m F.S.D. with a 10 x 10 cm field at 5 cm depth in a water phantom.
Measurements have been made at the points designated A to G in Fig. 2, which shows the general layout of the treatment room.

Fig. 2: Ground plan of the Asklepitron room

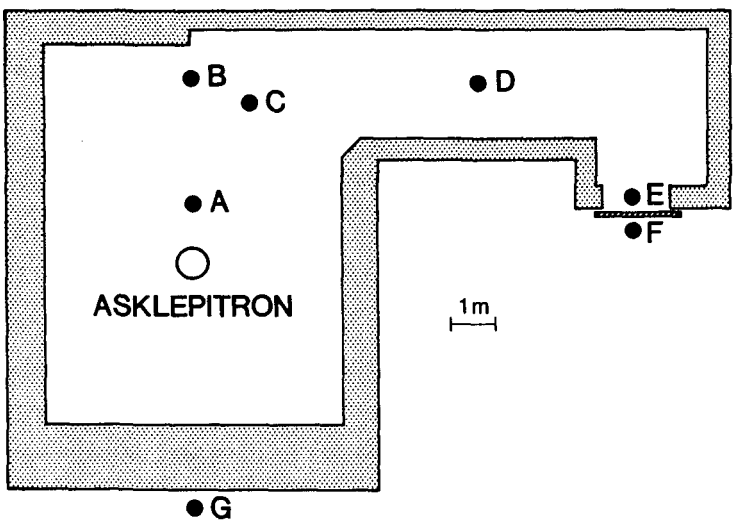
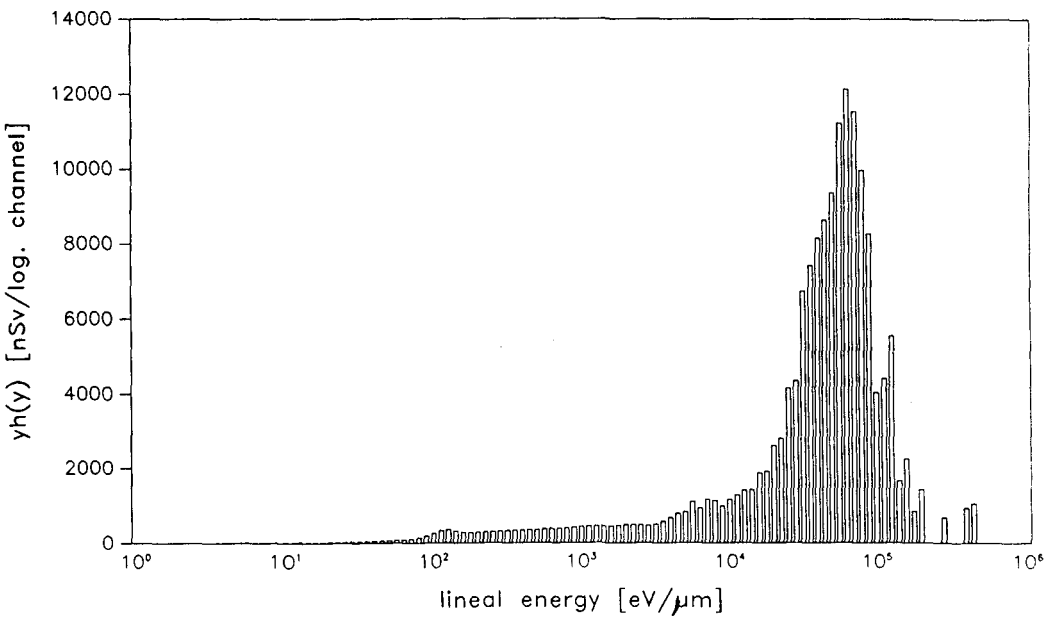


Fig. 3: Dose equivalent spectrum at point A in Asklepitron room



The D.E. spectrum shown in Fig. 3 was obtained at point A, with the counter shielded by 5 cm Pb on all sides. This is a necessary precaution when working that close to the treatment head, for it cuts out most of the photons which, being so numerous, would lead to intolerable pile-up in a pulsed field with duty cycle ≈ 0.001. With the treatment head pointing downwards, as was the case for measurements A, C, D, E and F, a 30 x 30 x 15 cm Plexiglass phantom was placed in the beam at 1.10 m F.S.D. and was the source of much of the stray radiation, especially neutrons, the remainder coming from the floor below. In the case of the measurements B and G, the treatment head pointed horizontally towards the concrete wall at the bottom of Fig. 2, making that wall the main source of stray radiation (some of course also penetrate the shielding). The dose rates, D.E. rates etc. of the measurements A - G are given in Table 1 as well as the results of remcounter measurements performed alongside our counter; they were measured by H.W. Nemec, Basle University Hospital.
The very noticeable discrepancy between the neutron D.E.'s as measured by remcounter and TEPC (the latter being a factor of three to five lower than the former) begs for an explanation:
Both remcounter and TEPC have a weak spot at intermediate energies, this being worst at about 100 keV for the TEPC (underresponding by a factor of about two to three, depending on outer wall thickness and material) and around 200 - 300 keV for the remcounter (overresponding by about a factor of two), so as to lead one to jest about taking the geometric mean as the "true" value. The behaviour of the TEPC improves as the simulated diameter is reduced, but we already use 1 µm simulated diameter, 0.5 µm leading to frequent electric breakdown.

OUTLOOK

We hope to have convinced our dear reader that the TEPC is a viable technique for measuring dose and D.E. in mixed radiation fields, which has by now evolved into the early operational stage. Pacific Northwest Laboratories and one or two other institutions report having developed compact, rugged instruments designed either to be worn on the belt as a personal dosemeter or in a portable version, as a latter-day "true" remcounter. Ours is a bit bulkier and needs mains power, which, however, is nearly always available at neutron installations. We hope to use our instrument for further measurements at medical installations, and at SIN, whose exotic radiation field makes the traditional techniques questionable. This would eventually result in a radiation survey with TEPC's of SIN, especially of its planned additions.

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NUCLEAR ACCIDENT DOSIMETRY FACILITIES AND RESEARCH & DEVELOPMENT TO IMPROVE MEASUREMENTS AND INTERPRETATION

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INTRODUCTION

Nuclear accidents involving an uncontrolled criticality excursion have been very rare in the nuclear industry but even so it is necessary to provide alarms for warning of such events and personal dosemeters for measuring doses to those in the vicinity of such an event. A reappraisal of past criticality accidents and more recent experimental studies of slow, delayed critical, excursions in liquid systems have focussed attention on the need to detect this type of excursion as well as fast transient ones[1]. This need is reflected in the new ISO[2] and IEC[3] recommendations for alarm systems. Personal dosimetry for nuclear accidents was the subject of intense activity by the IAEA in the 1970s which culminated in the publication of a comprehensive manual[4] (with an appendix summarising the 4 intercomparisons). The IAEA also provided a compendium[5] of neutron leakage spectra from critical assemblies for evaluating the doses for such accidents. The UK Ionising Radiation Regulations 1985[6] and the associated Code of Practice[7] require approval for an accident dosimetry service. In order to obtain approval and to minimise the administrative costs, we have set up a users group to: provide traceability of dosimetry techniques to national standards; provide irradiated dosemeters for exercises; conduct any necessary R&D; and provide liaison with international organisations.

This paper will include a discussion of the requirements for alarm systems and new developments in nuclear accident dosimetry. Also indications will be given as to future developments in computer control of counting systems and the provision of an expert system to interpret the measurements.

CRITICALITY DETECTION AND ALARM SYSTEMS

The design criteria and principles were produced by Delafield and Clifton[1] as a major development of earlier work by Aspinall and Daniels[8] incorporating the latest methods of detection (rate-of-change of dose-rate and neutron detectors) and knowledge of a wider range of criticality excursions (slow delayed to fast transient). The criteria have now been incorporated into an Atomic Energy Code of Practice[9], which is at the moment provisional to allow experience on its implementation to be included.

The characteristics of a criticality incident are described in terms of the minimum incident of concern and the radiation field. Consideration is given to: criteria for the threshold for detection; methods of detection; selection and siting of detectors; design principles for the alarm systems; testing and post-alarm procedures. For integrating detectors, it is recommended that the threshold of detection for solution systems of plutonium and uranium should be equivalent to a yield of 10^{15} fissions occurring over any duration between 1 ms and 1 minute. For metal or other systems for which it can confidently be judged that the probability is acceptably low that there are any mechanisms which could lead to a slow delayed criticality excursion, the upper limit of duration may be relaxed down to 5 s. In adopting a threshold yield for detection, the corresponding maximum (n+γ) dose to personnel should be considered. In particular, where personnel are working

close to an assembly, eg at a glove box, then a tighter threshold of detection of 10^{14} fissions within 1 ms to 1 min (solutions) or 1 to 500 ms (metal) should be used. The time to alarm from passing the detector threshold to the alarm system reading full sound should be not more than 200 ms. For more detail of the recommendations reference should be made to the full report[1].

The code of practice provides a step by step approach to the design of a complete system from the plant assessment to the provision of detectors, alarms and evacuation procedures. It includes:

(a) a review of the criteria and principles;
(b) plant assessment leading to the required characteristics for detection and warning;
(c) detection criteria including thresholds;
(d) detector siting to allow for shielding, background radiation, cabling and maintenance;
(e) types of alarm and warnings, viz audible and visible, and their positioning;
(f) characteristics including radiation response, time to alarm, reliability & spurious, alarms, maintainability, fault diagnosis and power supplies;
(g) installation and testing;
(h) management procedures in providing warning notices, evacuation procedures (eg routes), post-incident control and practical exercises.

The code is produced to aid designers, operators, plant managers and health physicists to ensure that proper design principles are implemented and maintained to adequate standards.

PERSONNEL NUCLEAR ACCIDENT DOSIMETRY

In response to the regulatory requirements in the UK[6], Harwell set up a Nuclear Accident Dosimetry Users Group (NADUG) in 1986 to:

(i) provide assistance to member establishments with the maintenance of their approvals for nuclear accident dosimetry services with the HSE;
(ii) maintain UK expertise in nuclear accident dosimetry in order to provide consultancy and immediate assistance to a member establishment in the event of a criticality accident;
(iii) undertake a modest programme of research & development to be steered by the Group members.

Harwell has more than 20 years experience in the field including taking part in the first three IAEA intercomparisons and organising the fourth. In addition to regular steering committee meetings, two successful training courses have been organised to give members experience of evaluating doses to personnel from theoretical criticality accidents.

The general principles, methods of dose assessment and interpretation employed in nuclear accident dosimetry are set out in a three part manual prepared in 1973[10]. Neutron dose is determined by simple measuring techniques which involve the counting of the activation in: gold (thermal and intermediate-energy neutrons); sulphur (fast neutrons); indium (fast neutrons); body sodium (total neutron fluence). A thermoluminscent dosemeter is used to measure gamma-ray dose. Indium ($^{116}In^{m}$) activation in the criticality dosemeter, or film dosemeter or the site pass is used as an exposure indicator to identify those who have received the highest dose for immediate dose assessment. Following counting of the gold, sulphur, indium

($^{115}In^{m}$) and body sodium, an evaluation procedure is followed to give successively improved estimates of the neutron dose received. The total (n+γ) dose is then used to advise the doctors treating the patient as to who may require special hospitalisation. A full description of the system is given in the manuals[10] and an up-to-date summary by Delafield[11].

R&D in nuclear accident dosimetry for NADUG has concentrated on 3 main areas:

(i) development of a Mark IV Criticality Dosemeter containing a silicon diode[12] to provide an immediate measure of neutron dose above 200 keV (this project is well advanced to the final design stage before production);
(ii) reassessment of the interpretation of body sodium measurements (this project is complete);
(iii) revision of the assessment procedures for AERE/NRPB film dosemeter (this project is complete).

In the service area, NADUG has provided a revision of the assessment procedures in SI units which incorporates the latest information from the R&D; sets of traceable standard sources for calibrating counters for gold, sulphur, indium and sodium; a review of sites' instructions for use in an emergency. Future work will be discussed below but in the 18 months of operation, NADUG has demonstrated that for a small annual outlay (£5k) by up to 10 organisations it is possible to provide a service and maintain a reasonable level of R&D in order to improve and simplify its operation.

FUTURE DEVELOPMENTS

New systems for criticality incident detection are being developed to meet the new code of practice[9]. These include the use of neutron detectors for operation in areas containing high transient gamma-ray dose rates.

Silicon diodes could make a considerable contribution to simplifying the immediate determination of the neutron dose after a criticality accident and in improving the accuracy of dose assessment especially for well-moderated leakage spectra. Gold, sulphur, indium and sodium measurements will still be required to provide neutron spectral information for estimating organ doses and dose equivalent information for record purposes. The biggest impact will come from the use of computers. Computerisation of the system is being introduced gradually. Firstly, interactive programs have been written to facilitate the routine testing of the system and undertake a first stage evaluation of the counts from the activation detectors. Secondly, interactive programs will be provided to directly control the counters. Finally the aim is to move towards an expert system which will compare the evaluation with data from the compendium of spectra[5] and prompt for gamma-ray doses, orientation of the man, type of event, shielding, distance, etc. This third step is important in order to retain the expertise built up over the years and to reduce training costs.

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CALCULATION METHODS FOR DETERMINING DOSE EQUIVALENT

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INTRODUCTION

A series of calculations of neutron fluence as a function of energy in an anthropomorphic phantom was performed to develop a system for determining effective dose equivalent for external radiation sources. Critical organ dose equivalents are calculated and effective dose equivalents are determined using ICRP-26 [1] methods. Quality factors based on both present definitions and ICRP-40 definitions are used in the analysis. The results of these calculations are presented and discussed.

The effective dose equivalent determined using ICRP-26 methods is significantly smaller than the dose equivalent determined by traditional methods. No existing personnel dosimeter or health physics instrument can determine effective dose equivalent. At the present time, the conversion of dosimeter response to dose equivalent is based on calculations for maximal or "cap" values using homogeneous spherical or cylindrical phantoms. The evaluated dose equivalent is, therefore, a poor approximation of the effective dose equivalent as defined by ICRP Publication 26.

DISCUSSION

Preliminary work was started in 1987 to develop methods to calculate and possibly experimentally determine effective dose equivalent. This work consisted of modifying the powerful computer code, Monte Carlo Neutron and Photon Transport (MCNP) [2], to run on a minicomputer. Next, a suitable anthropomorphic phantom was selected for the effective dose equivalent calculations. The male MIRD phantom was modeled as an input file for the MCNP code.

A number of calculations were performed with the MCNP code and the MIRD anthropomorphic phantom. The results were then used as input to a second code written at The Pacific Northwest Laboratory (PNL) to calculate dose equivalents to various tissues and calculate the effective dose equivalent as defined in ICRP Publication 26, with certain exceptions. The weighting factors used in the PNL calculations were specific to the male phantom and not an average of male and female as defined in ICRP Publication 26. The weighting factors used are given in Table 1. The weighting factors are slightly different than those given in ICRP-26 in that the female breasts were not included. Future calculation will be done with a female phantom and appropriate weighting factors.

TABLE 1. Effective Dose Equivalent for Monoenergetic Neutron Beams

Cell	Weighting Factor	Dose Equivalent for Each Tissue Per Unit Neutron Fluence Incident Upon Body, mrem per n/cm^2 100 keV Neutron Beam	1 MeV Neutron Beam
Lung	0.12	2.65×10^{-7}	8.85×10^{-6}
Liver	0.00	2.15×10^{-7}	8.22×10^{-6}
Stomach	0.06	2.69×10^{-7}	1.07×10^{-5}
Small Intestine	0.06	4.45×10^{-8}	4.62×10^{-6}
Upper Large Int.	0.06	7.10×10^{-8}	6.44×10^{-6}
Lower Large Int.	0.06	3.50×10^{-8}	3.94×10^{-6}
Testicles	0.25	1.97×10^{-6}	2.44×10^{-5}
Thyroid	0.03	9.33×10^{-7}	1.78×10^{-5}
Red Bone Marrow	0.15	2.89×10^{-9}	7.31×10^{-7}
Trunk	0.06	5.22×10^{-7}	8.43×10^{-6}
Effective dose equivalent per unit fluence incident upon body, mrem per n/cm^2		7.17×10^{-7}	1.16×10^{-5}

The methodology of ICRP Publication 26 requires that the dose equivalent be determined for various tissues in the body. The dose equivalent for each of these tissues is then multiplied by a weighting factor, and the products are summed to give the effective dose equivalent for the entire body. This method thus accounts for the attenuation of radiations within the body and the radiosensitivity of various tissues within the body. It also allows for the determination of effective dose equivalent in nonuniform fields and for the addition of dose equivalents from internal and external sources. In general, the effective dose equivalent is a more accurate estimator of the risks of radiation exposure than the currently defined dose equivalent.

Results from the MCNP calculations are shown in Table 1 for 100 keV and 1.0 MeV monoenergetic neutron beams. All organs listed are modeled mathematically in the computer code. The effective dose equivalent shown is for neutrons only and does not include the dose due to secondary photons. The MNCP code does, however, determine the effective dose equivalent due to secondary photons; that information is included in the curves shown in Figures 1 and 2, which show the conversion factors for the full energy range of the calculations.

Figure 1 demonstrates the differences between the conventional fluence-to-dose equivalent conversion factors and the effective dose equivalent conversion factors for the anterior irradiation with a parallel beam of monoenergetic neutrons. The effective dose equivalent is at least a factor of two lower than the conventional dose equivalent for most neutron energies. At the higher neutron energies above about 10 MeV, the effective dose equivalent values are closer to the conventional dose equivalent.

In ICRU Report 40 [3], a joint task group of the International Commission on Radiation Units and Measurements and the International Commission on Radiological Protection proposed a radical change in the definition of quality factor. The joint task group proposed that quality factor, which is now defined as a function of linear energy transfer (LET), be redefined as a function of the microdosimetric quality lineal energy. Lineal energy is a stochastic quantity that can be experimentally measured with instruments such as the tissue equivalent proportional counter. The quality factors in ICRU Report 40 are calculated at a point in tissue and were used to calculate the dose equivalent to the various tissues in the MIRD phantom, and the weighting factors of ICRU 26 were applied to determine the effective dose equivalent. The results of these calculations are presented in Figure 2 along with the conventional dose equivalent.

CONCLUSIONS

This calculational method is appropriate for most anticipated health physics monitoring situations. Variations in physical size of the human body can be taken into account as well as most of the potential exposure geometries (point source, beams, and isotropic). For many spectra, the effective dose equivalent is smaller in magnitude than dose equivalent determined by present methods. The effective dose equivalent is also more accurate in that it takes into account the radiosensitivity of the individual organs.

In the future, differences between the male and female phantoms and different sizes of phantoms will be studied. Experimental measurements will also be conducted with an anthropomorphic phantom to verify the calculations.

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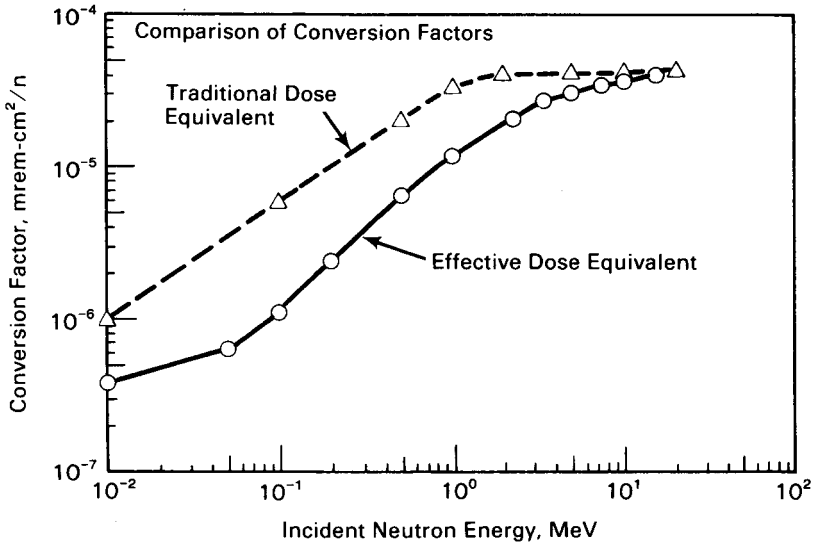

FIGURE 1. Comparison of Traditional and Effective Dose Equivalent

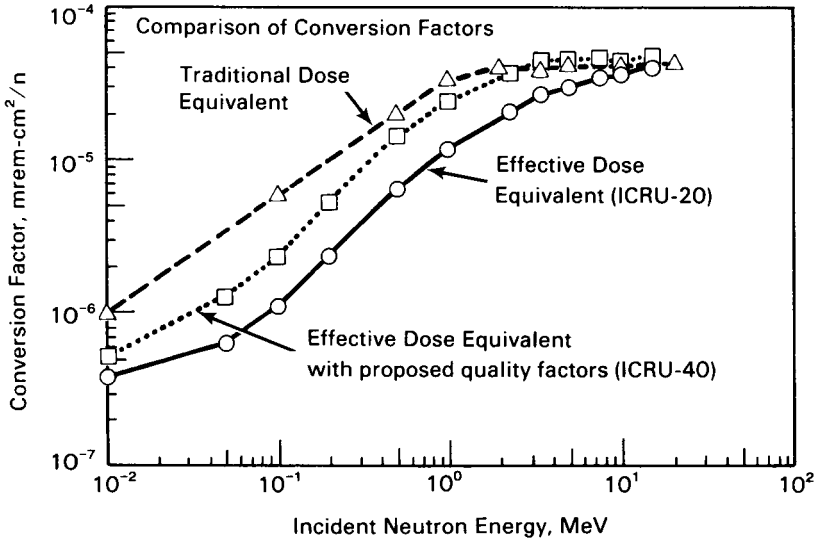

FIGURE 2. Comparison of Three Different Dose Equivalent Methods

REASSESSMENT OF ATOMIC BOMB NEUTRON DOSIMETRY OF HIROSHIMA BY USING RESIDUAL ACTIVITY OF ROCK

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ABSTRACT

The neutron dose reassessment of atomic bomb at Hiroshima was done by using residual activity of ^{152}Eu of granite rock specimen taken out from 41.5cm depth position inside the Motoyasu bridge pillar. The neutron dose (tissue kerma in air) at a point of 132m of SSW direction from the hypocenter was experimentally evaluated to be 15.5Gy. The obtained value shows that it is needed to argue and include the anisotropic performance of Hiroshima atomic bomb warhead. After the correction of anisotropic effect of warhead, the source yield of Hiroshima atomic bomb is estimated to be 17.7kt by combining the measured value obtained here with DS86 calculated value.

1. INTRODUCTION

The results of epidemiological survey of atomic bomb survivors at Hiroshima and Nagasaki give an important fundamental data in considering influences of radiation to a human body. The radiation dose of T65D (Tentative 1965 Dose; 1,2,3) has long been used as a foundation of dose data in estimating radiation risk for atomic bomb survivors. But from 1980 to 1982, Loewe pointed out(4,5,6) the uncertainty of T65D dose (especially about neutron dose). After that, the reassessment work of atomic bomb dose at Hiroshima and Nagasaki began widely. Four times conferences were held between U.S. and Japan till now (7,8,9). In the fourth conference, the DS86 (Dosimetry System 1986) dose assessment system was adopted and the revised dose was presented. In this system, the neutron dose was obtained by neutron transport calculations from the atomic bombs. The verifications of calculations were done for the following old measured data; ^{32}P (10,11,12) by $^{32}S(n,p)$ activation reaction and ^{60}Co (13) by $^{59}Co(n,\gamma)$. But these data have big uncertainties caused by statistical errors, sample position ambiguity and sample element composition vagueness. In (14), the residual activity measurement of ^{152}Eu was carried out by us for a deep part sample of rock at Hiroshima. In this report, this measured data were analyzed by a Monte Carlo transport calculation to get a neutron dose information.

2. MEASUREMENT OF EUROPIUM RESIDUAL ACTIVITY IN A ROCK

Europium exists about 1 ppm in the rock. Atomic abundance

ratio is 47.9 % for ^{151}Eu and 52.1 % for ^{153}Eu. Reactions of ^{151}Eu+n→ ^{152}Eu and ^{153}Eu+n→ ^{154}Eu were caused by Hiroshima atomic bomb. We mainly measured γ rays of ^{152}Eu (^{154}Eu was difficult to measure for a big decrease). Neutron absorption cross section of ^{151}Eu(n,γ) reaction is picked out from ENDF/B-V cross section library, which is nearly 1/v shape. The magnitude of this cross section is about 5900 barn at 0.025eV.

As a specimen, we chose a bridge pillar (82.5Lx82.5Wx178Hcm) made of granite in Motoyasu bridge at 132 m point of SSW of hypocenter of Hiroshima (Fig.1). This specimen has a clear situation with no material around it. We hollowed out stone cores of 6.8 cmϕ to the direction from east to west, north to south and top to bottom at about center position of each surface. We made stone board slices of 2 cm thickness from these cores and measured residual γ rays by an intrinsic Ge detector of 124 cc. The measured result was corrected into the absolute activity at the time of 8:15 a.m. on August 6 in 1945. The rock composition was measured for the calculation. Moisture, which has big influence to neutron transport, was also checked precisely by Karl Fischer volumetric method etc. The water content was 0.60± 0.08%. The ratio of europium in a granite specimen was 0.85～1.05μgEu/g-granite, which is 0.88± 0.06 ppm, by an activation method using a nuclear reactor. The obtained activity was 3.27± 0.28(Bq/mgEu) (30.6± 2.6(Bq/mgEu) on August 6 in 1945) at the rock center.

3. ANALYSIS

The continuous Monte Carlo code MCNP (15) was used for neutron transport calculations. The neutron cross sections that are incorporated in MCNP code were used in calculations. About ^{151}Eu(n,γ) reaction cross section, ENDF/B-V data were divided into appropriate groups.

The calculation was conducted by two steps. First, a point source term of atomic bomb based on two dimensional calculated result of Streetman (16) was prepared at 580 m altitude in air from the ground, and neutron transport calculation in air was conducted. The spectral shape and angular distribution of neutrons were obtained by this calculation at a position of a rock pillar specimen. In the next step, a subroutine was especially prepared that supplied a box type plane source in the neighborhood of the bridge pillar. In this case, the first step results of spectral shape and angular distribution of neutrons were used as an input for the second step calculation. Angular and energy dependent neutrons were produced toward a bridge rock pillar from the box type plane source. Finaily, the calculated results were summarized in the form of europium activity to one neutron incidence. Calculated europium activity to one neutron that was obtained by these two combinational calculations is compared with measured one. By this analysis, neutron fluence at a stone bridge pillar position is obtained. Tissue kerma in air is also obtained by multiplying a kerma factor.

4. RESULTS AND DISCUSSIONS

The relation between measured and calculated values are

$$A_{exp} = f \cdot \lambda \cdot R_{cal} \cdot p . \qquad (1)$$

Here, A_{exp} is a measured ^{152}Eu decay rate per Eu 1mg at the time of atomic bomb explosion (30.6± 2.6). f is an incident neutron number for these produced ^{152}Eu. R_{cal} is generation rate of ^{152}Eu per Eu-atom per source neutron obtained by a calculation at the depth of 41.5 cm (0.2162×10^{-24}). ρ is Eu atom number per Eu 1mg (3.963×10^{18}). λ is decay constant of ^{152}Eu (1.649×10^{-9}). By substituting these values into the eq.(1), the incident neutron number f at a box type sauce position became 2.166 x 10^{16}[n·(produced ^{152}Eu)$^{-1}$].

Geometrical factor for one neutron incidence of a whole area of a box type neutron source was $\phi_0 = 5.217 \times 10^{-4}$ [n/cm^2] by the Monte Carlo calculation with no stone bridge pillar. Total neutron flux obtained by this analysis are $\phi = f \cdot \phi_0 = 1.130$ x 10^{13} [n/cm^2]. Tissue kerma in air was obtained by multiplying this neutron flux by kerma factor determined in ICRU Report 26. The obtained tissue kerma in air at the bridge pillar position of 132 m from Hiroshima hypocenter is 15.5Gy (1550rad).

Corresponding to this 132m point, T65D(2,3), LLNL(Loewe;6), ORNL(Kerr;17,18) and DS86(9) gave tissue kerma in air to be 12iGy/12.5kt, 65.2Gy/ 15kt, 24.6 Gy/12.5kt and 31.4 Gy/15kt respectively. The difference between the measured value and these calculations is caused by the anisotropy of emitted neutrons from Hiroshima atomic bomb (Fig.2), and we can correct this effect by using Whalen's experiments (16,19) of angular neutron flux distribution using a replica of Hiroshima atomic bomb warhead. The forward component suffered by a metal shielding of about 56 cm thickness was decreased to 41.6% in comparison with an isotropic emission. Thus the corrected results of calculated values are shown in a column of "anisotropic" in Table 1 together with original calculations.

According to this attenuation correction and assuming a source yield of Hiroshima atomic bomb to be 15kt, the values by T65D, LLNL, ORNL and DS86 became over or underestimation of +289 %, +75 %, -21 % and -15 % respectively. By contraries, we could assume the source yield to be unknown, and could obtain the yield by a reciprocal calculation. The new source yield was obtained by comparing our measured result of 15.5Gy with DS86 of 13.1Gy/ 15kt, because it is considered that the DS86 system is most reliable depending on the knowledge at the time of present. Whereupon, the source yield of Hiroshima atomic bomb became 17.7 kt. This value is very near the value of 18 kt presented by Kaul(20) recently.

5. CONCLUSION

The atomic bomb neutron dose at 132m point to SSW direction from Hiroshima hypocenter was reassessed through the analysis of experimental value of the measured ^{152}Eu residual activity at 41.5cm depth in granite rock. The obtained neutron tissue kerma in air at this point is 15.5Gy (1550rad). By comparing the previous estimation of T65D, LLNL, ORNL and DS86, it was shown that the estimation of anisotropic effect of emitted neutrons from Hiroshima atomic bomb warhead was required for the correct dose evaluation on the ground. The source yield of Hiroshima atomic bomb was also estimated to be 17.7kt. This value is very close to 18kt, the latest estimation of Kaul in 1987.

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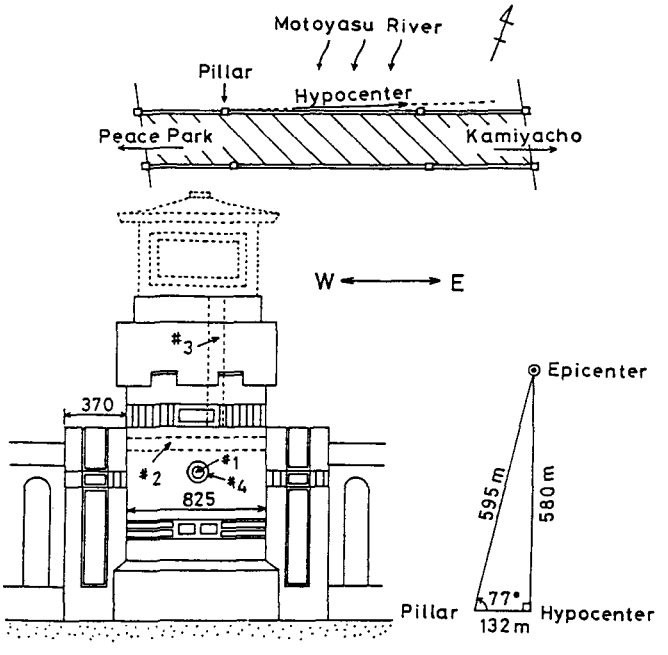

Fig.1 Motoyasu Bridge Pillar

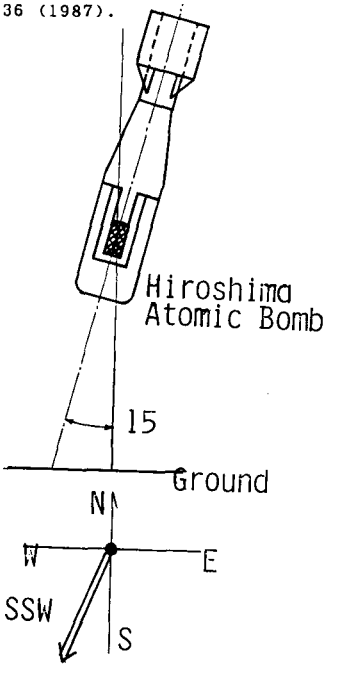

Fig.2 Hiroshima Atomic Bomb Warhead

Table 1 Neutron Tissue Kerma in Air at 132 m Position from the Hypocenter of Hiroshima

Reference	Burst height	Source yield	12.5kt Isotropic	12.5kt Anisotropic	15kt Isotropic	15kt Anisotropic
T65D	570m&580m	12.5 kt	121 Gy	50.3 Gy	(145 Gy)	(60.3 Gy)
Loewe	579m	15 kt	(54.3 Gy)	(22.6 Gy)	65.2 Gy	27.1 Gy
Kerr & Pace	580m	12.5kt	24.6 Gy	10.2 Gy	(29.5 Gy)	(12.3 Gy)
DS86	580m	15 kt	(26.2 Gy)	(10.9 Gy)	31.4 Gy	(13.1 Gy)
This work through the analysis of experiment.	Real height	Real kt			15.5 Gy	

NEUTRON FLUENCE TO DOSE AND DOSE EQUIVALENT CONVERSION FACTORS: A COMPARISON FOR SPECTRA OF INTEREST

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ABSTRACT

Many different sets of neutron fluence to dose and fluence to dose equivalent conversion factors are available for use by dosimetrists. Different sets exist because there is a variety of dose and dose equivalent quantities in use today (e.g.; kerma, effective dose equivalent, individual dose equivalent-penetrating) and because different authoritative bodies (e.g.; International Commission on Radiological Protection) advocate the use of their own set of factors. For the dosimetrist to make informed decisions concerning the selection of a set of conversion factors, differences in results due to the use of various data sets should be known. A good way to compare data sets is by use of the spectrum averaged fluence to dose and fluence to dose equivalent conversion factors developed from those data sets for spectra of interest.

Three different sets of neutron fluence to dose conversion factors and four different sets of neutron fluence to dose equivalent conversion factors have been used to calculate spectrum averaged values for twelve different neutron energy spectra. The spectra for which values have been calculated include Cf-252 (unmoderated and D_2O-moderated), pressurized water reactors (four locations), and the Health Physics Research Reactor (unmoderated and five moderated spectra). Spectrum averaged values are compared and discussed. Use of different sets of conversion factors leads to dose variations of more than 30% and dose equivalent variations of more than 50%.

Research sponsored by the Office of Health and Environmental Research, U. S. Department of Energy under contract DE-AC05-840R21400 with the Martin Marietta Energy Systems, Inc.

A NEW TYPE ACTIVE PERSONAL DOSEMETER WITH A SOLID STATE DETECTOR

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INTRODUCTION

At present there is no active personal neutron dosemeter which is commercially available and covers a wide energy range from thermal to MeV. There have been a few works on pocket neutron dosemeters that indicate the dose equivalent and a review on these dosemeters has been reported(1). Among of these works, two works by Tyree and Falk(2), and Eisen et al.(3,4) have described the use of a silicon surface barrier detector and proposed a small size and real time dosemeter for personnel monitoring. Tyree and Falk have used a polyethylene radiator which is sensitive in the MeV energy range, and Eisen et al. have extended the sensitivity to the energy from 1 eV to 15 MeV by using both B-10 and polyethylene radiators.

We have developed a new type personal dosemeter by using a B-10 doped silicon p-n junction detector with a polyethylene radiator and a polyethylene moderator. The purpose of this study was to develop a real time neutron dosemeter with a nearly flat response in the energy range from thermal to 15 MeV and low angular dependence to the incident neutron direction. The neutron response of the dosemeter was obtained with the Monte Carlo calculation and the monoenergetic neutron experiment in a free air field and also under a condition attached on a phantom.

DETECTOR DESIGN

A planar-type silicon p-n junction detector fabricated by Fuji Electric Co. Ltd. was applied for a neutron dosemeter. A boron film enriched in 90% B-10 is deposited in about 0.6 μm thickness on an n-type silicon crystal. A polyethylene radiator of 0.8 mm thickness is positioned in front of the boron film. The silicon detector encapsulated in a stainless steel case is covered with a hemispherical polyethylene moderator of 1 cm thickness.

This dosemeter has a sensitivity to neutrons of wide energy range. Low energy neutrons can be detected by alpha ions from the B-10(n,α)Li-7 reaction and fast neutrons by the recoil protons from the elastic scattering of hydrogen in the polyethylene radiator, and the polyethylene moderator has a role to increase its sensitivity to the intermediate energy neutrons and also to depress its angular dependence.

EXPERIMENT

The neutron response of this dosemeter was measured in the monoenergetic neutron field at the Fast Neutron Laboratory of Department of Nuclear Engineering, Tohoku University and the moderated Cf-252 neutron field at the Cyclotron and Radioisotope Center, Tohoku University. The five monoenergetic neutrons of energies of 150 keV, 500 keV, 1 MeV, 5 MeV and 15 MeV were produced by the p-T, p-Li, d-D and d-T reactions by using the Dynamitron accelerator.

Figure 1 shows the experimental arrangement. The absolute neutron fluxes were measured by the fission chamber placed in front of the dosemeter. The contribution of the room scattering was evaluated by the shadow shield method, consisting of 20-cm long iron and 30-cm long boron polyethylene shield. The hydrogen proportional counter was also used subsidiarily as a neutron flux monitor. The dosemeter was placed in a free air field or in front of a ellipsoidal water phantom or a tissue-equivalent phantom developed by the Central Research Institute of Electric Power Industry (5). The dosemeter response to neutrons were measured under these three conditions. The dosemeter was operated at very low voltage (+5V) in order to suppress the gamma-ray sensitivity and the output pulses were fed into the multi-channel pulse height analyzer. The thickness of the depletion layer was estimated to be about 50 μm.

RESULTS AND DISCUSSIONS

Figures 2 and 3 show examples of the pulse height spectra for 151.3 keV and 5.0 MeV monoenergetic neutrons normally incident to the front surface of the dosemeter attached on the water phantom. The pulse height spectrum for 151.3 keV neutrons shows a Li-7 peak and two alpha peaks corresponding to 1.47 MeV and 1.77 MeV energies from the B-10(n,α) reaction, while for 5.0 MeV neutrons a plateau peak of recoil protons from the H(n,n) reaction is added to Li-7 and alpha peaks. The recoil proton peak could be noticed for neutrons of energy above 1 MeV and made an increase of the sensitivity of the dosemeter to fast neutrons. Only the pulses beyond the discrimination level indicated in Fig. 3 were integrated to obtain the real neutron counts, since the lower pulses included the electrical noises and gamma-ray pulses. The neutron response of the dosemeter was obtained experimentally by dividing this real neutron counts by the incident neutron flux measured with the fission chamber. The results obtained in a free air field and on the water phantom are shown in Fig. 4.

The experimental results are limited to neutron energy above 150 keV in the present stage, then the dosemeter response to neutrons below that energy was calculated by the group Monte Carlo code MORSE(6). The MORSE calculation gave the neutron flux $\phi(E)$ at the position of the silicon detector through the polyethylene moderator for thermal to 15 MeV neutron incidence. The dosemeter response was estimated as follows,

$$R = N \int \phi(E)\ \sigma(E)\ dE$$

where N is the total number of B-10 atoms and $\sigma(E)$ the B-10(n,α) reaction cross section. The calculated results are also shown in Fig. 4 in a free air field and on a water phantom for comparison. The calculated results show very good agreement in absolute values with the experimental results below 1 MeV, where the contribution of recoil protons can be negligible. The neutron response of the dosemeter on the water phantom becomes much higher in the higher neutron energy range than that in a free air field, because of the increase of slow neutrons backscattered from the phantom.

From the figure, it was clearly found out that the dosemeter has rather flat response to neutrons in a wide energy range from thermal to 15 MeV under a condition attached on a phantom. This neutron response is much better than that of personal dosemeters now in use, such as film badge, albedo dosemeter and solid state track dosemeter. We are now performing to increase its sensitivity to neutrons and to fit the response further closer to the flux-to-dose conversion factor defined by the ICRP-21(7).

ACKNOWLEDGMENT

The authors wish to thank Drs. M. Baba and T. Iwasaki for their kindful operation of the Dynamitron accelerator during our experiment.

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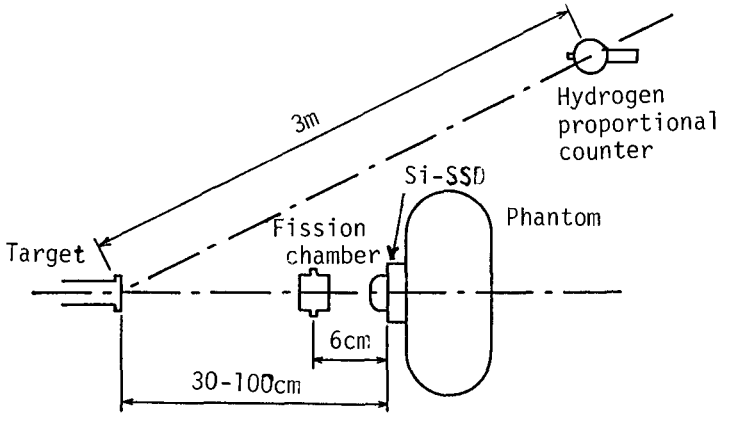

Fig. 1 Experimental arrangement

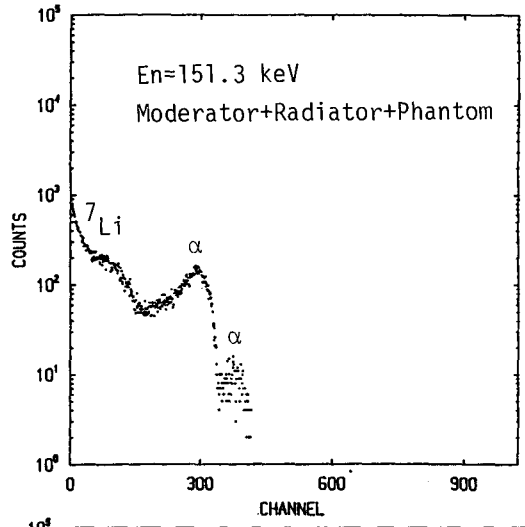

Fig. 2 Pulse height spectrum for 151.3 keV mono-energetic neutrons

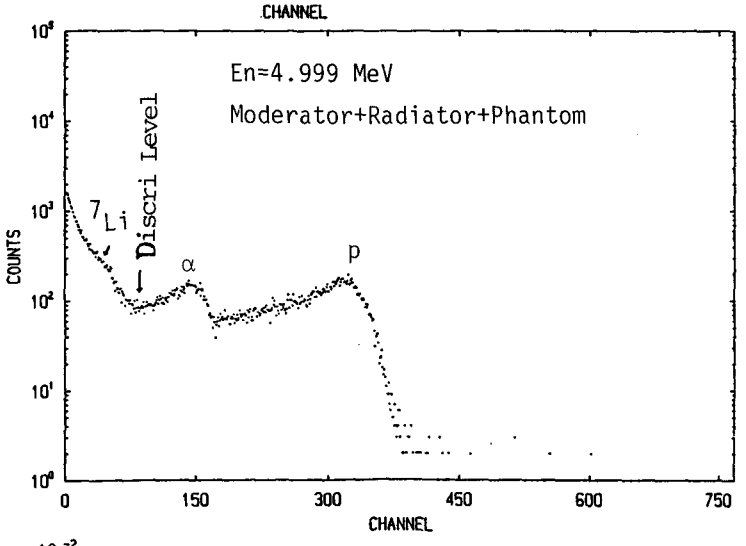

Fig. 3 Pulse height spectrum for 5.0 MeV monoenergetic neutrons

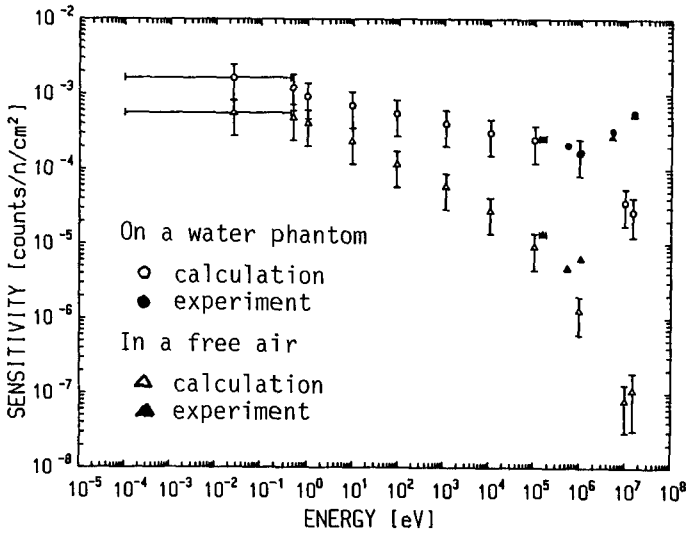

Fig. 4 Neutron response of the dosemeter in a free air and on a phantom obtained by experiment and calculation

DEVELOPMENT OF A NEW SOLID STATE NEUTRON DETECTOR FOR POCKET DOSEMETER

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SUMMARY

A new solid state neutron detector for a pocket dosemeter was developed. This is one of planer type p-n junction silicon semiconductor detectors. A thick p+ layer of boron-10 is made on a n type high-purity silicon substratum by a plasma CVD (Chemical Vapor Deposition) process.

The detector, amplifier and discriminator are assembled as a hybrid integrated circuit for compactness together with a polyethylene radiator to obtain neutron sensitivity up to MeV region.

DETECTOR STRUCTURE AND FABRICATION

Figure 1 shows the structure of the present detector. A Sio_2 film of about 1 μm is formed by SiH_4 on a surface of n type silicon substratum of 10k ohm-cm. A window for doping boron-10 with enriched B_2H_6 is provided on the surface by photo-etching process. The window is then coated with a boron-10 film of 0.6 um in thickness by dc plasma doping process. A p+ layer is also formed in the process simultaneously. Electrodes for signal lead are made by vacuum depositioning of aluminum onto both surfaces of the substratum.

Figure 2 shows the density distribution in the boron layer of boron-10 atoms, boron-11 atoms, hydrogen atoms and silicon atoms. This distribution was obtained by a secondary ion mass analysis. Figure 3 shows the block diagram of the present detector. We have employed a charge amplifier as a preamplifier and a bipolar amplifier having 1.5 μsec shaping time as a linear amplifier. All these circuits consist of chip electronic parts and miniflat integrated circuits, and compose one hybrid integrated circuit mounted on a ceramic substratum together with a 0.8 mm thick polyethylene radiator (1). A bias voltage of the detector is +5V, which is common with a power source voltage.

RESULTS

Figure 4 shows the pulse height spectrum of the detector for thermal neutrons. The thermal neutron sensitivity measured over the discrimination level in this figure is 3.50 x 10^{-4} cps/nv, the sensitivity to 1 MeV and 144 keV fast neutrons is 2.52 x 10^{-4} cps/nv and 3.13 x 10^{-4} cps/nv, respectively. Although the moderator is as small as 1 cm, the sensitivity to fast neutrons is not much different from the thermal neutron sensitivity by measuring recoil protons from the radiator. A water phantom of 20 x 30 x 40 cm was used for these measurements.

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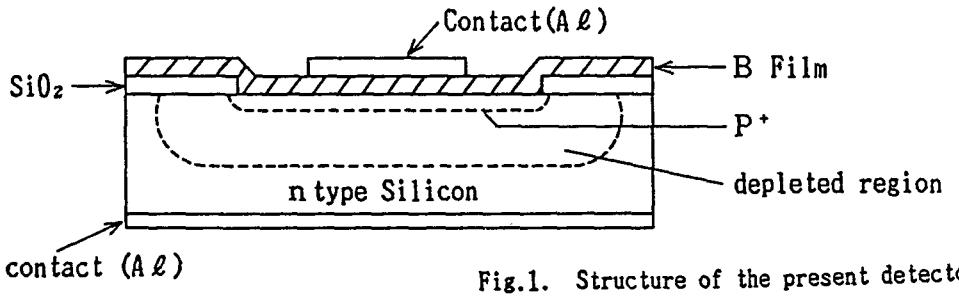

Fig.1. Structure of the present detector

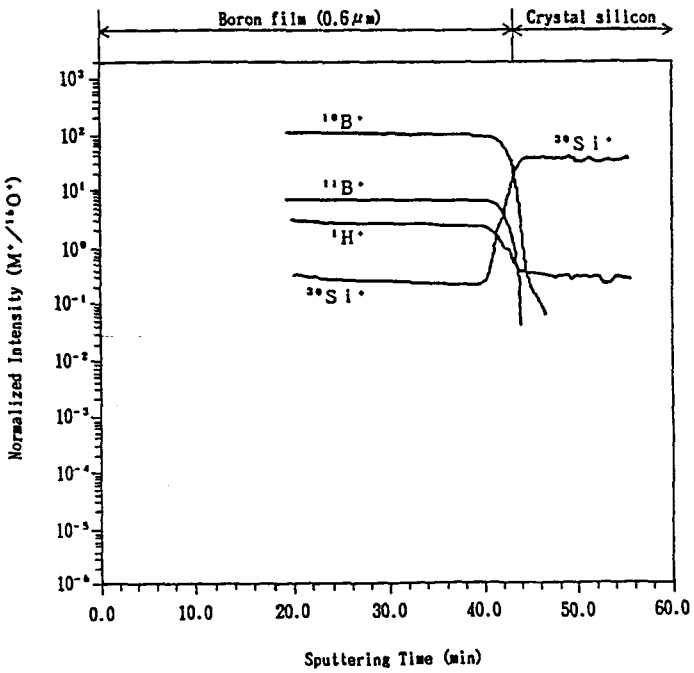

Fig.2. Measured distribution of atomic density for ^{10}B, ^{11}B, ^{1}H and ^{30}Si in the surface of the present detector

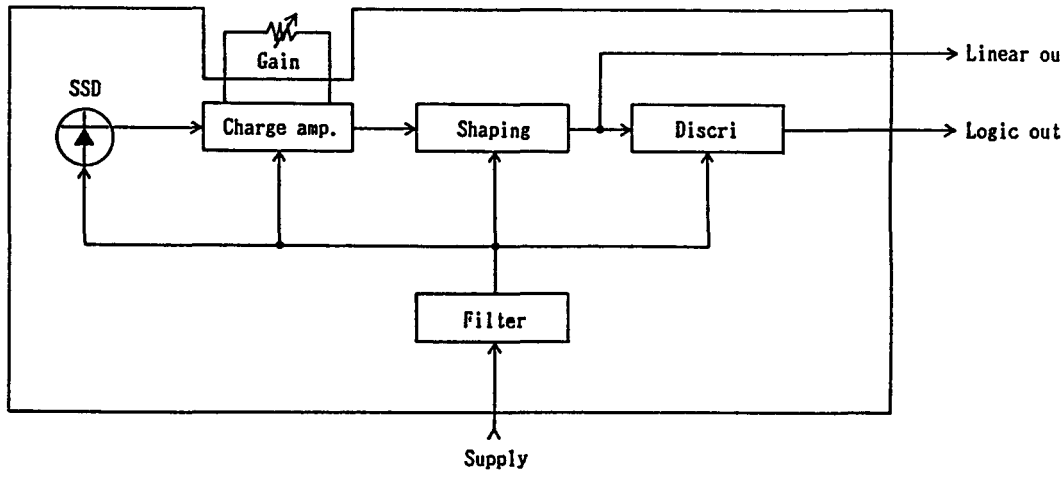

Fig.3. Block diagram of the present detector

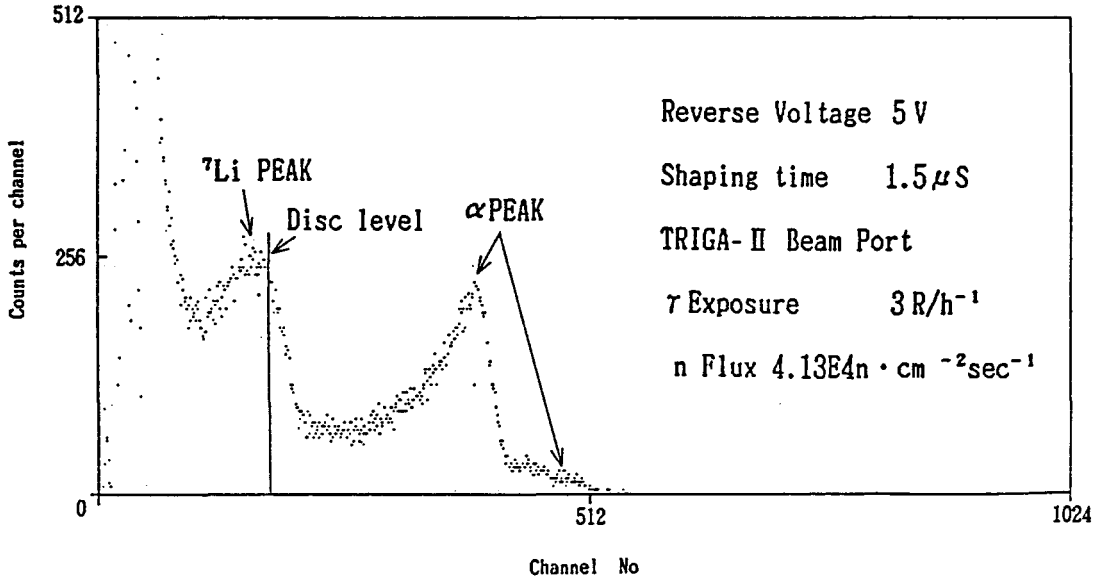

Fig.4. Pulse height spectrum for thermel neutron obtained by the present detector

EXPERIENCE WITH A FIVE ELEMENT NEUTRON DOSEMETER

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ABSTRACT

The Health Physics Department of the Netherlands Energy Research Foundation ECN operates a dosimetry service specialized in beta and neutron radiation. The Department serves over 100 clients with a total of 2000 radiation workers. Recently all film badges were replaced with TLD's.

In the poster the experience with this new fully automated system will be reported, with emphasis on the neutron dosemeter results. This dosemeter consists of a four element albedo system, complemented with a polycarbonate track-etch detector. This combination is expected to yield reliable dose-equivalents for neutron energies from thermal to fast, also in mixed fields.

PERSONAL DOSIMETRY MEASUREMENTS AT THE BEVALAC WITH CR-39 AND NTA EMULSIONS AT A HIGH ENERGY ION ACCELERATOR

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ABSTRACT

An intercomparison of neutron measurement techniques was made outside of a 3-foot thick concrete shield at the Bevalac during a period when 670 MeV·A neon ions were stopped in a copper target. Measurements at 90° to the beam axis were made with CR-39 and NTA emulsions on both stationary and rotating phantoms. Comparisons were made with LiF thermoluminescent dosimeters in Andersson-Braun moderators, moderated BF_3 neutron fluence monitors, carbon-11 production above 20 MeV,[3] and with measurements made with a Bonner multisphere neutron spectrometer. Of particular interest is the relative performance of CR-39 and NTA emulsion in a radiation field in which a large fraction of the dose equivalent is delivered by neutrons of energy greater than 20 MeV.

COMPARISON OF NEUTRON TRACK DETECTOR SENSITIVITY AS A FUNCTION OF THE ETCHING PROCESS

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The use of neutron track detectors made of allyl diglycol carbonate (ADC) is increasing as the international nuclear community seeks better methods to measure dose to personnel occupationally exposed to neutrons. The dosimetry-grade polymer ADC (CR-39), produced by PPG Industries, has undergone extensive evaluation in the past six years to prepare for its routine use at U.S. Department of Energy facilities. During this time many changes have been made in the way the CR-39 is processed. Each major improvement in the speed of processing or ease of readout has been adopted and current procedures provide excellent sensitivity and a nearly energy-independent energy response over the range of 144 keV to 2.8 MeV.

Chemical etching with potassium or sodium hydroxide was first used in processing CR-39, and some laboratories continue to chemically etch their neutron-exposed samples. Chemical etching alone is the easiest etching method, requiring little sample handling and the fewest processing variables. The main drawback with chemical etching by itself is that the pits etched at the damage sites are very small. Material artifacts may be indistinguishable from radiation-induced tracks and two or more tracks in close proximity may merge to form a single large track.

Some years ago, Luigi Tommasino from Italy and Dick Griffith from Lawrence Livermore National Laboratory (LLNL) collaborated in using electrochemical etching after a short chemical etch cycle. This greatly enlarged track size and facilitated automatic track counting. This combination also made discriminating between radiation damage tracks and material artifacts less difficult. The main disadvantage with this method is the double handling required for the samples. More recently, Luigi Tommasino tried a one-step electrochemical process and was encouraged by the results. Optimization of this process by Dale Hankins at LLNL demonstrated that the one-step electrochemical etch actually improved neutron sensitivity while reducing processing time.

There are three major processes used to etch CR-39: 1) chemical etch at high temperatures, 2) short chemical etch at high temperatures followed by electrochemical etch at ambient temperatures, and 3) electrochemical etch at high temperatures. There are many variations used in the processing of the first and third methods. The second process has largely been abandoned in favor of the third process.

Based on differences in results among the processes, there was evidence that the etching process selected determined the shape of CR-39's energy response curve. Pacific Northwest Laboratory (PNL) recently had an opportunity to participate in a

joint European/U.S./Canadian CR-39 dosimeter intercomparison and had sufficient numbers of samples exposed to a variety of energies to try all three processing methods. The samples were exposed to the following energies: 144, 250, 1200, 2800, 5300, and 14,700 keV. Some of these were from accelerators and some were from reactor beams. Monoenergetic exposures to the same energies were also performed using PNL's Van de Graaff accelerator to provide an additional comparison of results of the electrochemically processed samples only. All samples were exposed in air to incident beams. A nominal 5 mil covering of polyethylene on top and bottom of each sample served to protect the samples from alpha irradiation from radon exposure and to act as a neutron radiator.

An attempt was made to use standard conditions for the etching processes. Some variations in the energy response curve would be expected as parameters are changed. Parameters used for the chemical etch were 6.5 N KOH at 70°C for 14 hours. The samples were etched in an agitated water bath and were manually read out using a magnification of 400X. The resulting energy response curve is presented in terms of tracks/cm^2-mrem (Figure 1) and the data points are summarized in Table 1. The results show no response above background at 144 keV and a maximum at 2800 keV.

Samples processed by chemical/electrochemical etching were etched using the following parameters: chemical etch at 6.5 N KOH and 60°C for 3 hours followed by electrochemical etch using 6.5 N KOH and a voltage of 2500 V and 2000 Hz for 5 hours. This combined process detected the 144-keV neutron exposures and peaked at 1200 keV (Figure 2). An automated image analysis system was used with a 4X objective lens magnification to count the tracks in these samples.

The electrochemical etch at high temperatures was performed using the following parameters: electrochemical etch with 6.5 N KOH in an oven at 60°C at 2500 V and 60 Hz for 5 hours, followed by an increase in frequency to 2000 Hz for 23 minutes. This process produced a curve that detected the 144 keV with much improved efficiency and showed much less change in the response up to 2800 keV (Figure 3). The response at 14,700 keV was detected with good efficiency. These samples were counted automatically using a 4X objective lens magnification. As was illustrated with both other processes, the response drops at the 5300 keV point. Because the curve was not as smooth as seen in previous monoenergetic studies, samples exposed at the PNL accelerator for comparison were processed by electrochemical etch. Results of these exposures show a smooth curve (Figure 4) and a maximum response holding quite steady from 144 keV to 565 keV.

A summary of the results from the three processes, including the additional accelerator exposures, records the tracks/cm^2 per mrem and mSv (Table 1).

TABLE 1. Summary of neutron energy response as tracks/cm^2-mrem, mSv

Energy, keV	Chemical Etch	Chemical/Electro	Electrochemical Etch: Intercomparison (I), Accelerator (A)	
			(I)	(A)
144	-0-	0.35 (35)	5.98 (598)	9.43 (943)
250	1.15 (115)	0.63 (63)	9.92 (992)	9.73 (973)
565	1.46 (146)	1.75 (175)	7.83 (783)	9.25 (925)
1200	3.50 (350)	3.30 (330)	7.35 (736)	7.63 (763)
2800	4.76 (476)	2.00 (200)	5.75 (575)	7.11 (711)
5300	2.72 (272)	0.49 (49)	1.89 (189)	1.86 (186)
14700	1.63 (163)	0.86 (86)	2.31 (231)	4.20 (420)

Analysis of the shapes of the energy response curves resulting from the three processing methods was initiated to identify differences in the response curves. This was carried out for two reasons: to confirm that the electrochemical process had the flattest energy response curve over the widest range and to run all three methods on the same samples to look for useful differences that may assist in passive spectrometry. With the 5-mil radiator thickness and the stated processing parameters, the results showed that chemical etching by itself did not detect 144-keV neutrons and the efficiency in terms of tracks per mrem was low up to 1200 keV. An interesting aspect is that the efficiency at the higher energies of 5300 and 147,000 keV was reduced more slowly than with the other two processes. The chemical/electrochemical process showed a higher efficiency than chemical etching alone in detecting the lower energy neutrons, and it has a smoother curve than the others, but its overall energy dependence is greater than desired.

Samples processed by electrochemical etch at high temperatures demonstrated a higher response for each exposure energy and a flatter overall response. Compared with the other intercomparison samples, the curve shows a smaller change in the sensitivity between 144 and 2800 keV with a peak at 250 keV. The efficiency in terms of tracks/dose equivalent is considerably higher for electrochemical etch than for the other two processes. The curve resulting from exposures at the PNL accelerator provides an especially notable curve in showing very little energy response from 144 keV up to the 5300-keV point. This greatly reduces the complexity in interpreting results from samples exposed to more than one primary energy in this range. From the aspects of energy dependence and neutron sensitivity, this study shows that under the conditions stated, electrochemical etching at elevated temperatures is superior to chemical and chemical/electrochemical etching of CR-39 neutron track detectors.

[This work was performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830 in support of the Personnel Neutron Evaluation and Upgrade Project managed by Pacific Northwest Laboratory.]

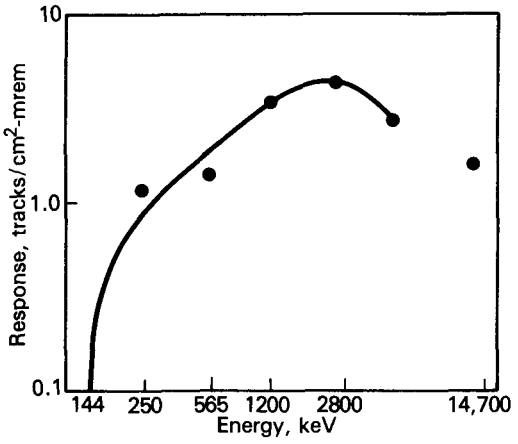

FIGURE 1. Neutron Energy Response Curve from Chemical Etching

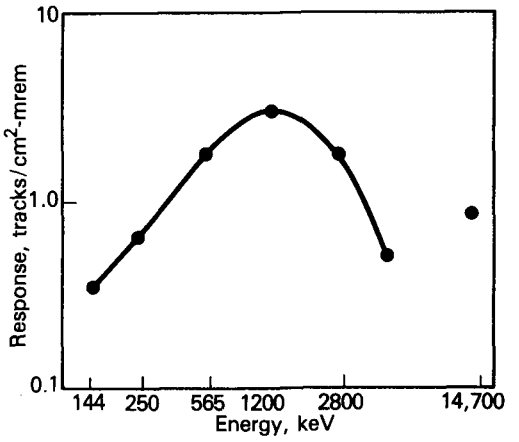

FIGURE 2. Neutron Energy Response Curve from a Combination of Chemical and Electrochemical Etching

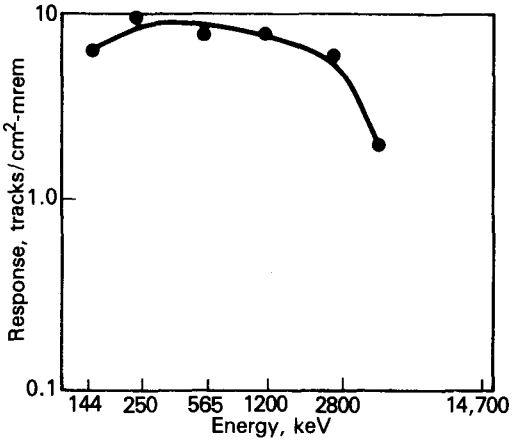

FIGURE 3. Neutron Energy Response Curve from Electrochemical Etch Accelerator Exposures

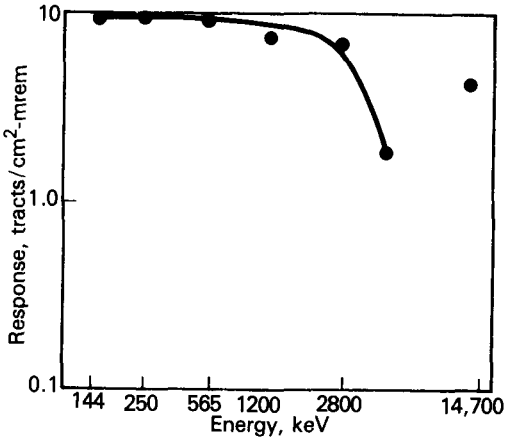

FIGURE 4. Neutron Energy Response Curve from Electrochemical Etch -- All

ADVANCES IN THE DEVELOPMENT OF CR-39 BASED NEUTRON DOSIMETERS

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A combination thermoluminescent dosimeter (TLD) and track etch dosimeter (TED), which can be used for detecting neutrons over a wide energy range, has been developed through recent research in passive neutron dosimetry. This dosimeter uses Li-600 TLDs to detect thermal and low energy neutrons reflected from the body, and the TED polymer of CR-39[a], to detect fast neutrons from proton recoil interactions with the polyethylene radiator or with CR-39 itself. Some form of the combination dosimeter is currently in use at several U.S. Department of Energy (DOE) facilities, and its use is expected to expand over the next year to include all DOE facilities where significant neutron exposures may occur. The extensive research conducted on the TED component over the past six years has continually focused on material improvements, reduction in processing time and dosimeter handling, and ease of sample readout with the goal of automating the process as much as possible.

Material Improvements

Since the beginning of DOE's involvement in track detector development, many improvements have been made in the materials and sample processing. Of the track detector materials available, CR-39, the highly cross-linked polymer of allyl diglycol carbonate, is the best material found to date for neutron detection over a wide range of energies. In recent years a higher purity of CR-39 monomer has been available and has led to a polymer with greatly improved detection consistency and reduced background tracks. With a nominal purity of 93%, this monomer has been dubbed "dosimetry grade." Anticipating that higher purity material may prove even better, further purification on a small scale led to almost nonexistent background on the resulting samples. A larger scale demonstration is underway to determine whether an ultra-high-purity material is worthwhile.

It was believed that the neutron damage of CR-39 was occurring at the polymer cross linkages. Researchers at the University of Connecticut, in conjunction with the Pacific Northwest Laboratory (PNL), provided experimental evidence that proved this was the case. They encased CR-39 into sealed aluminum containers and exposed the samples to several hundred thousand rad. Chemical analysis of the exposed samples indicated the presence of diethylene glycol and led to the conclusion that the carbonate bonds in the side-chain cross linkages were the sites of the majority of neutron-induced damage. With this in mind, they set out to add a second component to CR-39 with more proton-sensitive functional groups

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(a)Trademark of PPG Industries, Inc.

to make a CR-39 copolymer that is more neutron-sensitive. Chemical bonding and mechanical problems have prevented real success to date, but the efforts continue to develop a more sensitive material.

In addition to material improvements, current research goals are to expand the neutron energy range of CR-39 with the use of specialized radiators and process changes.

Processing Improvements

Processing of CR-39 was initially done by chemical etching and then by a chemical/electrochemical etch. A process using only electrochemical etching at elevated temperatures has been developed that shows a superior efficiency in track detection and has a nearly flat energy response curve over the energy range from about 150 keV to about 3 MeV. Considerable optimization of the electrochemical process has led to selecting parameters that include a two-cycle etch.

For this process, sample chambers are preheated in an oven at 60°C and a 6.5 N potassium hydroxide solution is added to the chambers to begin the etch. The first cycle is operated at about 47,000 V/cm of material thickness and a frequency of 60 Hz for 5 hours. The second cycle uses the same voltage but raises the frequency to 2000 Hz. This cycle runs for 23 min. The samples are then ready for track counting. This process produces 7 to 9 tracks/cm^2-mrem (700 to 900 tracks/cm^2-mSv) over the most efficient portion of its energy range from about 150 keV to 3 MeV.

The sample chamber design has also been greatly improved over the past few years from single cell chambers to 8- and 24-cell chambers. These multi-cell chambers have facilitated processing large numbers of badges and reduce the variation in the microenvironment surrounding the samples. A programmable power supply has been developed to keep the voltage steady and automatically change the frequency at the appointed time.

New image analysis systems provide many more options in sample readout. Optical systems capable of providing a direct count of the sample numbers and digitizing imaging systems capable of also analyzing the size and shape of the tracks are currently in use. The extent of automation varies from track counting only to automated stage movement, sample hoppers, autofocusing of the microscope, and sophisticated software to interface the components and manage the data files.

Track Size Distributions

With the use of automated track counters with size discriminators, it has been possible to evaluate the track size (area) distributions as a function of energy. A difference in track size with energy has been theorized by many researchers, and it is evident from observing tracks from monoenergetic sources versus continuous energy sources that differences in track size exist. Evaluation of samples from a series of

monoenergetic exposures shows such differences exist. Histograms of the shape of the size distribution curve changes in an orderly pattern with energy (Fig.1). A pattern developed going from a single peak to a bimodal formation and then back to the decreasing size distributions. The exposures represented in the figure were about 200 mrem (2 mSv) but fluences were not identical and the graphs were not normalized. Therefore, their shapes, not their magnitudes, are important. Not all of these are specifically distinguishable from each other, and it is too early to tell if they truly have identifying signatures. The 144 keV and 2.8 MeV results, for instance, look very similar.

With too little dose, a pattern may not be evident. With high doses, the tracks become deformed, smaller, and too uniform for analysis. It is not yet known how samples exposed to several spectra may be interpreted. Work is continuing to clarify the track-size distribution relationship and to determine whether it may have any benefit in evaluating dosimeters for spectra to which the personnel were exposed.

[This work was performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830 in support of the Personnel Neutron Evaluation and Upgrade Project managed by Pacific Northwest Laboratory.]

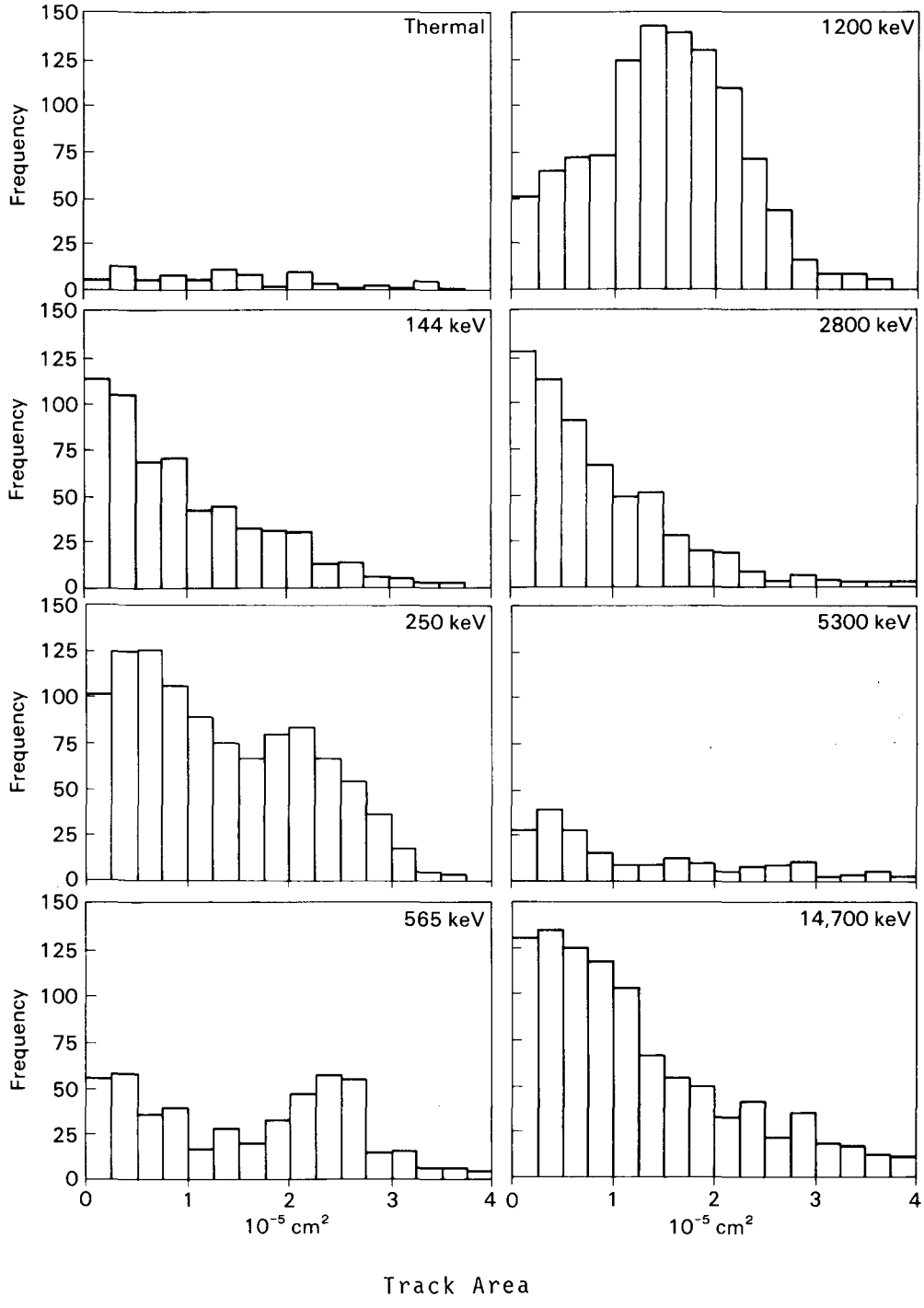

FIGURE 1. Track Size Distribution Histograms for Indicated Neutron Energies (not normalized for fluence)

A PERSONNEL NEUTRON DOSIMETRY SYSTEM USING LOW FREQUENCY ELECTROCHEMICAL ETCHING

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ABSTRACT

We have developed a personnel neutron dosimetry system based on the electrochemical etching of CR-39 plastic at elevated temperatures. We use a three-step etching procedure. Our first etching step is at 60 Hz, 3000 V for five hours in a 60°C oven using 6.5 N KOH. The second etching step consists of an additional 23-minute etch at 60°C, 2.0 kHz, and 3000 V, which increases the track size. The final step is a 15-minute post etch at 60°C, which gives the track a smoother appearance.

The foils are etched in etch chambers that can process up to 24 foils. Several of these chambers can be used simultaneously with a single power supply. These chambers are rugged and have been used daily for over a year.

The sensitivity of the system is about 5 tracks/mrem with a background of about 8 mrem, giving us a lower limit of sensitivity of approximately 10 mrem for the dosimeter (when three foils are used). At 400 mrem the precision of the foil's results has a standard deviation that averages about 5%. The dosimetry system is linear up to about 600 mrem and can be corrected for doses up to 2 rem. The energy dependence is fairly flat from 150 KeV up to 5 MeV, but drops by 42% at 14 MeV. It has no response to thermal or intermediate energy neutrons.

We observe no fading of the latent tracks. The dosimeter has a significant variation in directional response, dropping to about 30% at 90 degrees. We have studied the effect that changes in the etching parameters have on the results. These parameters include KOH normality, high voltage, frequency, etch time, oven temperature, and pre- and post-etch procedures. We are presently studying the relationship between the track size distribution and the incident neutron spectrum.

*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.

A 1/V ABSORBER DETECTOR FOR MEASURING NEUTRON SPECTRA

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INTRODUCTION

Accurate neutron spectrometry for intermediate energy neutrons (in the range of about 1 eV to 100 keV) is extremely difficult with any existing portable neutron spectrometer. Time-of-flight and crystal scattering methods are used, but the techniques cannot be applied in portable instruments. The Bonner sphere or multisphere spectrometer is often used by health physicists for measuring neutron spectra over a wide range of energes, but the unfolding technique requires choosing one of an infinite number of mathematically correct solutions to the Fredholm integral equations. For intermediate energy neutrons, the responses of the various conventional Bonner sphere configurations are not distinctive enough to provide meaningful intermediate energy spectra.

The problem of determining intermediate neutron energy spectra can be solved by using a material that responds as a 1/v neutron absorber, such as ^{6}Li, or ^{10}B, to cover neutron detectors such as a ^{3}He proportional counter or ^{6}Li scintillator. The term "1/v neutron absorber" indicates that the material has a neutron cross section that varies as the inverse of the velocity of the neutron, or as the inverse of the square root of the neutron energy. Isotopes with 1/v cross sections include ^{3}He, ^{6}Li and ^{10}B. A neutron detector covered by a 1/v absorber material will produce a signal that is dependent on the energy of the neutrons incident on the absorber material. A judicious choice of 1/v absorber material configured around one or more detectors can result in an instrument capable of unfolding the energy distribution of the incident neutrons.

RESPONSE FUNCTIONS FOR CANDIDATE CONFIGURATIONS

The 1/v detector system being developed at the Pacific Northwest Laboratory (PNL) consists of a small 1/v detector inside a material which both moderates and absorbs neutrons with a 1/v cross section. The detector(s) would be placed inside the moderator/ absorber, with count rates recorded at a number of locations inside the moderator/absorber.

Possible configurations for a 1/v detector system are being investigated by calculating the theoretical responses of each. Since the set of calculations needed to characterize a proposed configuration is easier to set up than an experiment, a number of different combinations of neutron moderating and absorbing materials can be investigated and compared before the most promising configurations are actually assembled and tested in the laboratory.

Calculations are performed using a radiation transport computer code, MCNP. MCNP is a code that uses Monte Carlo

methodology to determine the transport of neutrons and photons through various media. It uses a point-wise representation of cross sections rather than a multigroup scheme, so inaccuracies that can be introduced by cross section collapsing are avoided.

In one example calculation, a proposed system consisting of a 25-cm-diameter sphere filled with a homogeneous solution of boron-loaded glycerine was modeled. The sphere was exposed to an isotropic field of monoenergetic neutrons, and neutron interaction rates were calculated for a number of different depths within the sphere. The interaction rates recorded were the (n,alpha) reactions, indicating the response of a typical LiI neutron detector. Separate calculations were performed for several different neutron energies incident on the glycerine-filled sphere. For each of the energies, the calculated reaction rates were plotted as a function of depth within the sphere.

A graph of the results of this set of calculations is presented as Figure 1. It is apparent in this graph that there is a clear difference in reaction rates at depths of greater than about 5 cm, with higher energy neutrons producing appreciably higher count rates than lower energy neutrons. This discrimination is an important feature for a detector system that will be used to unfold neutron spectra.

LABORATORY TEST OF A 1/V DETECTOR

The results of the theoretical study were used to design and test a candidate 1/v detector system. In this study a detector system was assembled consisting of a spherical flask, 12.6 cm in radius. The flask was filled with glycerine, and a small spherical ^{3}He tube was used as the detector. The system was first modeled using the MCNP transport calculations, and count rates were determined at various depths in the sphere for exposure to neutrons emitted by an unmoderated ^{252}Cf source. For the ^{3}He tube the dominant interaction producing counts is the (n,p) reaction for neutrons.

The detector system was then exposed to an unmoderated ^{252}Cf source under the same conditions (distance from the source, source strength) as in the calculations. The detector was connected to standard nuclear electronics and a multichannel analyzer, and count rates were recorded with the detector in a number of positions within the sphere, matching the depths in the calculations.

The results of this comparison are presented graphically in Figure 2. One of the curves on the graph is the count rate predicted by the MCNP calculations, and the other curve is the count rate actually measured in the experiment. For shallow detector depths the agreement between the predicted and measured count rates is excellent. For deeper positions the agreement is not as good, primarily due to some features of the modeling. The modeling of only (n,p) reactions in the detector ignores other interactions which may contribute to the pulses generated in the detector, especially at high neutron energies. The

otherwise good agreement, however, indicates the system of calculations can be used to predict and optimize the responses of proposed systems.

CONCLUSIONS

The preliminary studies have not yet arrived at an optimized system. More investigation must be done to study other possible configurations. Once a promising configuration of moderator/absorber material and detector is identified, a laboratory system will be set up and tested. A set of response functions for count rate versus neutron energy at various depths must then be determined, and an unfolding algorithm will then be implemented to derive incident neutron flux as a function of energy from the measured count rates.

The methodology of a 1/v absorber detector system is in many ways similar to the Bonner sphere system. The 1/v system will have the advantage of being more portable and easy to use, since it will consist of a single moderating device with the detector(s) inside, rather than a set of five or more spheres. The search for an optimized detector system is necessary because it has the potential for overcoming a major drawback of the current Bonner sphere system: the poor energy discrimination in the intermediate energy region. An optimized 1/v absorber detector system could be a very useful tool for the determination of neutron fluxes, energy distributions, and dose equivalent rates in nuclear facilities.

Work supported by the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

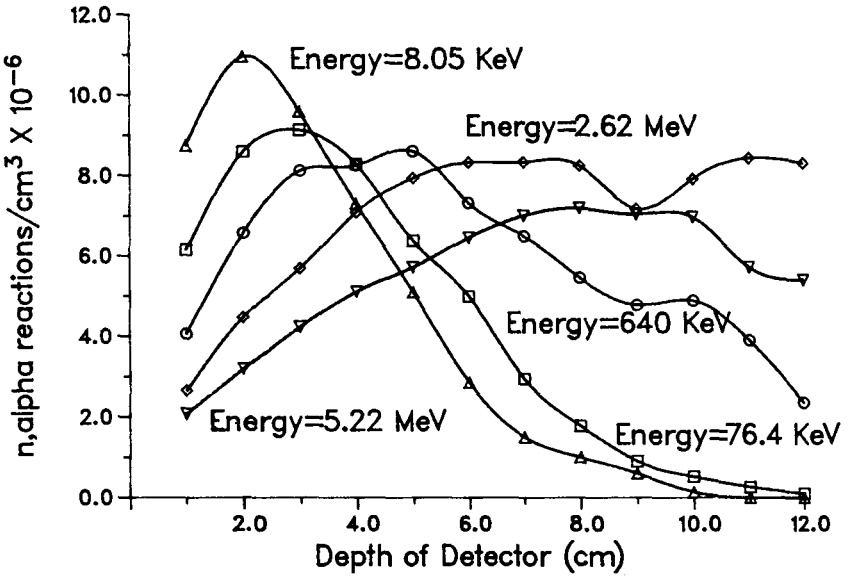

FIGURE 1. 1/v Absorber Detector Response

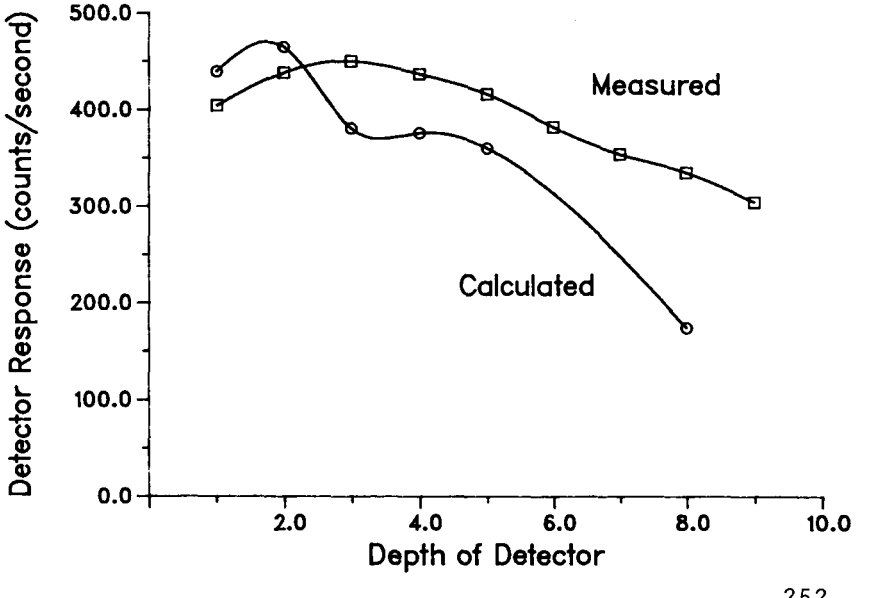

FIGURE 2. 1/v Absorber Detector Exposed to ^{252}Cf

PRACTICAL DETERMINATION OF DOSE EQUIVALENT USING LOW PRESSURE TISSUE EQUIVALENT PROPORTIONAL COUNTERS

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BASIC ASPECTS

Conventional dose equivalent meters for neutrons are based on the detection and registration of thermal or thermalised neutrons. The count rate of the detector sensitive to thermal neutrons and suitably combined with a hydrogenous moderator is calibrated in terms of dose equivalent rate by irradiation in a reference field. An important disadvantage of this principle is that the dose equivalent response is not constant with neutron energy.

Dose equivalent meters based on the microdosimetric technique of using low pressure tissue-equivalent proportional counters (TEPC) are expected to have a response which is closer to being constant and this has been the main reason for recent efforts by several groups to develop TEPC area and individual monitors[1]. The expectation is based on two facts: (1) The TEPC is a cavity chamber which enables precise determination of absorbed dose to tissue-like material, D, to be made for penetrating radiation. (2) Not only pulse rate but also pulse height information is used to evaluate dose equivalent. The puse height spectra are related to radiation quality and can be used to assess the mean quality factor Q. The product of Q and D is by definition dose equivalent H and the product of the measured values is the dose equivalent reading of the detector which can be used to estimate ambient dose equivalent H*(10). Figure 1 shows the dose equivalent per neutron fluence measured by a TEPC in monoenergetic neutron fields (counter wall thickness 2.5 mm, simulated dimeter[2] 2µm) in comparison to the flunce-to-H*(10) conversion function and the reading of a rem counter.

The TEPC detects events due to photons as well as to neutrons. The measured dose equivalent therefore includes both components. The measured pulse height spectra can be also used to evaluate the neuton and photon components separately and to calculate total and neutron mean quality factors. This diagnostic capacity of TEPC is an additional reason for the increasing interest in using this instrument as dose equivalent meter.

INVESTIGATION OF THE RESPONSE TO NEUTRONS

The excellent dose equivalent response of the TEPC for neutrons above a few hundred keV (Fig. 1) is mainly due to the fact that the measured pulse height, calibrated in terms of the microdosimetric quantity lineal energy y, is a good approximation of LET for recoil protons and to the accuracy of the instruments in absorbed dose measurements. The interpretation of the response and of the underlying microdosimetric spectra for neutrons below 100 keV has to take account of two different physical aspects:

(1) The quality of the TEPC as a LET-spectrometer deteriorates rapidly with decreasing neutron energy due to the insufficient range of the recoil protons in comparison to the cavity diameter. (2) The modification of the primary radiation field by the detector walls becomes of increasing importance. With regard to dose equivalent there are the two competing processes taking place in the counter wall, that of neutron fluence attenuation and that of neutron thermalisation with subsequent capture reactions in H and N. This is illustrated by the two fluence normalised microdosimetric spectra for 73 keV monoenergetic neutrons in Fig.2, one measured with a counter wall thickness of 2.5 mm tissue equivalent plastic A-150 (B), the other with a total wall thickness (A-150 + PE) of 20 mm (A). The comparison reveals the obviously larger flunce attenuation of the thicker wall. The larger absorbed dose contribution of thermalised neutrons is documented by the larger photon component (<10 keV/µm) due to the capture reaction in hydrogen and the proton component around 100 keV/µm due to capture in nitrogen.

Several research groups in Europe and the U.S. have developed TEPC instruments for area monitoring and investigated the influence of design and operation parameters on the dose equivalent response and performance characteristics. In Europe the Eurapean Radiation Dosimetry Group (EURADOS) and the Physikalisch-Technische Bundesanstalt (PTB) organised a first intercomparison of different TEPC prototype area monitors with monoenergetic neutrons between 0.073 and 5 MeV and for a D_2O moderated ^{252}Cf-source[3]. The main results obtained to date in these investigations may be summarized as follows:

- Optimisation of the counter geometry, in particular the wall thickness is not sufficient to eliminate the problem of the too low response below 100 keV neutron energy[3,4]. Potentially useful parameters for further improvement of the response are the simulated diameter, the evaluation algorithm and modifications of the detector materials.

- Measurements with moderated ^{252}Cf-source have shown that the influence of detector geometry on the response for broad neutron energy spectra is much less pronounced than for monoenergetic neutrons[3,5]. There is evidence for the assumption that the too low response to monoenergetic neutrons below 100 keV of the TEPC is of limited relevance for the broad energy spectra encountered in radiation fields of practical importance.

- The calibration procedures with built-in alpha particle sources or external photon sources is adequate if the phantom-based definition of the quantity H*(10) is taken account of.

- The sensitivity of TEPC area monitors is adequate for practical purposes. It has to be pointed out, however, that the statistical uncertainty does not depend only on count rate but also on the pulse height distribution which has a wide dynamic range of more than four decades. he sensitivity therefore is a complex function of radiation quality, the photo dose fraction and, as can be seen from Fig. 2, of the detector geometry.

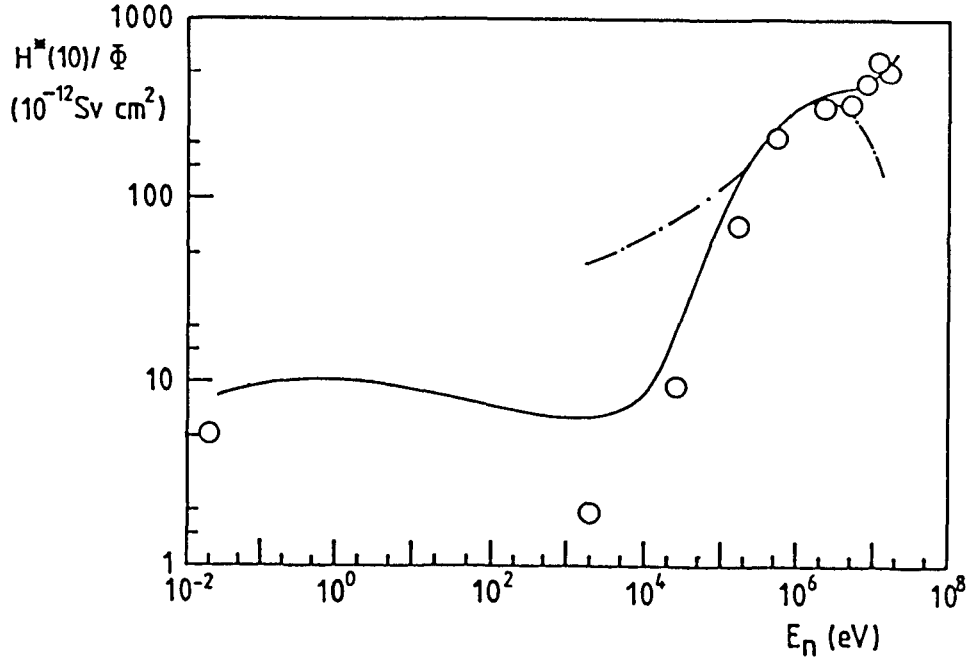

Figure 1: Flunce-to-ambient dose equivalent conversion function for neutrons (solid curve) (Wagner et al., Radiat.Prot.Dosim. **12**,231,(1985)) and experimentally determined TEPC dose equivalent reading (open circles). The dashed line is the response curve for a conventional neutron dose equivalent dose rate meter ("rem-counter").

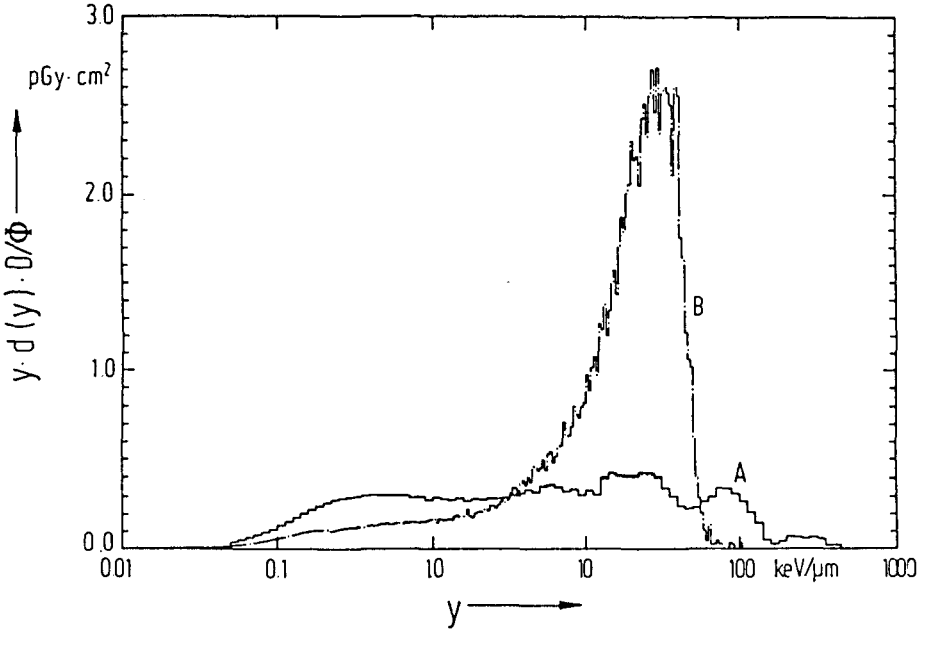

Figure 2: Microdosimetric spectra y·d(y) normalised to unit neutron fluence of a 73 keV moenergetic neutron beam measured at PTB. Spectrum A was measured with thick-walled TEPC (20 mm wall thickness), spectrum with a thin-walled counter[3].

PRACTICAL EXPERIENCES

Measurements of dose equivalent with TEPC have been made to date in the mixed radiation environments of high energy and medical accelerators, nuclear power plants and nuclear fuel processing plants. The results reveal the considerable variations in radiation quality between various facilities and in some cases within the environment of one facility. Variations of Q for neutrons between 4.3 and 10 were observed. The diagnostic information provided by the instrument enabled the interpretation of these variations. The neutron dose equivalent reading of the TEPC and of rem-counters in identical positions agreed well for neutron spectra moderated dy hydrogenous material as encountered at nuclear energy facilities. However, the results differed by factors of up to 3, for example at some locations in the environment of high energy accelerators. The eperiences gained are also of importance for neutron personnel dosimetry. The comparatively low energy dependence and the diagnostic information is expected to be useful in establishing local correction factors for currently used individual neutron monitors.

CONCLUDING REMARKS

The recent availability of modern microelectronics is the basis for the construction of portable dosemeters using spectroscopic methods. This enables the construction of portable TEPC based dose equivalent meters and the utilisation of their favourable properties in radiation protection routine. The final conceptual and technical development is still in progress. First commercial instruments, however, are available.

Acknowledgement: This report is based on work supported by the Commission of the European Communities (Contracts BI-G-0010-D and BIO-A-482-82-D).

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FACILITIES AND PROCEDURES FOR CALIBRATING A SELF-FABRICATED BORON-LOADED PERSONNEL NEUTRON DOSIMETER

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ABSTRACT

The inaccuracy of general personnel neutron dosimeters and more concern about the neutron damage to humans have brought about an intensive effort in designing a new type of personnel neutron dosimeter. Facilities and procedures for calibrating a self-fabricated boron-loaded neutron dosimeter are described here. An aluminum building designed and constructed by the Institute of Nuclear Energy Research was used as a neutron calibrating room.

Cf-252 was used as a primary neutron standard source. A variety of moderators were used to change the irradiated neutron spectra for testing the energy dependence of this new dosimeter. Dose equivalent rates of various spectra were evaluated by Bonner Multisphere spectrometer. Ratios of 9" to 3" sphere from BF_3 detector were determined as the energy correction indices.

This paper also discussed the unfold spectra as the number of energy group in Bon computer code was changed. Results of field measurements in LWR by this boron-loaded dosimeter were compared to those from multisphere spectrometer.

EXPERIMENTAL SIMULATION AND CHARACTERIZATION OF NEUTRON SPECTRA FOR CALIBRATION RADIATION PROTECTION DEVICE

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ABSTRACT

To improve the evaluation of dosimetric quantities, it is advisable to calibrate devices used for radiation protection in neutron spectra very similar to those met in practice (for instance in nuclear plants, plutonium technology laboratories, ...).

Among the neutron sources and monoenergetic neutron radiations recommended by the ISO for calibration, appears a lack of neutron sources with broad spectral distribution extending to lower energies ; spectra where thermal component represents an important part are for example not available.

To complete the range of neutron calibrating sources, it seems useful to develop several wide spectra distributions representative of typical spectra.

The prototype equipment of our laboratory is based on the moderation of fission neutrons by shields of different materials modifying the spectral distribution. The experimental set up is described and results are presented.

Two characterization techniques have been employed : on one hand, the Bonner spheres system of which the results are handled by the MORELPA program. This method is able to give spectral informations on the whole scale of neutron energy, but in a rough manner ; on the other hand, the spectrometry by recoil protons using spherical proportional counters and a NE 213 liquid scintillator gives accurate results between 5 keV and 20 MeV.

The dosimetric quantities deduced from these spectra are compared with direct Kerma measurements obtained with a tissue equivalent ionization chamber and a Geiger Müller counter.

USE OF AMERICAN AND TAIWANESE COINS AS QUICK SORT INDICATORS OF NEUTRON ACCIDENT EXPOSURE

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ABSTRACT

Nuclear accident dosimetry system usually include quick sort indicators and accident dosimeters to assess the accurate dose to personnel. A neutron accident exposure indicator is required for the rapid identification of neutron exposed personnel. Although the most common material are indium and dysprosium, coins possessed by workers also can be seen as an assistant indicator especially for unmonitored personnel. This paper describes the response characteristics of American and Taiwanese coins to four neutron energy spectra that could be encountered during a criticality accident.

Three American coins, Dime, Nickel and Penny, and two Taiwanese coins, $5 and $1, are exposed to neutron from Health Physics Research Reactor (HPRR) at Oak Ridge National Laboratory in U.S.A. Response characteristics of coins were obtained for four different neutron spectra produced by operating the HPRR with the following shield conditions; unshield, shielded with concrete, steel and lucite. In irradiation, coins were mounted at front, side and back of BOMAB phantom to evaluate directional dependence. Coin activities were measured by GM survey meter in units of dpm and dose rate as a function of time after the irradiation. Nuclides were analyzed by HPGe detector also.

Results of this study show that: (1) coin activities are strongly depend on the incident neutron spectrum, (2) directional dependence is serious, (3) Taiwanese $5 coins show the highest activity and penny is the lowest, and (4) all coins show distinguishable activities within 24 hours after exposed to 0.25 Gy of neutron of four different spectra. These results indicate the possibility to use coins as an neutron accident exposure indicators. This also provide unmonitored personnel the opportunity to receive immediate medical care. Absorbed neutron dose can also be estimated approximately with some limitations which will be discussed in this paper.

EMISSION RATE MEASUREMENT OF A Cf-252 NEUTRON SOURCE BY MANGANOUS SULFATE BATH METHOD

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1. INTRODUCTION

The manganous sulfate($MnSO_4$) bath method has been in use over two decades for the most accurate measurement of neutron emission rate. The $MnSO_4$ bath has been developed accordingly and the bath system has been operated for the standardization of neutron measurements at the Korea Standards Research Institute(KSRI) since 1986. There are three types of the radioactive neutron sources: (γ,n), (α,n) and spontaneous fission. In this research work, a spontaneous fission source, Cf-252, as the reference neutron source calibrated by the standard neutron source, Am-Be, is introduced to determine the characteristics of the bath system in terms of the bath efficiency ($\bar{E}$), the saturated manganese activity (A_S), the inverse manganese(^{55}Mn) neutron capture fraction (f^{-1}) and eventually the bath system correction factor (C_F).

2. PRINCIPLE OF THE METHOD

Figure 1 shows the events which take place in the $MnSO_4$ bath during a Cf-252 neutron source calibration. The neutron source is

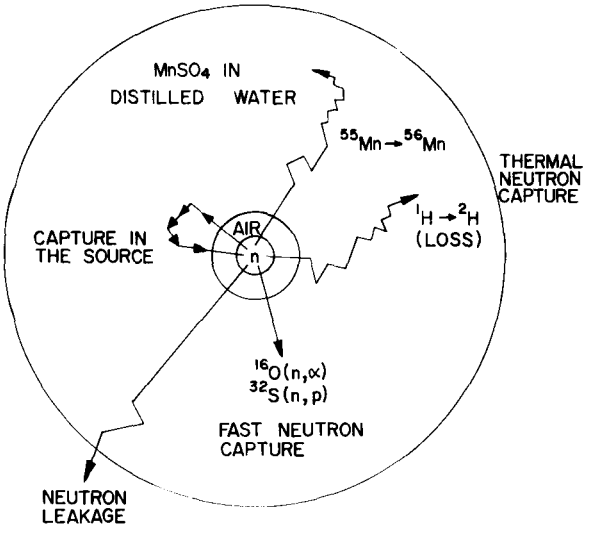

Fig. 1. Schematic Diagram of Principle of $MnSO_4$ Bath Method

mounted at the center of a spherical bath containing an aqueous solution of manganous sulfate. The solution is continuously stirred by the impellers and at the same time pumped in a closed circulation circuit through a detection system. Figure 2 shows the schematic arrangement for the bath system. Neutrons from the neutron source are thermalized in the $MnSO_4$ solution and captured by the various nuclei. The ^{56}Mn resulting from the capture of neutron by ^{55}Mn is measured by the counting system in the bath system.

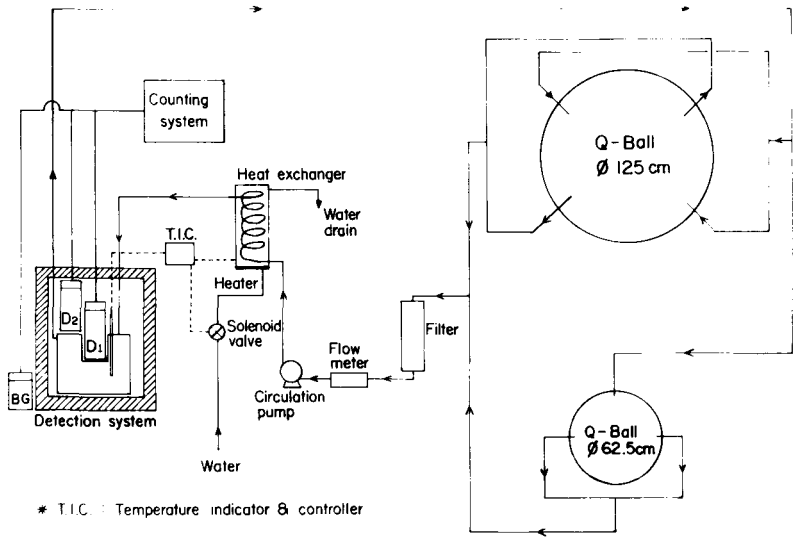

Fig. 2. Schematic Diagram of $MnSO_4$ Bath Circulation System

The neutron emission rate of a Cf-252 source, Q, is given by

$$Q = (A_s/E) \cdot (f^{-1}) \cdot C_F \qquad (1)$$

where f is the fraction of neutrons captured by ^{55}Mn. In the existance of impurities in the solution, the inverse ^{55}Mn capture fraction becomes(1)

$$f^{-1} = 1 + (N_{Im}/N_{Mn}) \cdot (\sigma_{Im}/\sigma_{Mn}) \cdot ((1 + \bar{r}s)_{Im}/(1 + G\bar{r}s)_{Mn}) +$$

$$(1 + (N_{Im}/N_{Mn})) \times (\sigma_S/\sigma_{Mn}) \cdot ((1 + \bar{r}s)_S/(1 + G\bar{r}s)_{Mn}) +$$

$$(N_H/N_{Mn}) \cdot (\sigma_H/\sigma_{Mn}) \cdot (1/(1 + G\bar{r}s)_{Mn}) \qquad (2)$$

where N_{Mn}, N_H, and N_{Im} are the respective concentrations of manganese, hydrogen and impurities in the solution. The factor $1 + G\bar{r}s$ allows for the resonance capture in manganese. In the notation of Westcott et al.(2), s is the resonance activation integral, $\bar{r}$ is the the epithermal flux parameter averaged over the $MnSO_4$ bath and G is the resonance self-shielding factor for the solution. Rearranging eq. (2), a straight line is fitted to the data(3).

3. EXPERIMENTAL PROCEDURE

3.1 Measurement of Bath Efficiency, E

A quantity of radioactive ^{56}Mn of high purity is divided into portions by weight, some of which are used to activate the bath and others are absolutely measured by the $4\pi\beta$-γ coincidence counting technique to determine the specific activity. To obtain the efficiency, the mean counting rate of ^{56}Mn γ-rays measured by the $MnSO_4$ bath counting system and corrected for decay back to a suitable reference time is divided by the ^{56}Mn activity measured absolutely and corrected to the same reference time. Since this quantity is concentration dependent, E is determined at least once for each concentration.

3.2 Calculation of Saturated Manganese Activity, A_s

The ^{56}Mn activity produced during neutron irradiations in the $MnSO_4$ bath is counted by the two 1.5" ϕ x 1.5" NaI(Tl) scintillation detectors equipped in a 12.5-liter detector bath of s.s. Marinelli beaker-type. A third detector monitors the room background. The saturated activities are calculated using the computer program taken from the Gilliam model(4) in the growth and decay phases of ^{56}Mn activity. The parameters used in the calculation are the total number of counts (C_i) observed during the counting period which began at a time after insertion of the neutron source into the bath, the duration of the neutron irradiation, the decay constants of ^{56}Mn and Cf-252 and the mixing constants.

3.3 Calculation of Inverse ^{55}Mn Neutron Capture Fraction, f^{-1}

To obtain the inverse ^{55}Mn capture fraction from eq. (2), the necessary data are taken from Mughabghab et al.(4) and Hwang et al. (5). The N_H/N_{Mn} ratio is determined by the gravimetric method by drying and weighing weighed samples of the $MnSO_4$ solution.

4. EXPERIMENTAL RESULTS

The neutron emission rate of the reference source, Cf-252, calibrated by the comparative method to the standard source, Am-Be is determined to be 2.232×10^7 n/s at the reference time, 12:00 on November 15, 1987(6). The bath efficiency and the inverse ^{55}Mn neutron capture fraction, the saturated manganese activity are given in Table 1 and Table 2, respectively.

According to the data, the following empirical formulae as a function of N_H/N_{Mn} are derived by the least squares fitting method:

From Table 1, $Y_E = X^{-4.348}$ and $Y_{f-1} = 0.024842\ X + 1.025252$

From Table 2, $Y_G = -0.8270\ X + 166.3797$ (Growth phase of ^{56}Mn)

$Y_D = -0.8292\ X + 165.5645$ (Decay phase of ^{56}Mn)

Substituting these results into eq. (1), the bath system correction factor, C_F is obtained at a reference time.

Table I. Bath Efficiency, E and Inverse ^{56}Mn Neutron Capture Fraction, f^{-1}

X (N_H/N_{Mn})	E (10^{-5} cps/dps)	f^{-1}
45.034295	—	2.147862
58.191625	1.433550	2.470228
71.586090	1.502855	2.798636
94.411795	1.457515	3.370241
111.04887	1.476193	3.786437

Table 2. Saturated ^{56}Mn Activity, A_S

X (N_H/N_{Mn})	A_S (cps) in Growth Phase	A_S (cps) in Decay Phase
45.034295	133.8850	131.20900
58.191625	115.32310	—
71.586090	102.67140	101.21610
94.411795	89.45201	—
111.04887	76.07417	75.49648

5. CONCLUSION

From the graphic analysis of data obtained by the manganous sulfate bath method, the bath system correction factors to the neutron emission rate can be determined as a function of the N_H/N_{Mn} ratio. Figure 3 shows the straight lines fitted to the data in order to determine the unknown correction factor of a Cf-252 neutron source at the given ratio of N_H/N_{Mn} in a $MnSO_4$ solution. In comparison of the line in ^{56}Mn growth phase with the line in ^{56}Mn decay phase, the difference of the correction factors between them turns out to be a range of 0.77 % to 1.44 % to the growth phase activity.

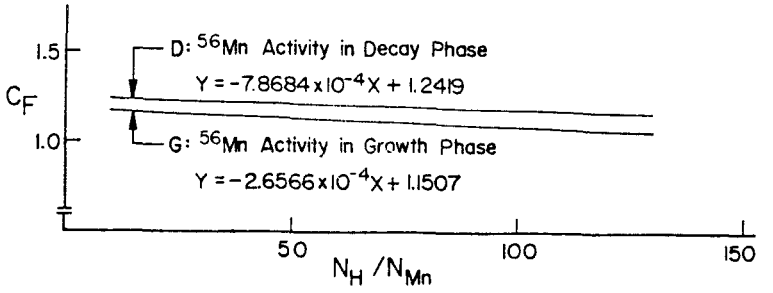

Fig 3. Bath System Correction Factor

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ACTIVITIES OF THE INTERNATIONAL NON-IONIZING RADIATION COMMITTEE OF IRPA

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ABSTRACT

The work achieved by the International Non-Ionizing Radiation Committee of IRPA since the 6th International IRPA Congress in May 1984 will be briefly reviewed. Stress will be laid more particularly on the new problems which had to be dealt with.

Following important progress made in research on the interaction of radiofrequency radiation with biological systems, particularly in the field of dosimetry, the Committee amended on several points the "Interim guidelines on limits of exposure to radiofrequency electromagnetic fields" which were published in April 1984.

In recent years, concern developed, in a number of industrialized countries, about the health risks which might be associated with the increasing propagation of electric fields, for instance from high power transmission lines, and with the expanding use of static and slowly varying magnetic fields in advanced technologies and medical applications. The presently available data and conclusions reached by an international group of experts, as contained in UNEP/WHO/IRPA Environmental Health Criteria documents 35 for Extremely Low Frequency Fields (1984) and 69 for Magnetic Fields (1987), will be summarized. Some difficulties involved in the drafting of internationally acceptable recommendations in these fields will be outlined.

CARCINOGENIC POTENTIAL OF NON-IONIZING ELECTROMAGNETIC RADIATION

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INTRODUCTION

The mutagenic and carcinogenic potential of the ionizing radiations such as X-rays, gamma rays, alpha and beta particles are well known and their mechanisms of interaction with living organisms are well characterised. Aside from ultraviolet light producing various skin cancers, the carcinogenic ability of other non-ionizing radiations (visible, infrared, microwaves, radiofrequency, extremely low frequency, static electric and magnetic fields) had, until recently, been dismissed.

In Australia, general public and employee concerns about exposure to electromagnetic radiations, predominant at power frequencies (50/60Hz) have been raised to such a level that electric power utilities have had very significant resistence to the construction of new installations. Uninformed media coverage and lack of knowledge about the difference between established and unsubstantiated biological effects has contributed to the concerns.

This paper will briefly review experimental data as well as cancer registry and epidemiological studies that suggests exposure to extremely low frequency (50 or 60 Hz) fields may be associated with an increased incidence of leukaemia, brain tumours, or other forms of cancer.

Laboratory Studies

An in-depth review of the interaction mechanisms, biological effects and health risk assessment of ELF electric and magnetic fields has been published (WHO, 1984, 1987). Studies on Drosophila and animals have indicated that exposure to 50/60Hz fields do not produce mutagenesis (Krueger et al, 1975; Bender, 1976; Phillips et al 1979; Frazier et al 1982). It is generally believed that 50/60Hz fields are not primary inducers of cancer, however studies are continuing to determine if these fields act as cancer promoters (Adey, 1986). To date, no laboratory evidence has been produced to substantiate this.

Human Studies

In 1979 Wertheimer and Leeper published their study that came to the then remarkable conclusion that children resident near high current configurations (transformers; large guage, low-voltage, high-current power lines; power substations etc.) had a two-to three-fold higher incidence of cancer than children resident near low current configurations. This study, although having a number of limitations common to many epidemiological studies, was sufficiently well conducted that interest was raised and a large number of subsequent studies have resulted.

Table 1 gives a summary of the studies on cancer incidence and occupational exposure to electromagnetic fields. Of the 20 studies, about half suggest an association between exposure to these fields and an increased incidence of leukaemia. The other studies suggest no association or an associated increased incidence of other cancers. Armstrong (1986) suggests that although the balance of studies favours an association between work in electrical occupations and leukaemia, it is possible Milham's (1982) observation caused a number of investigator's with access to occupational cancer incidence or mortality data to examine their data, and that the "positive" results were, in the main, the only ones reported.

Table 2 summarises the studies on cancer incidence and population exposure to electromagnetic fields. Only 3 of the 11 studies suggest an increased leukaemia risk. It should be noted that the category of study design is indicated in Tables 1 and 2. The proportionate mortality ratio (PMR) or proportional incidence ratio (PIR) studies provide the most limited and potentially flawed data, case-control (CC) studies provide better results, while the retrospective follow-up (RFU) studies are the most methodologically sound (Cole, 1987). One notes that most of the RFU studies do not suggest an increased incidence of leukaemia with electromagnetic field exposure.

Cole (1987) claims the epidemiological evidence is weak because: Studies suggesting an association between lung cancer and smoking, or mesothelioma and exposure to asbestos fibres, were very consistent - the causal factors were never repudiated, and large risk ratios were identified. None of these features occur in the electromagnetic field studies. The two major studies by Wertheimer and Leeper (1979, 1982) were not conducted blind, leading to the distinct possibility of bias. Both birth and death addresses were used giving different answers depending on which were used. Both studies contain an "inordinate consistency." All but one of about 30 comparisons contained relative risks in the range 2.0 to 3.0. Cole (1987) calculated the probability that the studies could be so consistent as less than 1 in 100. The two studies claim that electromagnetic fields from high current configurations (suggesting magnetic fields as the causative agent) produces a broad spectrum of different concerns - a broader spectrum than ionizing radiation or polyoma virus. This stretches credibility. Cole concluded the studies could only be biased. No other study approaches these for strength or consistency of findings despite better methodology.

The best of all studies conducted thus far is from Savitz (1986b). Cole claims this is a small study because there are nearly as many risk ratios below 1.0 as above, and there is no consistent dose-response. Savitz data shows a tendency to an isolated high risk ratio in the highest exposure category. A reasonable explanation for this result almost certainly lies in the acknowledged poor cooperation from controls leading to an under-representation in controls in the VHCC (very high current configuration) category.

Table 1
CANCER INCIDENCE & OCCUPATIONAL EXPOSURE TO ELECTROMAGNETIC FIELDS

REFERENCE	WORKERS	STUDY	CANCER RISK
Wiklund et al., 1981	Telecom	PMR	No cancer risk.
Milham, 1982, 1985b	Electrical	PMR	Increased leukaemia.
Wright et al., 1982	Electrical	PIR	Increased leukaemia.
McDowall, 1983	Electrical	PMR/CC	Increased leukaemia.
Coleman et al., 1983	Electrical	PIR	Increased leukaemia.
Vagero & Olin, 1983	Electrical	RFU	No leukaemia risk. Increased pharangeal, respiratory cancer.
Swerdlow, 1983	Electrical	PMR	Increased eye melanoma.
Pearce et al., 1985	Electrical	CC	Increased leukaemia.
Lin et al., 1985	Electrical	CC	Increased brain tumours.
Milham, 1985a	Amateur radio	PMR	Increased leukaemia.
Gilman et al., 1985	Miners	RFU	Increased leukaemia.
Vagero et al., 1985	Electrical	RFU	No leukaemia risk. Increased urinary cancer, Malignant melanoma.
Calle & Savitz, 1985	Electrical	PMR	No leukaemia risk.
Olin et al., 1985	Electrical	RFU	Increased malignant melanoma.
Stern et al., 1986	Electrical	CC	Increased leukaemia
Tornquist et al., 1986	Electric power	RFU	No leukaemia risk. No brain tumour risk.
Flodin et al., 1986	Electrical	CC	Increased leukaemia
Thomas et al., 1987	Electrical	CC	Increased brain tumours.
McLaughlin et al, 1987	Electrical	PIR	No brain tumour risk.
Lin, 1987	Electric power	CC	Increased cancer risk.

Table 2
CANCER INCIDENCE & POPULATION EXPOSURE TO ELECTROMAGNETIC FIELDS

REFERENCE	SUBJECTS	STUDY	CANCER RISK
Wertheimer & Leeper, 1979	Children resident near HCCs.	CC	Increased leukaemia.
Fulton et al., 1980	Children resident near HCCs.	CC	No increased leukaemia.
Wertheimer & Leeper, 1982	Adults resident near HCCs.	CC	Increased cancer.
Coleman et al., 1985	Persons resident near HV.	CC	No increased leukaemia.
Myers et al., 1985	Children resident near HV.	CC	No increased cancer.
Rodvall et al., 1985	Persons resident near HV.	CC	No increased cancer.
Tomenius, 1986	Children resident near HV.	CC	Increased CNS tumours. No increased leukaemia.
McDowall, 1986	Persons resident near HV.	RFU	No increased leukaemia.
Savitz, 1986b	Children resident near HCCs.	CC	Increased cancer. Increased leukaemia.
Stevens et al., 1986	Persons resident near HCCs.	CC	No increased acute non-lymphocytic leukaemia.
Coleman & Bell, 1987	Persons resident near HV lines & sub-stations.	CC	No leukaemia risk.

The World Health Organisation (WHO, 1987) concluded that "These associations cannot be satisfactorily explained by the available theoretical basis for carcinogenesis by ELF electromagnetic fields. The preliminary nature of the epidemiological evidence, and the relatively small increment in reported incidence, suggest that, although these epidemiological data cannot be dismissed, there must be considerable further study before they can be accepted."

CONCLUSIONS

At this point in time, one would have to agree with the conclusions of Savitz (1986a) when he stated that "The findings of human studies of 50/60 Hz electric and magnetic fields are not sufficiently indicative of health hazards to warrant exposure limitation, nor is there high quality, convincing research to exonerate these exposures as threats to human health." "Given this array of incomplete information, there is not an empirical basis yielding an answer to the critical question of human health consequences of 50/60Hz electric and magnetic field exposures.

Using standard levels of scientific proof, the argument that these fields cause cancer, reproductive damage, or other health effects falls far short of convincing. From the perspective of public health protection, however; one might ask whether the suggestions of health effects raised in some studies have been convincingly negated by superior research. The answer is that they clearly have not. The social and political choices relating to this issue fall outside the bounds of science, but there is a major role for further epidemiologic study in providing the basis for those decisions."

One of the major arguments against there being an association between electromagnetic radiation exposure and increased incidence of cancer, is the fact that laboratory studies have not identified a mechanism by which these radiations at 50/60Hz could either induce or promote cancer.

In the final analysis, if an increased cancer risk is definitely found to be associated with chronic low-level exposure to 50/60Hz fields, the risk must be quantitated (presumably increasing risk with higher exposures) so that a risk-benefit analysis can be completed. If, for example, chronic exposure to 50 Hz fields at a given low level increased the risk of contracting cancer from 1 in 1,000 to 3 in 1,000 people must determine whether they wish to reduce their use of electricity in order to lessen this risk. To date, chronic low-level exposure to 50/60Hz fields has not been established to increase the risk of contracting cancer. If it is established in the future, it would in all likelihood be a very weak carcinogen.

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HAZARD CONTROL MEASURES FOR LASERS

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INTRODUCTION

Considerable attention has been paid to the establishment of exposure limits (EL's) for laser radiation [1,2,3], but it has been generally not well recognized that EL's are infrequently measured in actual hazard evaluations of laser installations. Inasmuch as many laser beam irradiances are at least a thousand-fold higher than applicable EL's, a very careful measurement is seldom needed [4]. A movement of a measuring instrument a few mm from the beam axis may result in completely missing the beam. Recognition of these difficulties led to an approach in laser safety that departs from the methods followed in evaluating and controlling ionizing radiation sources.

It was recognized in the 1960's that regardless of detailed measurements certain similar types of lasers repeatedly required the same hazard control measures regardless of the specific application and environmental setting [5]. This recognition led to the development of the laser hazard classification scheme now almost universally applied throughout the world. Despite some small differences in national standards, the basic concepts embodied in the hazard classes (i.e., Classes 1, 2, 3A, 3B and 4) are the same as in Standard 825 of the International Electrotechnical Commission [6].

LASER HAZARD CLASSIFICATION

Lasers fall into one of four general hazard classes. The basic concept is that a laser's potential hazard is defined by wavelength and output power/energy. A Class 1 laser product is not hazardous--either by virtue of an extremely low-powered output (e.g., some low-powered injection diode lasers) or from the use of an enclosure.

A Class 2 laser product is limited to a laser which emits visible light between 400 nm (violet) and 700 nm (red) and is sufficiently bright (up to 1 mW power) to evoke an aversion response to bright light (i.e., blink reflex with eye and head movement). The hazard of looking into a beam ("intrabeam viewing") of a Class 2 laser is similar to that of staring at a welding arc or looking into a slide projector beam. Forced staring into such bright lights for minutes to hours can result in a photochemically initiated retinal injury (equivalent to eclipse blindness). The output power of a Class 2 laser must be less than 1.0 mW--a level which equates to the permissible occupational exposure limit of 2.5 mW/cm^2 over a 7-mm diameter pupil for a 0.25-s exposure (the aversion, or blink-reflex exposure time). One should consider Class 1 lasers "safe" and Class 2 lasers "safe for all practical purposes," since Class 2 lasers pose only a theoretical hazard.

Class 3 and 4 laser products are potentially dangerous. Class 3 lasers can injure the eye upon direct intra-beam viewing. Class 3 is divided into 3A and 3B; Class 3A is a transitional class which includes lasers that pose a hazard only under certain worst-case conditions. A Class 3A laser product is generally a 1 to 5 mW visible laser (most commonly a He-Ne, 632.8-nm laser). In all countries except the USA, Class 3A lasers are limited in output irradiance to the EL for $\frac{1}{4}$ s (i.e., 2.5 mW/cm^2) [6-8]. Class 3 laser products are hazardous to the eye from intrabeam exposure--even for brief exposures less than 0.25 s. Class 4 laser products, like Class 3, have a warning label with the familiar starburst logo representing laser radiation (Figure 1); and unlike Class 3 lasers, they may pose a significant fire hazard or skin burn hazard, and pulsed lasers may pose a beam reflection hazard even from diffuse surfaces. The hazard classification is marked on the safety label by the manufacturer (since 1976) and the classification scheme is complicated. However, for the visible lasers used by Itek, those less than 1.0 mW output power are Class 2; those with an output power of 1.0 to 500 mW are Class 3; and those with an output power exceeding 0.5 W are Class 4. Any continuous-wave (CW) lasers in the infrared or UV are Class 4 if their output power exceeds 0.5 W average. Hence, most CO-2 lasers in the infrared are Class 4. Most excimer lasers which emit in the UV, and many argon lasers are Class 4 since their average output power normally exceeds 0.5 W, or the output radiant exposure exceeds a given limit. Visible and near-infrared lasers (400-1400 nm) may cause retinal injury; whereas, far-infrared and UV lasers may injure the cornea or lens.

HAZARD CONTROL MEASURES FOR THE USER

For Class 2 lasers, one need only caution the uninitiated not to stare into the light source, and treat it with caution as one would a bright movie projector or spotlight. For Class 3 lasers, persons should be instructed to wear laser eye protectors or enclose the beam or use baffles to preclude direct intrabeam exposure to the primary beam or specular reflections. Class 2 laser power levels should be used for alignment where feasible, and intrabeam exposure must be prevented if Class 3 lasers must be used for alignment unless the beam has been expanded to reduce the beam irradiance below 2.5 mW/cm^2. Laser eye protectors are particularly important for Class 3 pulsed lasers and for invisible (IR or UV) laser beams.

For Class 4 lasers, the requirements for Class 3 should be followed, and in addition eye protectors must be worn if any open beam is accessible. Most laser eye injuries have occurred from reflected beams from Class 4 lasers. The greatest "offender" has been the Neodymium:YAG Q-switched laser systems. Entryway door interlocks may be required for Class 4 laser laboratories to protect persons entering during laser operation unless the beam is well baffled or enclosed to greatly reduce the Nominal Hazard Area (or Zone). The skin should not be exposed to Class 4 laser beams. Entryway doors should have a standard warning symbol (Figure 1) and sign indicating the hazard and appropriate precaution to take (e.g., "Do Not Enter When Red Light is On, "Knock

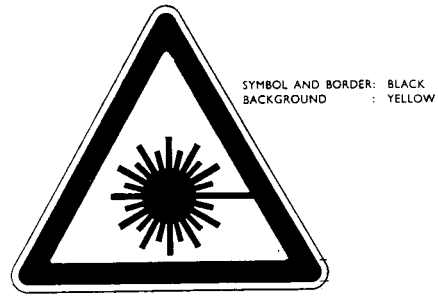

FIGURE 1. Laser Radiation Warning Symbol [6]. Detailed precautionary guidance and hazard classification is found in panel below the triangle.

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before Entering," etc.). A warning symbol without meaningful instructions for the reader is useless. For Class 3B and 4 laser systems emitting between 300 and 4,000 nm (where glass transmits) windows should be non-existant or covered with a light-tight, opaque screen. In this regard, many plastic "opaque" curtains actually transmit near-infrared (e.g., 1064 nm) wavelengths.

SYSTEM SAFETY HAZARD CONTROL MEASURES

Most countries have regulations for laser manufacturers which mandate certain system safety features depending upon hazard classification. These generally follow the IEC standard 825 [6]. For example, in the USA, a manufacturer of any commercial laser product must comply with provisions of Title 21, Code of Federal Regulations, Part 1040, Performance Standard for Laser Products [8]. The manufacturer must certify that his laser product meets these requirements, and must file documents which detail this certification with the Center for Devices and Radiological Health (CDRH) of the Food and Drug Administration (FDA), Rockville, MD.

The laser system safety features required by such performance standards which are applicable to most commercially constructed research and industrial Class 3b and 4 lasers are: (1) an interlocked or secured protective housing, (2) a remote connector which can be used to interlock an entrance door, (3) a key operated switch, (3) an emission indicator, such as a pilot light, (4) a beam attenuator, e.g., a mechanical shutter, (5) specified warning labels, (6) protective viewing optics (i.e., by a filter or shutter system), and (7) operator controls located to limit the chance for exposure. In addition to these general requirements, all medical laser products must also comply with three other requirements: (1) a means to measure the output within ±20 %, (2) a measurement calibration schedule, and (3) a "laser aperture" label. In some instances, a self-monitoring fixed laser output is considered to fulfill the first medical requirement. In the USA, manufacturers can obtain variances from

the above standards if alternate and effective controls are provided. A few controls (e.g., the protective housing), are also required for laser products assigned to lower classes.

It is important that each person working with a Class 3B or 4 laser product understand that the system safety controls are only helpful if used, and cannot be relied upon alone without other user control measure. Airborne contaminants must be evaluated [9]. Controls (e.g., filters in viewing optics) must be checked to be in place after servicing. Use of the key switch can prevent unauthorized use of the device by untutored persons. Since hazardous reflections and high voltages are accessible within the protective housing, the interlocked or screw-fastened enclosure should not be tampered with except by a well trained serviceman.

MEDICAL SURVEILLANCE

Medical Surveillance is not a control measure. At one time, many regulations required that all employees working with lasers have pre-placement and periodic eye examinations. However, today some examinations remain only for legal reasons [10]. But, if an accidental over-exposure to laser radiation occurs, it is critically important that an examination be performed within 1 - 2 days following the accident or suspected accident [10].

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RISKS OF CHRONIC EXPOSURE OF THE EYE BY OPTICAL RADIATION

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ABSTRACT

In recent years a growing body of data has accumulated in relation to the effects of optical radiation on the incidence and chronology of age related ocular pathologies. Chronic exposures to optical radiation result in both artificial and natural sources, but in practice solar radiation is by far the most significant. The solar exposure of the eye varies with geographic location, the altitude, the time of day and the geometry of the exposure. The absorption sites within the ocular tissues are dependent upon the wave length of the incident radiation. Each tissue of concern in the eye selectively absorbs specific wavelengths, thus precluding these from falling on the next underlying tissue along the path length. The actionspectra of eye pathologies will be discussed in relation to tissue, age, and specific waverlengths absorbed.

SOLAR RADIATION AND THE ULTRAVIOLET RADIATION EXPOSURE STANDARD

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INTRODUCTION

Partly as a result of increased concern over the possible depletion of stratospheric ozone and concomitant biological effects such as an increased incidence of skin cancer, ARL is extending the scope of its solar ultraviolet radiation (UVR) measurement programme. The results obtained thus far will be presented and their implication for the outdoor worker will be discussed in terms of the Australian occupational UVR exposure standard and the need to change existing work practises and personal protection.

EXPERIMENTAL METHOD

Solar UV spectral measurement are made at Yallambie (37.8°S) using a microcomputer controlled spectroradiometer incorporating a Spex 1680B double monochromator (DM). The DM contains two 50x50mm, 1200grooves/mm gratings blazed at 250nm, resulting in a dispersion of 1.8nm/mm and a resolution of 0.2nm. The detector is an EMI 9653QA end-on PMT which is held at -10°C. The current is measured with a Keithley 616 digital electrometer. The typical dark current is 50 pa which results in a noise equivalent spectral irradiance of approximately $1\mu W.m^{-2}.nm^{-1}$ at 300nm. The optical input involves a 100mm integrating sphere coupled to the DM entrance slit by a 1m liquid light guide. The sphere is raised into position through a hatch in the laboratory roof and global radiation measurements are made at solar noon on cloud-free days. The diffuse component of this radiation is also determined by making alternate measurements with an occulting band in place. Scans are made at 1nm intervals over the range 280-400nm with a bandwidth of 1nm. At each wavelength data is automatically collected and averaged until the required (usually 1-10%) standard deviation is met. Calibrations are performed in-situ using a calibrated 100W quartz halogen lamp. All data is stored on disk for eventual transfer to the ARL computer.

Solar UVR is monitored continuously with an Eppley UV radiometer. This has an interference filter which limits the spectral response to the 295-385nm wavelength interval. The UVB component is measured with an International Light IL700A UVB (280-315nm) radiometer. Total sun and sky radiation are measured with an Eppley precision spectral pyranometer. This has a spectral response covering the 0.285-2.8 μm region. All three radiometers are interfaced to a datalogger which stores data as 10 minute averages. This data is transferred weekly to the ARL computer for analysis. Three further broad-band detection units have been constructed and these will be placed at Brisbane (27.5°S) and Mackay (21.1°S) (both on the east coast of

Australia) during early 1988. A detector with an 'erythemal-like' response will also be included in these units.

RESULTS AND DISCUSSION

Typical spectral power distributions (SPD) for solar UVR are shown by log and linear plots in Figure 1 for four days representing the four seasons of 1987. The measurements are of global radiation (direct plus diffuse) incident on a horizontal plane. The log plot shows the dramatic increase in spectral irradiance through the UVB region (five orders of magnitude over a 20nm wavelength interval). The changing onset with time of year is also evident (from 287 to 292nm). The daily and annual variation of UVR and UVB is given in Figure 2. In summer during the five hours centred on solar noon 59 and 70% of the daily total of UVR and UVB respectively is received -this has important implications for the outdoor worker.

In determining exposure doses it is important to know the direct and diffuse components of the global radiation. Figure 3 shows the variation of the solar ratio with wavelength and SZA during 1987. Measurements were made at solar noon on the indicated dates. At low SZA's the wavelength variation of the ratio may be predictable for given atmospheric conditions and this would allow the determination of the diffuse component from the calculated direct component and a few measurements of the ratio at selected wavelengths [1]. This would provide spectral information for data collected at remote locations.

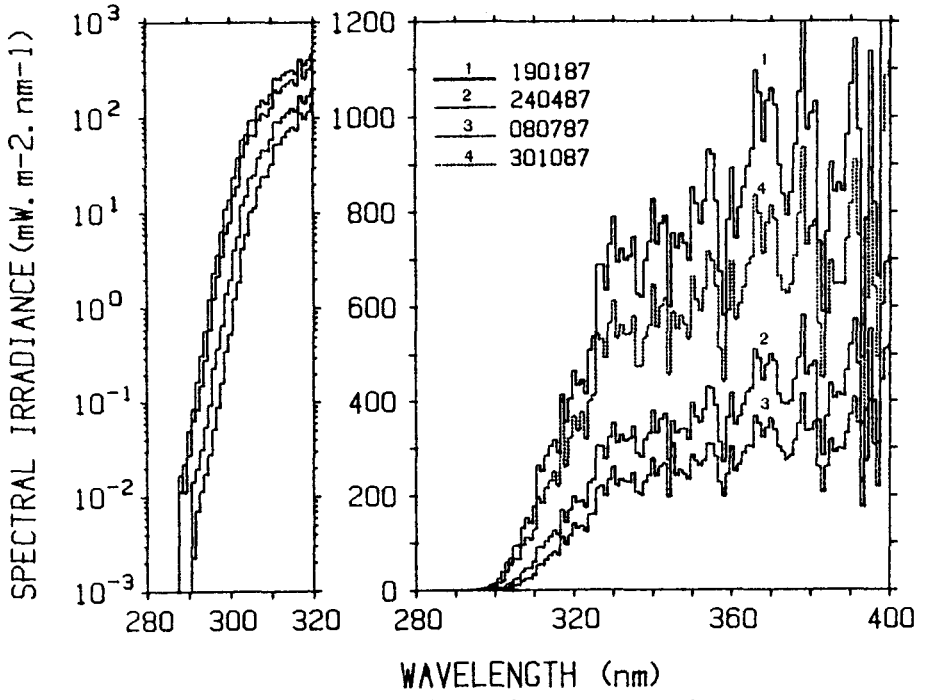

Figure 1. Solar spectral irradiance distributions for four representative days in 1987.

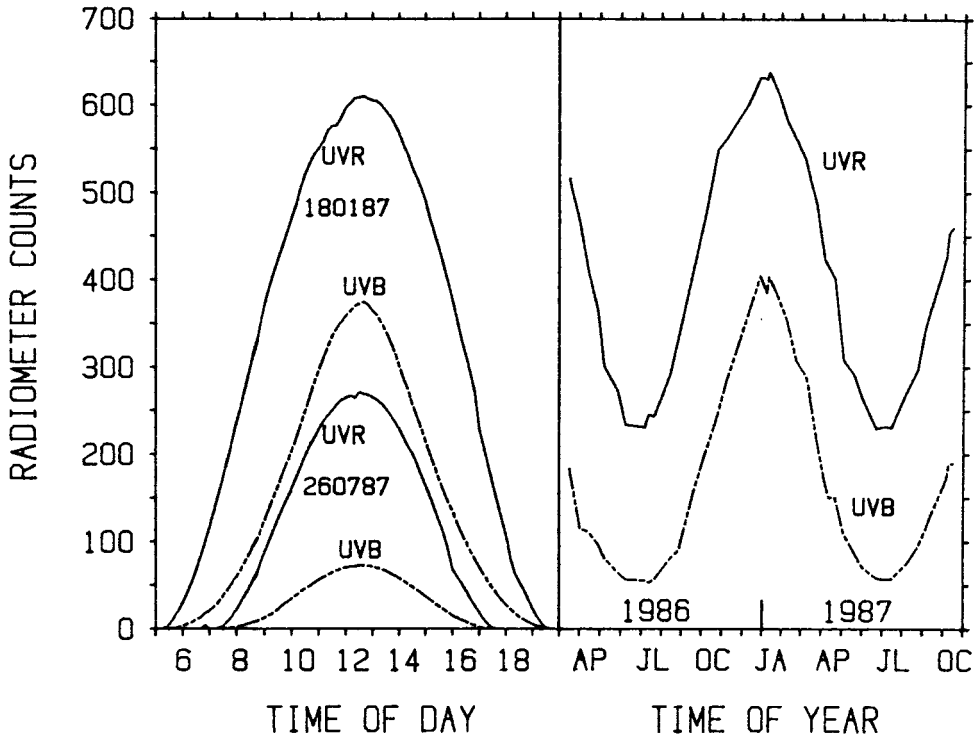

Figure 2. Daily and annual variation of UVR and UVB. Note that the radiometers have different calibration factors.

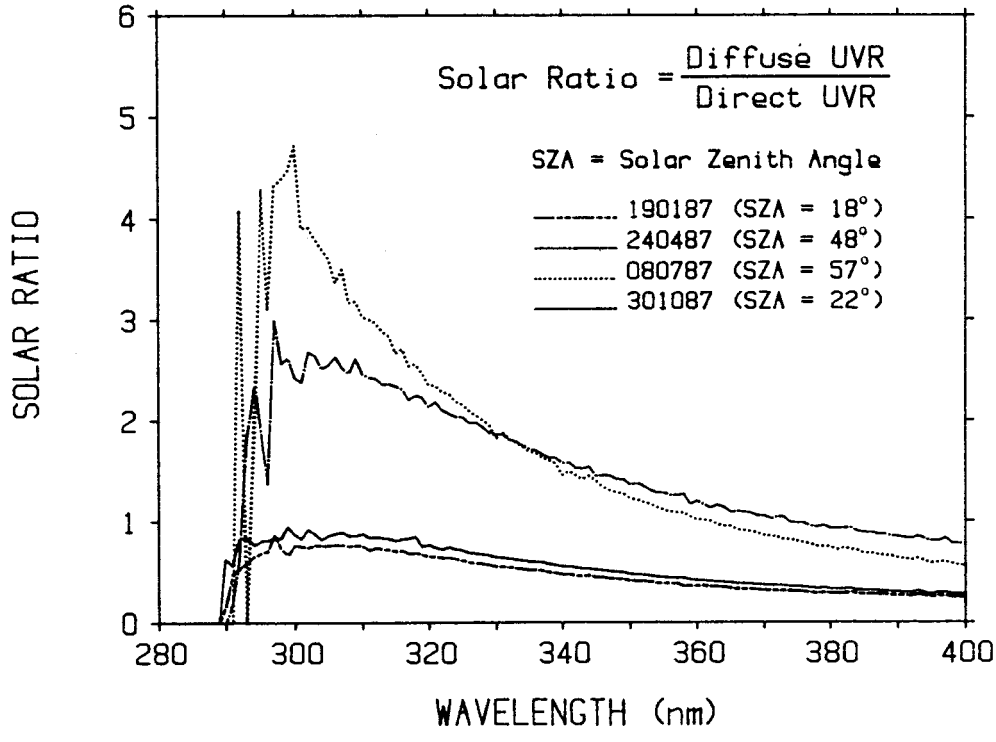

Figure 3. Variation of solar ratio with SZA.

Exposure Standard

The Australian National Health and Medical Research Council recently adopted the IRPA UV Occupational Exposure Limits [2] for Australian workers. For an 8-hour working day these are :-

1. UVA radiation (315-400nm). The total irradiance incident upon the unprotected skin or eye should not exceed 10W.m^{-2} for periods greater than 1000s.
2. UVC and UVB radiation (180-315nm). An effective irradiance (E_{eff}) of 30J.m^{-2} is recommended. The E_{eff} for a broadband source is calculated by weighting the SPD by the spectral effectiveness curve [2].

Table 1 gives maximum irradiance data for four representative days of 1987 as well as the maximum allowable exposure time (t_{max}) for the UVB component calculated according to the IRPA guidelines [2]. It is immediately obvious that the unprotected outdoor worker is exposed to UV irradiances that exceed the standard throughout the entire year and can receive the maximum permitted daily dose in less than 10mins in summer.

Table 1

Date(1987)	UVA(Wm^{-2})	UVB(Wm^{-2})	E_{eff}(mWm^{-2})	t_{max}(min)
Jan 19	67.6	2.3	73.4	6.8
April 24	31.5	0.8	16.5	30.3
July 8	22.7	0.4	7.3	68.5
Oct 30	51.9	1.7	48.4	10.3

Protection

The incidence of nonmelanoma skin cancer is known to be high amongst Australians, especially Queenslanders, although scant information is available for outdoor workers. Research at ARL and elsewhere has demonstrated the effectiveness of broadrim hats, clothing, sunglasses and sunscreens in reducing solar UVR to acceptable levels. It remains the responsibility of management and the unions to provide the encouragement and the means for outdoor workers to protect themselves.

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ALLEGED RADIATION RISKS FROM VISUAL DISPLAY TERMINALS

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A number of careful scientific studies have been focussed on the measurement of electromagnetic radiation or fields due to VDTs based on the cathode ray tube technique (CRT), while limited attention has also been given acoustic radiation (see reviews, e g Bergqvist 1984 or Guy 1984). The discussion as to whether work at VDTs can affect human health has been centered on different types of effects such as eye damage or discomforts, neck and shoulder discomfort, adverse reproductive outcomes, skin disorders and different stress reactions. In the present paper a short review is given of some of the alleged radiation hazards from the VDTs, mainly with emphasis on pregnancy outcome.

THE CLUSTER PHENOMENON

The discussions as to whether VDT work may influence pregnancy outcomes among operators originated with the observation that some groups of pregnant women had an unusually high frequency of miscarriages or birth of malformed children. Several investigators have found, however, that the reported number of such clusters is not higher than what is to be expected; although the occurrence of such a cluster is a very unusual event, the number of pregnant VDT workers is so large that also such events should be expected to occur, with a certain (low) frequency. Thus, a plausible explanation of the occurrence of these clusters is that of random distribution of adverse pregnancy outcomes - which will result in a low frequency of clusters due to chance without any VDT-specific causal factors being involved.

THE POSSIBILTY OF TERATOGENIC FACTORS ATTRIBUTABLE TO VDT WORK

A number of alternate explanations for these clusters have been discussed; X-ray radiation, UV-radiation, microwaves, electric fields, magnetic fields, light air ion depletion, PCB emission, stress annd worry and sedentary work. Two of these factors are at present appearing in the debate: stress and worry and low frequency pulsed magnetic fields.

Stress and worry: Some recent literature reviews have suggested a relationship between stress/worry and miscarriages /Björseth et al 1985, MacKay 1984/. The possibility warrants further research, but available data, although indicative, appear insufficient to establish such

a link. Furthermore, consistent relationships between 'VDT work' and 'higher stress levels' have not been found. It appear prudent, however, to rectify stressful work situations for pregnant VDT operators (as well as non-pregnant operators). The explicit worry due to the debate concerning pregnancy outcomes remains a factor whose possible influence on pregnancy outcomes have not been tested.

Effects of pulsed magnetic fields on embryos: Based on a series of studies, Delgado and coworkers have suggested that square pulsed, low frequency magnetic fields can disturb the embryonal development in chick embryos /Delgado et al 1982, Ubeda et al 1983/. Later investigations have however revealed serious technical shortcomings in these studies - invalidating at least the 'exposure data' presented. Some other groups have failed to replicate these studies on square pulsed magnetic fields, in chick embryos /Maffeo et al 1984, Hansson-Mild unpublished, Tell-cited by Guy 1984/ or mice embryos /Tribukait et al 1987/. A possible exception is a recent study /Juutilainen & Saali 1986/, where some effects on chick embryos of various forms of magnetic fields were indicated. According to the authors, the result did correlate with the amplitude of the field (i e the B levels), but not with the time derivate (dB/dt levels, or current induction).

It has been suggested, that the relevance of these studies to VDT-like situations hinge upon the similarity of the dB/dt-levels. It is then worth noting that the positive study of those cited above, does not indicate that dB/dt is a relevant parameter for the effect. In a review for the WHO Regional Office for Europe, the working group concluded: "Although one or two parameters of the exposure fields employed in these studies are similar to those around VDTs, there are significant differences in numerous other parameters. Any extrapolation of these study results to exposure of VDT operators can only be considered tenuous." /Marriott & Stuchly 1986/.

A few studies have also been performed, where the magnetic fields used were similar in pulse shape and frequencies to those found around VDTs ('saw-tooth pulses'):

In one study by Tribukait et al /1987/, pregnant mice were exposed to various field intensities, the strongest being 15 uT. The foetuses were examined after 18 days (15 days of exposure and 3 days following exposure). In the 15 uT group, 11 (2.9%) of the exposed foetuses showed malformations, compared with 7 (1.4%) among the controls. On the other hand, 2 (0.5%) of the exposed foetuses died after the exposure period, compared to 8 (1.4%) among the controls. Results from this study have been reported twice, both in January (approximately half of the material) and in May 1986. The differences described above were those for the full study. (It was noted, that such differences was only present in the first half of the study, data from the second half of the study showed identical results among the exposed and control groups as to these outcomes.)

Another study by Frölén et al has recently been concluded - with the purpose of

replicating the study of Tribukait et al (above) with saw-tooth pulsed magnetic fields. The study results (here expressed as numbers/100 dams were: Implantations 716 (exposed) and 688 (controls), living foeteses 625 and 644, number of malformations in living foeteses 45 and 45, dead foeteses 15 and 6, resorptions 76 and 38, and number of malformations in dead foeteses 7 and 2, respectively. Thus Frölén et al did not find any increase in the number of malformed living mice foetuses due to magnetic field exposure, and thus the earlier results have not been verified. Instead, a certain increase in the number of foetuses that have started development in the uterus, and a corresponding increase in the number of dead or resorbed foetuses were found.

In an ongoing study on clastogenic effects of pulsed magnetic fields on human amniotic fluid cells (Nordenson and Hansson Mild, 1987), some preliminary results have been released. For both 50 Hz sinusoidal and 20 kHz sawtoth fields experiments, increased frequencies of chromosome aberrations were seen. The authors point out that implications of this to human health is still unclear.

A study has been performed with chick embryos and saw-tooth pulses /Sandström et al 1987/. No significant differences in number of abnormalities were observed between controls and exposed eggs.

Stuchly et al (1987) have recently reported on a study on rats exposed both before and during pregnancy to sawtoth pulsed magnetic fields of similar characteristics. No adverse pregnancy outcomes were found.

EPIDEMIOLOGICAL STUDIES OF PREGNANCY OUTCOMES RELATED TO VDT WORK

Systematic comparisons between VDT operators and non-VDT workers as to pregnancy outcomes have been made in a number of epidemiological studies. Hitherto, about 10 such studies have been conducted. None of these studies have provided evidence for an effect of VDT work on the frequency of miscarriages or serious malformations. In some studies, the frequencies were somewhat increased - which were explainable by confounders and/or random variations. In other studies, the frequencies were somewhat decreased - likewise explainable. These studies were of varied size and quality. Five of them can be considered (according to the authors) as fulfilling some necessary quality criteria. These studies are briefly presented in table 1.

The existence of five large and apparently well conducted studies enables an evaluation not only based on the presence or absence of statistical significance or computed confidence intervals in each study, but that can also consider possible consistent ('nonsignificant') findings across these studies. Concerning miscarriages, there are then results which can be

Table 1. Outcome of five epidemiological studies that compared miscarriage and malformation occurrences for women with and without VDT work during pregnancy.

Study	Number of pregnancies	Did the study find evidence of link between VDT work and:	
		Miscarriage	Malformations
Swedish case-control study /Källén & Ericson 1986/	1 447	No	No?
Swedish Nat'l Insurance study /Westerholm & Ericsson 1987)	4 347	No	No
Finnish case-control study /Kurppa et al 1985/	2 950	Not inv.	No
Montreal study /McDonald et al 1986/			
- previous pregnancies	3 863	No	No
- current pregnancies	3 799	No?	No
Michigan study /Butler & Brix 1986/	817	No	Not inv.

Comments for table 1: No = no significant differences occurred. No? = some differences were noted, but the final evaluation by the authors was that these differences did not appear to constitute evidence for an increased risk due to VDT work. Some further details and the authors own conclusions regarding their studies are presented in the text.

evaluated in different ways in the Montreal study (current pregnancies only) and possibly also in the Michigan study (as discussed above). In contrast, the Montreal study (previous pregnancies), the Swedish Nat'l Insurance study and by and large the Swedish case-control study lack such results. Concerning malformations, results open to interpretations exist in the Swedish case-control study, but are not found in the Montreal study, by and large not in the Swedish Nat'l Insurance study and not in the Finnish case-control study. A consistent pattern of increased risks is thus not present.

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THE ANATOMICAL DISTRIBUTION OF SOLAR UVR WITH EMPHASIS ON THE EYE

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INTRODUCTION

Solar ultraviolet radiation (UVR) has been implicated in the induction of skin cancers and cataracts. Doses received by the body and the eye during different activities and over the lifetime of an individual have only recently begun to be quantified.

Polysulphone (PS) film is a possible UVR dosimeter and several studies investigating the anatomical distribution of UVR have been published [1-4]. Much of the work to date relates the UVR dose to that at a horizontal plane (defined here as ambient), and makes the assumption that the spectral power distribution (SPD) of incident radiation at all the exposure sites is the same. The spectral response of PS does not exactly match the biological response of human skin or eyes to solar UVR and variations in the incident spectrum could therefore introduce significant errors in the calculated UVR dose. Of particular interest is the SPD of UVR incident upon the eye, since the hazards of UVR vary greatly with wavelength, especially near 300 nm. The transmission of UVR through the ocular media is complicated by many factors. The natural protective features of the eye such as pupil dilation and squint mechanism greatly modulate the dose of UVR which enters the eye. This work presents the dose response of PS film to solar UVR, the anatomic distribution of solar UVR, and also the SPD of radiation incident on the eye.

EXPERIMENTAL METHODS

Ambient measurements of the solar UVR SPD were made using a Spex 1680B double monochromator system. Continuous radiometric measurements of solar radiation, solar UVR and solar UVB radiation were made using an Eppley pyranometer, Eppley UVR radiometer and an International Light IL700 UVB radiometer respectively, at the ARL in Yallambie (38°S). The complete measuring system is described in detail elsewhere [5].

A number of different headforms and mannikins with small PS film badges attached at various sites were placed either on a rotating turntable, or simply positioned to face in a given direction, and exposed to solar UVR. Exposures took place in a large open area well clear of any buildings, and the ambient radiation was monitored continuously with separate PS film while solar UVR SPDs were measured every 10 minutes.

Measurements of the SPD of solar UVR incident on the eye socket of one of the headforms were made using a 25 mm integrating sphere and a 2 m liquid light guide as input optics of the double monochromator, with the sphere positioned with its input port at the eye aperture. It was not possible to rotate the headform while scanning, so SPD measurements were made with the headform facing Nth, Sth, East and West, with the ambient SPD measured immediately before and after.

RESULTS AND DISCUSSION

Exposure of PS film to solar UVR and simultaneous measurement of the SPD of the incident radiation allowed quantification of the radiation received by each PS film badge. Weighting of the solar UVR SPD with the recently introduced CIE [6] erythemal response (ER) gives a dose response curve of change in absorbance A versus erythemally effective dose (EED) in J/m^2. This dose response curve (Figure 1) was used to determine all relative UVR doses with respect to ambient for headform PS exposures. For comparison the results of Diffey [7] are shown. The difference between the two curves is mostly due to the different ER's used in each analysis. When the CIE [6] ER is used on the results of Diffey [7] the agreement is markedly improved (Figure 1).

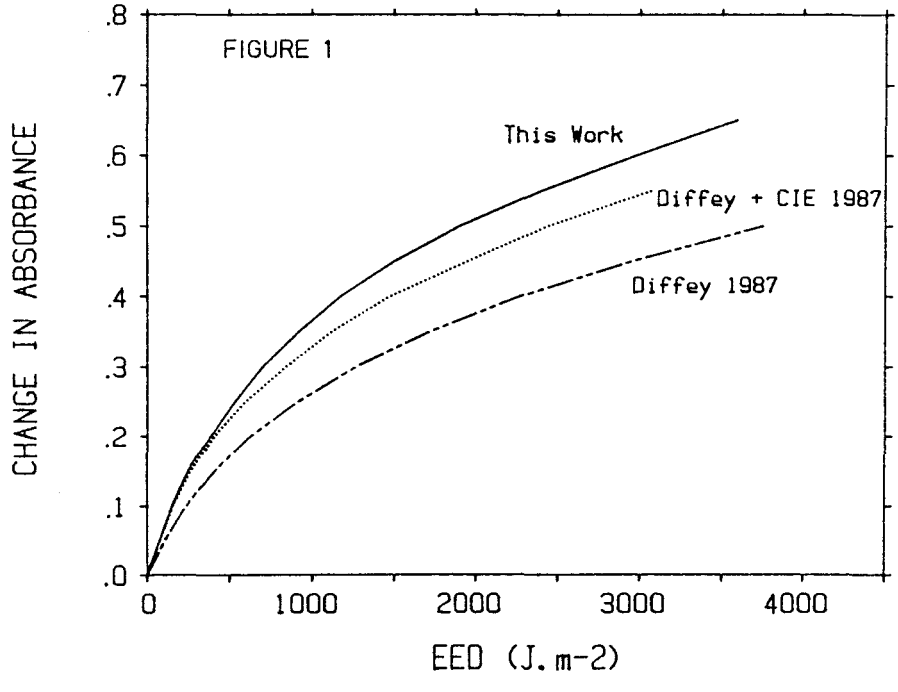

Figure 1. Change in absorbance versus erythemally effective UVR dose (EED) for polysulphone film.

Measurements of the anatomical distribution of solar UVR are shown in Table 1, and compared with previous measurements. While there are slight variations between the different studies, probably due to variation in conditions under which the exposures took place, similar distributions of relative UVR dose are evident. It was also observed that the relative dose at each anatomical site was independent of cloud cover [1].

Table 1

Anatomical Site	UVR dose as a percentage of the vertex (ambient) This Work *	Diffey[1]*	Holman[3]
Vertex	100	100	100
Cheek	17 - 30	30 - 31	15 - 47
Shoulder	89 - 99	80 - 68	66 - 70
Upper Arm	41 - 32	49 - 56	59 - 66
Lower Sternum	58 - 42	58 - 73	44 - 46
Lumbar Spine	47 - 24	43 - 51	58 - 71

* Cloud cover varying from overcast to clear

Measurements of the relative UVR dose at sites about the face using rotating headforms and PS film are shown in Table 2. Of particular interest is the ocular to ambient exposure ratio (OAER) which is found to vary from 0.12 to 0.40 depending upon the time of year, and this range encompasses previous measurements (Diffey et al [2] OAER=0.46, Rosenthal et al [4] OAER=0.19). Many of the anatomical sites show a variation of the relative UVR dose with season, the effect of solar elevation playing an important role in the irradiation process. Table 2 shows the variation of anatomic distribution measurements for the months of January, February and May. The results for summer (Jan-Feb) show a pronounced change from those for May. Wearing a hat reduces UVR dose at all sites, while sunglasses reduce the ocular dose.

TABLE 2

The relative UVR dose as a percentage of the vertex. The results of Diffey et al [2] are shown for comparison. Also shown are the effects of wearing sunglasses and hats (measurements made in February).

Position	Diffey[2]	May	January	February	When Wearing Glasses	Hat
Vertex	100	100	100	100	100	-
Forehead	58	72	31	44	39	<1
Side of Nose	42-51	74	31	32	23	5
Centre of Nose	66	77	50	66	52	16
Cheek	29-50	62	22	28	18	12
Eye	46	41	14	12	2	4
Upper Cheek	14-41	63	26	39	11	8

Exposures with PS film on the 4 headforms facing N,S,E and W gave average values in good agreement with those obtained using rotating headforms, so averaging the ocular scans should also be valid. Table 3 summarizes the SPD measurements for the 4 directions.

Table 3

Direction	Time	Conditions	UVR Irradiance (W/m^2)	Ocular Irradiance as a Percentage of ambient UVR	UVA*	UVB**
East	1230 (0900)	Clear	3.5 (6.1)+	6 (13)	6 (13)	6 (13)
North	1235 (0905)	Clear	4.2 (3.7)	7 (8)	7 (8)	6 (7)
West	1240 (0910)	Clear	4.1 (3.8)	7 (8)	7 (8)	7 (9)
South	1245 (0915)	Clear	3.6 (3.6)	6 (6)	6 (6)	6 (8)
North	1350	Overcast	3.1	16	16	12

* UVA = 315-400 nm ** UVB = 280-315 nm + (am values in brackets)

The irradiances listed are the integrations of the SPDs over the appropriate wavelength ranges and as expected, those with larger direct components of solar radiation are higher than those without (south). The SPDs for each of the directions was not significantly different from that of ambient.

CONCLUSIONS

(1) The anatomical distribution of UVR varies with time of year.

(2) The spectral distribution of UVR incident on the eye is not detectably different from that of ambient.

(3) Although the spectral response of PS film is not ideal the results obtained thus far indicate that it is suitable as a personal solar UVR dosimeter.

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UV - RADIATION: AN EPIDEMIOLOGICAL STUDY ON MALIGNANT MELANOMA IN GENERAL POPULATION AND A MODEL OF OCCUPATIONAL STUDY ON RESEARCH WORKERS

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ABSTRACT

Malignant melanoma of the skin has increased rapidly in the last decades,and the problem of its etiology has caused considerable concern. In Italy a threefold increase in the mortality rate of melanoma has been observed in the period 1955-1978. An epidemiological study has been conducted in the city of Rome and the incidence of melanoma has been determined retrospectively for the years 1970-79. All public and private health structures of the city were involved.A case was defined as patient resident in Rome with histological diagnosis of melanoma performed between 1970 and 1979. In the years 1970-72 the average incidence rate per year was 0.8 per 100,000 inhabitants both for males and females;while in 1977-79 the average rate was up to 2.0 per 100,000 for males and 2.4 for females (age standardized rates). The annual increase in incidence was 19% for males and 25% for females. Age specific incidence rates were quite similar in both sexes. In the aetiology of melanoma the authors suggest that one of the major factors that plays a role in the increase of melanoma is sun exposure. In order to analyse the occupational exposure to UV, particularly to workers of research referred, monitoring and control was planned. In the National Council of Research (CNR) we decided to screen the main kinds of sources used in relation to the number of workers involved, their exposure time and so on. In the Poster we wil show the results of the epidemiological study and of the screening with the subsequent consideration.

SAFETY IN THE USE OF NON-IONISING RADIATION:A FEW SAMPLE SURVEYS

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This paper summarises the results of a few sample surveys carried out in course of an on-going programme at Trombay, on safety during use of equipment generating non-ionising radiation in three different categories, namely, visible radiation from gas-filled discharge-tube flashes, lasers and ultra-sound. Some of the measured levels in the above three categories which deserve attention have been discussed in the light of exposure limits (ELs).

SURVEYS ON OPTICAL RADIATION HAZARDS FROM XENON TUBE FLASHES

Measurement of visible radiation during flashes from 28 cm long 16mm dia cylindrical tubes have been carried out in a laboratory engaged in fabrication, assembly and testing of these tubes for use with glass type of lasers. Tubes, during the course of testing and calibration, are normally covered with an opaque box to contain all radiation but there are inadvertent leakages which vary widely. Surveys covering various distances and locations in the laboratory from the site of leakage were carried out with the use of a portable survey-meter (Sachdev, 87) incorporating PIN photo diode as sensing element. Instruments of this type can be used to measure radiation in any part of the optical spectrum with the use of suitable combination of band-pass and band-stop filters. While the study of spectral radiance from the flashes was carried out in the 0.26-0.60 μm range, the PIN photo diode at the time of measurements was incorporated with a set of optical filters limiting its range between 310 nm to 720 nm.

From a series of measurements made during these surveys, those carried out on leakage of radiation at the end-on side of the cylindrical tube are reproduced in Table 1. Retinal, thermal hazard and blue light hazard given by $\Sigma L_\lambda R_\lambda \Delta\lambda$ & $\Sigma L_\lambda t\, B_\lambda \Delta\lambda$ respectively were evaluated (WHO, 1982) with the use of retinal thermal (R_λ) and blue light (B_λ) weighting functions(sliney,1980). Weighted spectral radiances in the two cases are essentially as integrated over the entire available wavelength range for the duration of the flash (t). It is seen that retinal thermal hazard during these measurements is about two orders higher than the suggesed hazard limit of 1.716 kW/cm^2 computed on the basis of angle subtence by the source on the receptor. The blue light hazard from the flash is, however, a small fraction of the hazardous limit of 154 $watts/cm^2.sr$. It may be pointed out that similar measurements on broad-side of the tube would give even higher values of radiance for the radiation leakage.

SPECULAR REFLECTION MEASUREMENTS

It is common in environmental studies planned to measure concentrations of specific toxic gases using optical methods to use a set of metal deposited concave mirrors to increase the effective path-length by virtue of multiple reflections. Suitably located

mirrors, either in the laboratory or outdoor, are used to measure the extent of absorption of radiation of specific wavelength during the course of travel of the beam between the source and the detector. Mirrors in the experimental assemblies in our use were deposited with aluminium, known for good reflectivity in the middle infra-red. A He-Ne laser emitting at 3.39 μm was required to be used for detection and measurement of methane in the environment. The problem of a hazard from specular reflections in this type of assemblies arises during alignment of measuring set-up with the use of 0.6328 μm wavelength from He-Ne laser.

The results of studies on specular reflections in the assemblies incorporating concave surfaces with radii of curvature 30 and 575 cm are summarised in Table 2. It is evident from the observed ratios of irradiance in reflected beam to that of direct beam that even when incident irradiance is kept within EL, the irradiance in specularly reflected beam could be orders higher in certain cases. The obseved irradiances in general are a composite function of distance traversed in between reflections, radius of curvature of reflecting surface and the quality of finish of the surface (WHO, 1982). Caution is needed during alignment, while using class 2 lasers in this type of assemblies, since these invariably involve specular reflections from surfaces having large radius of curvature (IRPA/INIRC, 1985).

ULTRA SOUND SURVEYS

Safety surveys have been carried out at Trombay on about a dozen commercially available laboratory ultrasonic cleaners operating at 40 KHz and ultrasonic dental scaling units operating at 28.5 KHz. These studies were carried out with the use of type 2209 Bruel and Kjaer precision sound level meter receiving signal from a microphone type 4135 having an open circuit flat frequency response between 4 Hz and 100 KHz. Octave band filter-set type 1613 was used with the sound level meter. The meter had earlier been calibrated with sound level calibrator type 4230.

The observed values of sound pressure level (SPL) in most cases around operating dental scaling units are within recommended limits for continuous occupational exposure to airborne ultra-sound but those around some laboratory cleaners make interesting observations and are reproduced in Table 3. Sound pressure levels as given in these tables are those observed with octave bands with mid frequencies at 8 KHz, 16 KHz and 31.5 KHz respectively. Octave band filter centered at 16 KHz will have its upper and lower frequency limits as 22.64 KHz and 11.32 KHz respectively. The upper frequency limit of this octave band filter thus almost coincides with the upper frequency limit of a one-third octave filter centered at 20 KHz (limits 17.89 and 22.36 KHz). It appears reasonable to compare the observed SPL's with the recommended limit of 75db for the one-third octave band with a mid-frequency of 20 KHz (IRPA/INIRC, 1984). Values of 85 db and above, even at a distance of 6 m from the unit would appear to be high. Similarly the SPL's measured at a distance of 0.5 meter with the use of octave band filters centered at 31.5 KHz when compared with the recommended limit for continous

TABLE 1
Evaluated parameters on spectral radiance, retinal and blue light risk from xenon lamp flashes

Wattage of the source	:	1385	kW
Duration of the flash	:	650	μsec
Irradiance measured at a distance of 70 cm	:	2.5	mW/cm^2
Computed source radiance	:	6	kW/cm^2 .Sr
Weighted thermal retinal hazard	:	85.1	kW/cm^2 .Sr
Estimated thermal retinal hazard limit	:	1.716	kW/cm^2 .Sr
Weighted blue light retinal hazard	:	5.2	W/cm^2 .Sr
Estimated blue light retinal hazard limit	:	154	kW/cm^2 .Sr
Spectral radiance in 310-390 nm region	:	30.4	%
Spectral radiance in 400-495 nm region	:	35.3	%
Spectral radiance in 500-600 nm region	:	34.0	%

TABLE 2

Observed ratios of reflected and incident irradiances during a study of multiple specular reflections from concave aluminium coated reflectors with the use of a He-Ne laser operating at 632.8 nm

Sr. No.	No. of reflections	Effective distance (cm)	Radius of Curvature (cm)	Ratio of irradiances	Remarks
1	1	60	30	0.65	Laser and detector located at planes coinciding with centre of curvature of the surface.
2	2	110	30	0.36	Laser and detector located away from the planes coinciding with centre of curvature of surface.
3	3	2300	575	180.40	Laser and detector located at planes coinciding with centre of curvature of surface.
4	4	235	30	0.35	Laser and detector located away from planes coinciding with centre of curvature of surface.

TABLE 3

Typical ultra-sound pressure levels measured around a few operating laboratory ultrasonic cleaners

Ultrasonic cleaner No.	Distance from cleaner (metres)	Mid frequency of octave band (K Hz)			All pass (10-70 K Hz)
		8	16	31.5	
I	0.5	72	100	117	117
	2.5	77	95	109	109
	4.5	74	92	103	104
	6.0	72	89	99	100
II	0.5	73	94	116	116
	2.5	-	98	100-104	106
	4.5	67	87	97	100
	6.0	67	86	95	98
III	0.5	79	94	114	114
	2.5	72	86	101	104
	4.5	72	87.5	96-98	102
	6.0	71	85	99	103

occupational exposure (1/3 Octave band) of 110 db would appear to be high (WHO, 1977).

In a laboratory where such tanks are being operated to clean delicate mechanical and optical parts, authors have found it difficult to perform scientific tasks requiring concentration. This observation appears to confirm what several other workers have recorded (WHO, 1982).

CONCLUDING REMARKS

The studies reported in the paper are based on a few sample surveys carried out on equipment generating non-ionising radiation. They have helped locate areas which merit more detailed study. It can be concluded based on these studies, that while use of NIR in science, medicine and industry is on the increase, safety awareness and built in safeguards require to be looked into, in more detail, in several areas.

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SAFETY MEASURES FOR THEIR CORRECT USE IN THE WORKPLACES, AS A RESULT OF MANY EVALUATIONS OF DIFFERENT HUMAN-COMPUTER INTERACTION FACTORS

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ABSTRACT

The video display terminals (VDT's) use is widespread and continuously increasing, and VDTs can be found in practically in every modern office.

Although their use provides many economical benefits, they have also given rise to many questions about potential health problems of operators.

For this reason it has been necessary to investigate and evaluate all the factors (ergonomic, environmental, health, safety, radiation emission, and so on) which could provoke some health implications to operator.

The authors have evaluated a series of measurement and other determining factors in connection with the above problems, examining the widespread use of many different types of VDTs in different environmental locations.

As a result they hereby propose a series of standards which should be respected when it is utilizing VDTs in the working environment, in order to optimize their use.

EXPERIMENTAL DETERMINATION OF A LASER RETINAL LESION THRESHOLD IN THE VISIBLE SPECTRUM.

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The risks due to the use of laser have called for the definition of exposure limits. In France up to now, no experimental study was especially directed to determine the laser beam threshold's injuries or to verify the limits proposed by several international organizations, generally issued from the figures published by the American National Standard Institute (1976). In this aim, a contract of research was supported by the Direction des Recherches Etudes et Techniques of the Ministère de la Défense and then the Commissariat à l'Energie Atomique. The experimental study concerns the ocular effects of the laser beam in the visible spectrum where the exposure limit values are the lowest and the protection is delicate to assume.

The investigations were at first performed to study the effect of a single pulse whose the duration was widely shorter than the blink reflex and to determine the threshold level of a selected damage criterion corresponding to a pathologic change of the retina. The experimental results are compared to the actual limits proposed for one laser pulse emitted in the visible spectrum in cases of a point source and extended source viewing conditions.

Two experimental layouts were used in this study. The first source was a pulsed dye laser emitting at 593 nm. The pulsewidth was 600 ns. The second source was a locally constructed Q-switched Nd:YAG laser delivering one pulse of 40 ns. A KTP crystal was used to frequency double the laser output at 532 nm. Different systems were mounted to give the most uniform energy distribution of the laser beam in the observation plan. In both optical systems the retinal image, varying from 30 to 570 µm, is formed using a Maxwellian-view procedure. The experiments were carried out on monkeys and rabbits. We have used two different methods to perform this work testing both their sensitivity and their reliability. The eye fundus examination was made by direct ophthalmoscopy and fluorescein angiography immediately after the exposure and after 24 hr.

A retinal grey opacity which is in our experiments the slighest retinal damage detectable by a direct ophthalmoscopic method was selected as the threshold damage criterion. We have noted that if the most injuries are visible 15 mn after the exposure, it is only after 24 hr that all injuries can be observed with this technique. The same damage is observed by fluorescein angiography as a well delimited yellow-green spot revealing a persistent accumulation of dye anterior to the pigment epithelium. Immediately after the exposure the most lesions near the threshold detected by fluorescein angiography was not ophthalmoscopically visible. The size of the fluorescent spot was smaller in the lesions induced by exposures near the threshold. Depending on the radiant exposure the persistency of the fluorescence was varying from some ten minutes to some hours. In experiments carried out on the rabbit retina with minimal

exposures very near the threshold, the fluorescent spot like a granular pattern could not be distinguished by a new injection of fluorescein at time periods after the exposure varying from 3 to 6 hrs.

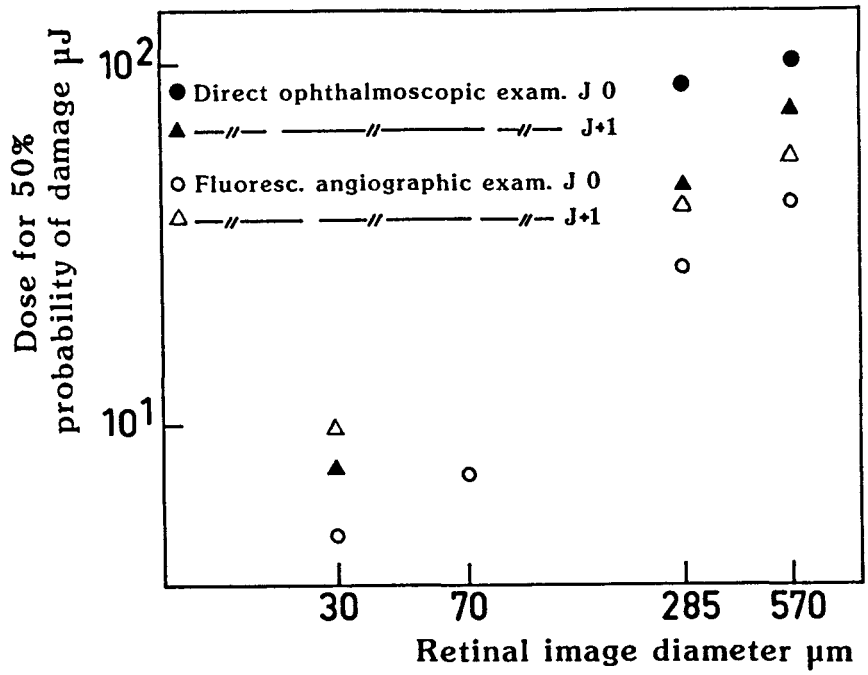

Figure 1. ED_{50} obtained on rabbit retina as a function of the retinal image diameter, immediately (J0) and 24 hr postexposure (J+1), with two methods of fundus observation.

Retinal grey opacities and fluorescent visible lesions were scored and analysed by a method of probit analysis. The statistical calculations had shown that fluorescein angiography is an investigative method allowing to detect laser induced lesions at radiant exposure levels below those which produce ophthalmoscopically visible changes (Fig. 1).

The experimental results obtained with fluorescein angiography could be extrapolated to human and the threshold damage values, commonly represented by the median effective dose (ED_{50}), could be compared with the limit values proposed. Of course, the comparison with the experimental values should be made with the limits corresponding in human to retinal images of same sizes.

In the case of retinal images of 30 µm it agrees to compare on the values found on the rabbit retina with the limit value recommanded for minimal retinal images (Fig. 2). This limit is lower than the energy related to the smallest probability of damage. The experimental results are in good agreement with the corresponding limit established for the vision of a point source.

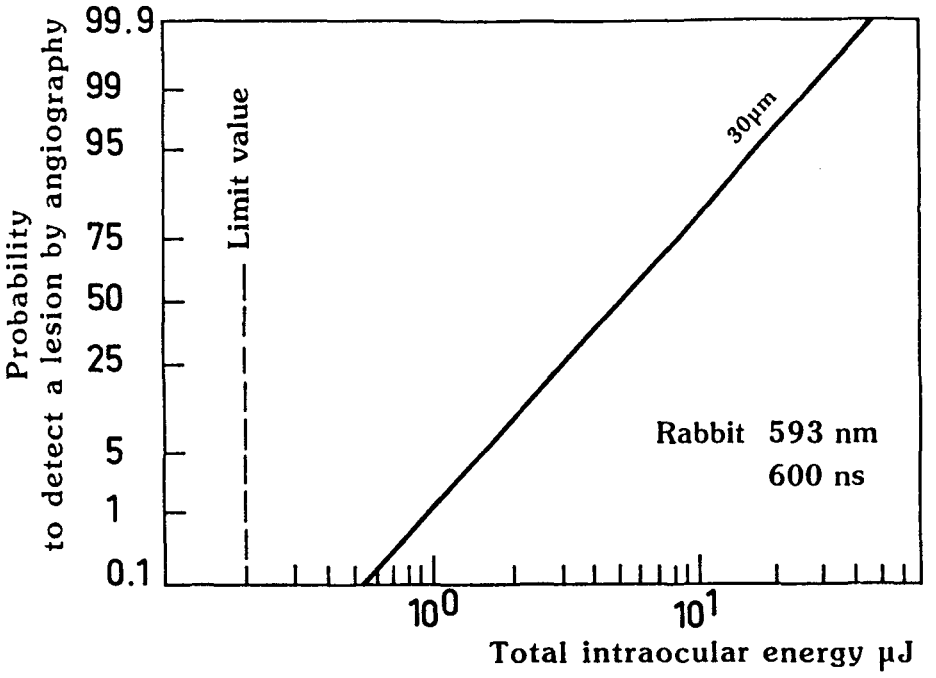

Figure 2. Point source vision. Comparison between the probit regression line and the corresponding limit value.

The exposures produced with the largest images (70 to 570 µm) correspond to the extended source viewing conditions (Fig. 3). Then, it agrees to compare the retinal radiant exposure thresholds experimentally found for each image diameter to the corresponding limit values expressed too in retinal radiant exposure. For extended source viewing conditions the limit values are given in function of the exposure duration. The shortest exposure studied in this experimental work was obtained with a pulse duration of 40 ns and a retinal image diameter of 250 µm on the monkey and 200 µm on the rabbit. The ED_{50} are 52.9 and 20.2 $mJ.cm^{-2}$ respectively. The corresponding exposure limit, equivalent to a retinal radiant exposure of 4.0 $mJ.cm^{-2}$, is lower than the ED_{50} experimentally determinated but the probit regression lines show this limit relates to weak probabilities to detect a lesion by fluorescein angiography (1.4% and 3% respectively).

For a pulse of 600 ns, the experimental results obtained on the rabbit retina with retinal spot sizes of 70, 285 and 570 µm are 184, 43.8 and 17 $mJ.cm^{-2}$ respectively. All these ED_{50} calculated by the method of probit analysis are superior to the retinal radiant exposure limit equivalent to 10 $mJ.cm^{-2}$ related to a pulsewidth of 600 ns. For the retinal image diameter of 70 µm this limit is also lower than values given by the probit regression line for the smallest probabilities. However, the comparison of the limit value with the probit regression lines fitting the experimental data obtained with retinal image diameters of 285 and 570 µm shows that exposure limit corresponds to a probability to observe a fluorescent spot on the fundus of 7 and 32 %, respectively.

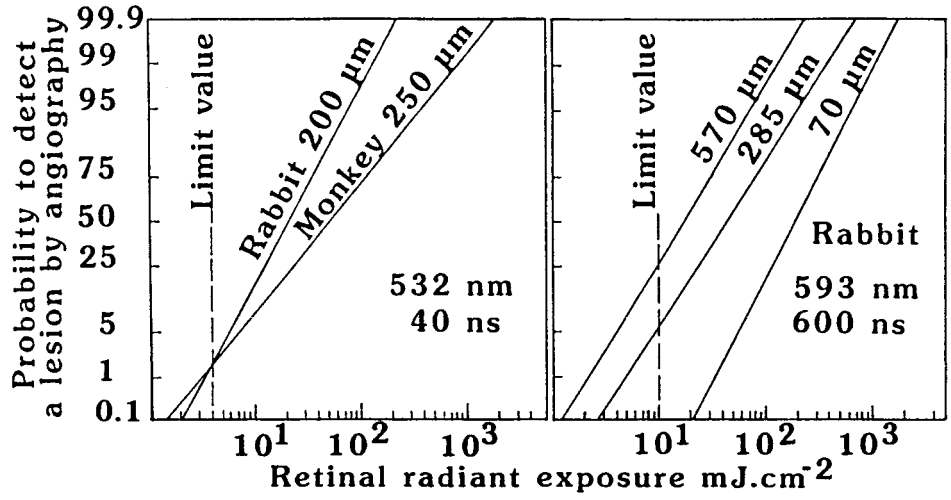

Figure 3. Extended source viewing condition. Comparison between the probit regression lines and the corresponding limit values.

Then the ED_{50} values, determinated in extended source viewing condition with both pulsewidths of 40 and 600 ns are upper than the limit values but the discrepancy is considerably reduced when the image size increased. The probability to detect a fluorescent spot increases. This is particularly evident with experiments carried out on the rabbit using retinal spot size of 570 µm where fluorescein angiography had permitted to detect retinal damages induced by energy levels very near or below the limit value. Such damages are described in the histological study performed by Guéneau et al. and reported in the present volume.

In conclusion, these experiments had shown for the study of retinal lesion thresholds but also for the clinical investigation of a possible accident, the importance of both the method and the delay of the observation after the exposure. In our experimental conditions, the fluorescein angiography appeared the most sensitive technique and should be used immediately after the exposure. At the opposite, the best delay using the direct ophthalmoscopic method is 24 hr after the exposure.

Our experimental results obtained with a fluorescein angiographic method, extrapolated to human eye, support the limit value established for the vision of a point source but do not agree with the exposure limit established for extended source viewing condition. In this case, the limit corresponds to a probability of damage increasing with the retinal image diameter.

RETINAL LESIONS INDUCED ON RABBITS BY SINGLE AND LOW ENERGY LASER IRRADIATIONS. HISTOLOGICAL AND ULTRASTRUCTURAL STUDY.

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The aim of this histopathological study is triple : I) to specify the retinal layers damaged by each radiant exposure studied, 2) to follow the healing reactions with observations carried out at different time intervals, 3) to determine the minimal radiant exposure to produce a cytological damage visible with the electron microscope.

The experimental animals are adult "Fauve de Bourgogne" rabbits. Two different laser sources were used : a pulsed dye laser and a frequency-doubled YAG laser.

RETINAL EFFECTS OF THE PULSED DYE LASER

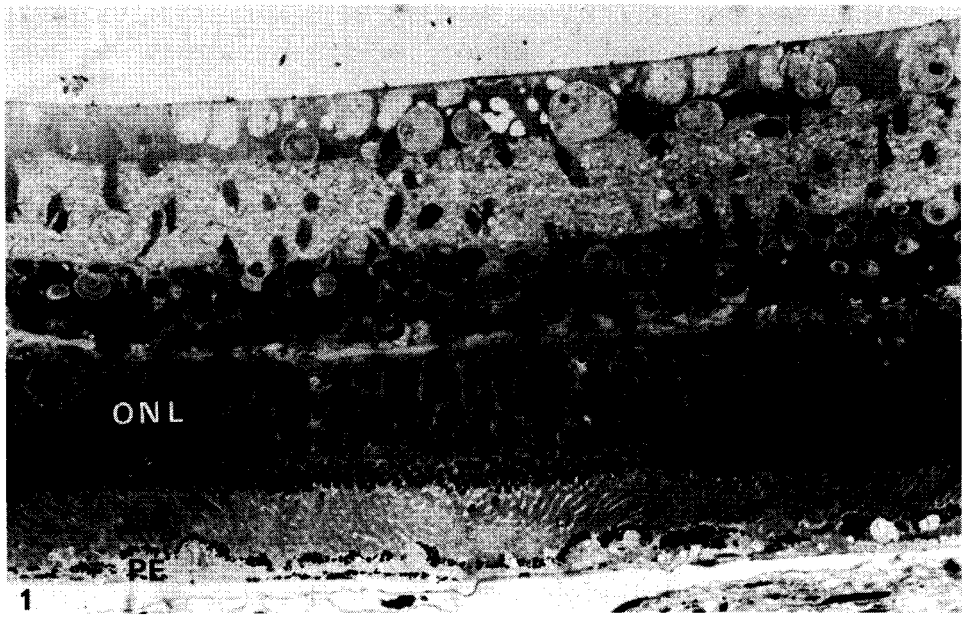
With the pulsed dye laser the wavelength is 593 nm, the single pulse lasts 600 ns, and the retinal image diameter is 570 µm. Five energies, expressed in radiant exposure at the retina, have been studied : II8, 45, 26, I6 and I2 $mJ.cm^{-2}$, and the examinations by light and electron microscopy have been carried out 24 h., 3, 7 and 42 days after the radiations.

The lesion onset in this experiment is always the retinal pigment epithelium (thermal effect). When the energy delivered at the retina is less than 45 $mJ.cm^{-2}$ the lesion remains localised under the outer limiting membrane.

After a pulse delivering I6 $mJ.cm^{-2}$ at the retina the swelling of this one is almost imperceptible 24 hours post exposure, but three facts prove the presence of the lesion : a) the ripple and destroyed aspect of pigment epithelium where only melanine granule subsist, b) the slight oedema and the disorganization observed in the outer segments layer, c) at last the very slight damage of inner segments layer which has not its normal thick and tight aspect (Figure I).

With the electron microscope we confirm that the lesion still remains severe : an oedema invades the outer segments layer where scattered melanine granules are observed, and above all many images of discs degeneration.

The I2 $mJ.cm^{-2}$ radiant exposure at the retina is the lowest energy level we have studied so far. 5 hours post exposure we can already observe on the I µm thick sections a slight injury of



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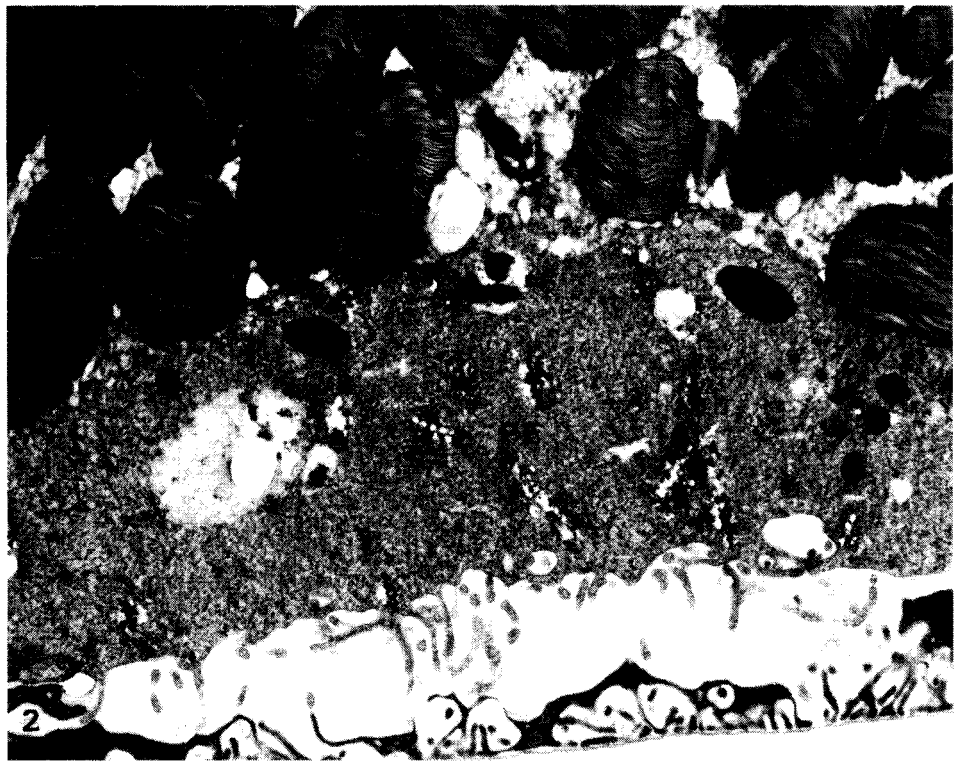
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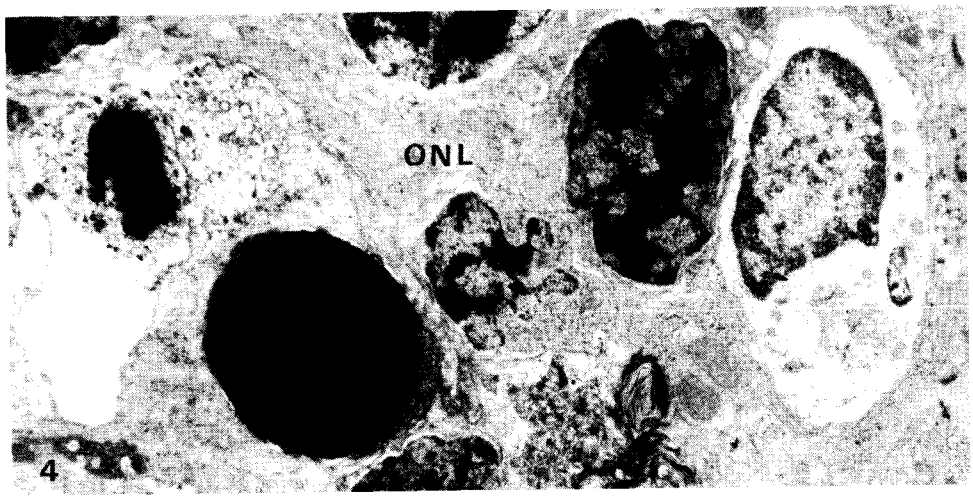

some epithelial cells and outer segments tips. The Figure 2 shows a totally burnt epithelial cell observed with the electron microscope 4 days post exposure.

Two facts hold the attention in the healing phenomena : the proliferation of new epithelial cells which are perfectly observed 7 days post exposure, and the progressive reorganization or "repair" of the outer segments observed 42 days post exposure.

RETINAL EFFECTS OF THE FREQUENCY-DOUBLED YAG LASER

With the frequency-doubled YAG laser, our physical parameters are different : the wavelength is 532 nm, the single pulse lasts 40 ns, and the retinal image diameter is 200 µm.

We have only studied up to now three radiant exposures at the retina : 80, 35 and 18 $mJ.cm^{-2}$, and the examinations by light and electron microscopy have been carried out 4 h. and 29 h. after the illuminations.

The main result is the importance of the damage with the relatively low radiant exposure (18 $mJ.cm^{-2}$). The Figures 3 and 4 show that the retinal layers involved in the lesion examined 29 h. after exposure are the pigment epithelium, the outer and inner segments of the photoreceptors, and also the outer nuclear layer.

We need other observations to specify the exact aspect of the damage induced by the frequency-doubled YAG laser, but what is already gained is that the minimal radiant exposure producing a cytological damage in the retina is far below 18 $mJ.cm^{-2}$. The study is carried on.

PROBLEMS IN APPLYING THE EFFECTIVE DOSE EQUIVALENT IN URANIUM MINING

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ABSTRACT

Application of the effective dose equivalent (EDE) in uranium mining requires limiting the combined exposure to radon daughters, thoron daughters, uranium and thorium and their long-lived daughters, and gamma radiation. This can be done by limiting to unity the sum of the exposure to each source divided by its respective exposure limit, but controversy exists about the appropriate limit for radon daughters.

The AECB included the radon daughter limit recommended in ICRP-32 (0.02 J/y or 4.7 WLM/y), as well as the EDE concept, in proposed regulations which it issued for public comment in 1983 and again in 1986. The resulting comments from industry, labour, and the scientific community included opposition to the radon daughter limit as well as to the application of the EDE to mining. Some opposed the basis of the ICRP-32 limit, which is a combination of epidemiologic and dosimetric considerations. Some favoured epidemiology whilst others supported the dosimetric approach. Some opposed using the EDE in mining because an epidemiology-based limit for radon daughters inherently accounts for some exposure to the other sources of radiation. To include each source in the EDE equation would therefore imply some degree of double accounting. In addition, apprehension was expressed about the practicality of complying with a limit on EDE.

The evolution of the AECB's regulatory approach to uranium mining, taking into account the scientific and practical problems mentioned, is discussed in the paper.

VERIFICATION OF THE PREDICTED LEVELS OF ENVIRONMENTAL RADON, ORIGINATING FROM MULTIPLE LARGE AREA SOURCES

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INTRODUCTION

An assessment of the contribution of radon-222, emanating from mine tailings, to the exposure of the population on the Witwatersrand[1], used an atmospheric transport model described in the AIRDOS-EPA computer code[2]. This code is well known and is used extensively for radiation impact assessments and for setting radionuclide emission standards[3]. It is imperative for the application that the accuracy and realism of the model calculations be determined by comparing actual measurements with predicted values under comparable conditions.

Validation studies of the AIRDOS-EPA code for elevated single source releases over extended distances have been reported on by Field **et al**[4]. Amongst other observations this study emphasizes the influence of the observation period on the results and it is shown that agreement between measured and predicted values improves dramatically as the period is extended. It also points to the role of topography and the importance of the vertical mixing height for long-distance applications. Other validation studies[5] emphasize the importance of the procedure used to determine the atmospheric stability class from the measured meteorological data, especially for predictions over shorter distances.

SITE CONDITIONS

For the assessment of radon concentrations on the Witwatersrand (WWR), 250 tailings dams and ventilation shaft sources distributed over a distance of 80 km were considered. As the significant contribution from a ground-level source diminishes sharply with distance, predictions were limited to distances of 10 km from each source. An area of 10^4 km^2 was divided into 1 km x 1 km grid spacings and overlapping values from single sources were summed to provide an integrated prediction from the multiple source distribution.

For the validation study an isolated area with four well-defined sources was selected and predicted values compared with field measurements for two observation periods of 10 and 60 days respectively. Short-term observations required sensitive measuring techniques. Ten monitoring points equipped with passive radon monitors were deployed in a down-wind direction. Longer observation times could be handled with track-etch detectors and observation points were established covering most grid sectors in a 100 km^2 area.

Validation studies were performed during the mid-winter periods which are dominated by highly stable atmospheric conditions, resulting in weak dispersion. These periods are therefore associated with maximum atmospheric radon concentrations.

A major weakness in the overall assessment is the quantification of the source strength. Because of the extended and inhomogenous nature of radium in the tailings dams, as well as the variability of the radon emanation which is dependent on atmospheric and environmental conditions, the source strength is uncertain and can be extremely variable. For the validation, the radon emanation from the four relevant tailings dams was carefully measured, during the mid-winter periods when the absence of rain or intense atmospheric pressure variations allows a reasonable measure of homogeneity in the source strength. Atmospheric concentrations are directly proportional to the source strength.

The Gaussian dispersion equation used in the model does not make provision for topographical features and comparisons were therefore performed over flat areas. The influence of topographical variations, in particular valleys, was investigated.

SENSITIVITY ANALYSIS

Approximations are often necessary for input parameters used in the transport model. Amongst these are source diameters, source heights, the vertical mixing depth and the atmospheric stability classification. The sensitivity of model predictions to variations in these parameters was determined.

The variation of the source diameter from 0,25 to 1,5 km resulted in significant differences within 2 km from the source. Smaller sources resulted in appreciably higher predicted concentrations at short distances from the source. The same effect is experienced with variations of the source heights between 0 and 30 m. At greater distances (more than 2 km) the differences become insignificant. The depth of the mixing layer was not significant and only small differences occur at all distances.

Atmospheric stability classification is a prerequisite for the determination of dispersion coefficients used in the Gaussian equation and requires input data of varying complexity. Two well-known methods, viz that of Turner (using cloud cover) and that of Vogt (using the vertical temperature profile), were compared for the same data base. The respective distribution of stability categories is illustrated in Fig 1.

The difference in the predicted concentrations, resulting from the use of Turner and Vogt classifications, is highly significant and also strongly dependent on distance, as shown in Fig 2.

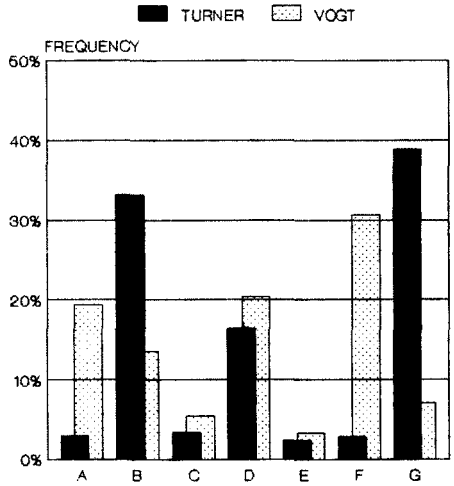

Fig.1 STABILITY DISTRIBUTION BY CLASSIFICATION METHOD

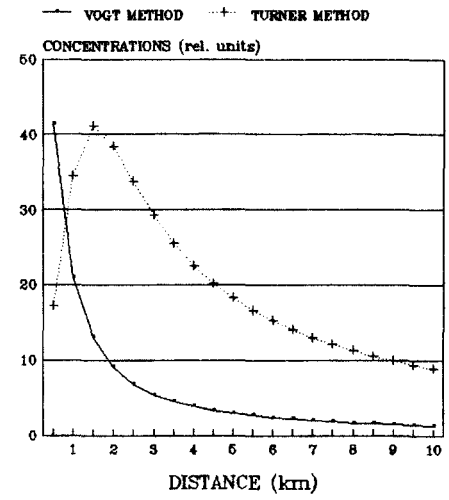

Fig. 2 COMPARISON OF CONCENTRATIONS PREDICTED BY THE AIRDOS MODEL USING DIFFERENT STABILITY METHODS.

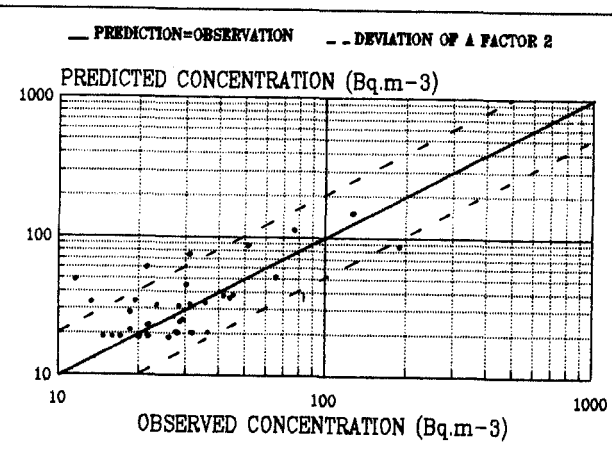

Fig.3 CONCENTRATIONS OF LOG-PREDICTED AND LOG-OBSERVED RADON CONCENTRATIONS FOR THE SHORT-TERM STUDY.

PERFORMANCE OF THE AIRDOS MODEL

The results of radon measurements, performed repeatedly for periods of 10 to 14 days at some 10 monitoring points, are compared with the predicted concentrations based on the Turner classification (Fig 3). The correlation coefficient was 0,73, with 88 % of the individual comparisons agreeing within a factor 2.

A further comparison using observation periods of 60 days at 75 individual points did not show the expected improvement in correlation. The coefficient was 0,5, with 96 % of the individual values agreeing within a factor 2.

Additional measurements were performed in valleys which acted as drainage channels for releases of radon from the tailings dams. Measured concentrations were consistently higher than predicted values by factors of 3 to 7.

CONCLUSION

When the uncertainly in the measurement of radon, which is of the order of 30 %, and the even larger uncertainty in the source term is considered, the comparison between predicted and measured values is most satisfactory. The model predictions used in the assessment of radon impact on the Witwatersrand are considered valid within the stated accuracy limitations and with due recognition for the effect of source proximity and topographical features.

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CRITERIA FOR INTRODUCING INDIVIDUAL DOSIMETRY FOR MINERS EXPOSED TO RADON DAUGHTERS IN NON-URANIUM MINES

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ABSTRACT

In Polish non-uranium mines two systems of routine monitoring for environmental and individual control on radiological hazards exist because of radon and its daughter products. These systems are based on passive dosimetry with track etch detector. Routine environmental control is obligatory in all mines, however, routine individual control of miners' exposure is introduced only in these mines in which the probability of exceeding 3/10 of basic limit exists, according to ICRP recommendation. This paper presents practical criteria for determining this probability, i.e. determines when routine individual control should be introduced parallel as intensivity of environmental control. This makes it possible to make rational and optimal decisions to introduce individual dosimetres of miners. It is very important because the population of miners exposed to radon daughters is very large (in Poland about 300 thousand persons). These criteria result from statistical analysis of routine environmental monitoring and individual exposures. All results of measurement from seven years' study are collected in the computer data base. We presented correlations between some parameters of general environmental hazard estimation and the number of individual exposures exceeding 3/10 of basic limit.The first of these parameters is average yearly radon daughters potential energy concentration and the second one is a fraction of the unit measurement of this concentration exceeding 3/10 DAC obtained in monthly cycles. These parameters were estimated from 25594 numbers of unit measurements. These correlations were determined using Monte-Carlo method for function type $y = a*(x-b)^c$ for $x > b$ and $y = 0$ for $x \leq b$. This correlation shows that the probability of exceeding 3/10 of limit occurs after crossing threshold b. In this way individual dosimetry is introduced only for these miners, whose radiation risk is important from radiation protection point of view.

EFFECTIVENESS OF A 1.5m THICK COVER OF SMELTER SLAG IN REDUCING RADON EMANATION FROM URANIUM TAILINGS.

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ABSTRACT

A 1.5m thick cover of smelter slag was found to reduce radon emanation from uranium mill tailings by a factor of approximately 40.

INTRODUCTION

From 1955 to 1961, the South Australian Department of Mines operated a treatment plant at Port Pirie, approximately 200km north of Adelaide extracting uranium from concentrates from the Departments' Radium Hill mine, 300km to the north-east. Aproximately 200,000 tonnes of concentrate were treated during this time with the tailings and other process residues being stored in dams adjacent to the plant.

In 1982 the State Government decided to rehabilitate the site beginning with a proposal to cover the tailings dams. The 6 tailings dams covering an area of 22 ha were constructed on a clay base with clay retaining walls (See Fig 1.). The uranium tailings were confined to Dams 2,3,4 and 5. The tailings contained approximately 15 TBq of Ra-226 with an activity concentration of 75 Bq/g.

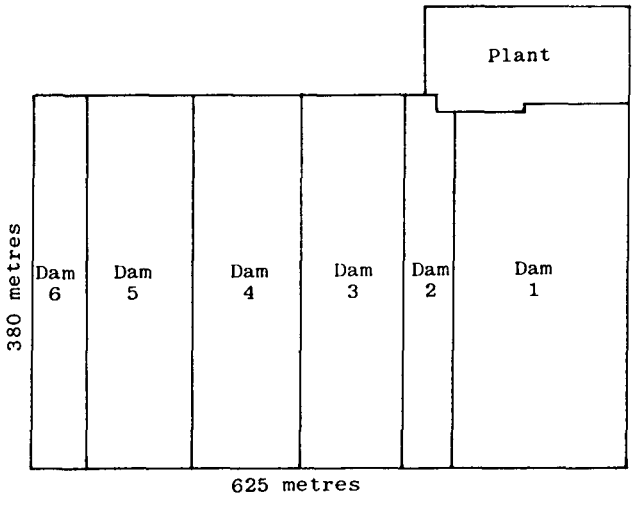

Fig. 1 Plan of Tailings Dam Area

The site is adjacent to a large lead smelter, and large quantities of smelter slag were available virtually free of cost. The slag is a sintered, granular material with 25% greater than 1mm and only 1% less than 100 μm. It has a density of about 1400 kg/m^3 uncompacted and 1900 kg/m^3 compacted, is freely draining and will not maintain a moisture content of more than a few percent.

Published results indicated that it would be a poor cover material as "sandy porous soil" has a reported half value layer for radon emanation reduction of 1.0m [1]. However an initial laboratory test indicated that the HVL for the slag was of the order of 0.3m, and so it was decided to conduct a field trial.

FIELD TRIAL

A test area (15m by 10m) was selected and the pre-cover emanation rate was measured. The area was then covered with slag to depths of 0.5m, 1.0m and 2.0m (see Fig. 2.).

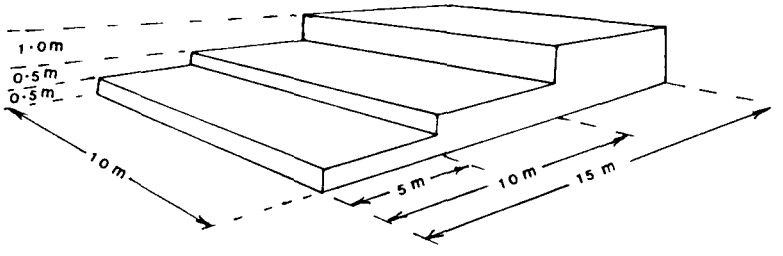

Fig. 2 Test Area Following Laying of Smelter Slag

After covering, emanation rates on each thickness and an uncovered 'control' area were measured. Both the charcoal canister [2] and drum accumulator [3] methods were used to measure emanation rate. The HVLs measured for the various layers were from 0.14m (0.5m cover) to 0.57m (2.0m cover). Some but not all of the variation could be explained by compaction - the 0.5m layer was compacted by vehicle movement in the laying of the other covers [4].

FULL COVER EFFECTIVENESS

As results of the field trial were sufficiently promising, it was decided to proceed with a full 1.5m cover of the tailings. A two year pre cover survey of the dams had shown a mean emanation rate of 5.0 Bq/m^2/s [4].

After the cover had been completed a further survey over a 12 month period indicated that the emanation rate had stabilised at approximately 0.12 $Bq/m^2/s$.

DISCUSSION

The initial slag cover of approximately 1.5m has been very effective in suppressing radon release from the Port Pirie tailings dams. The average annual emanation rate has been reduced by a factor of approximately 40 from 5.0 $Bq/m^2/s$ to 0.12 $Bq/m^2/s$. This represents an effective HVL for compacted slag of 0.28m, a value consistent with the laboratory trials and similar to that of compacted soil (0.3m)[1].

While there are presently no recommended Australian Standards for radon emissions from rehabilitated Uranium tailings piles, the value of 0.12 $Bq/m^2/s$ is well within the United States Environmental Protection Agency radon emission standard of 0.74 $Bq/m^2/s$[5].

The reasons for this relatively large reduction are not known, but an important factor is probably the increased moisture content of the tailings under the cover. The climate of Port Pirie is dry with hot summers, and the uncovered tailings become desiccated. The cover will maintain a higher moisture content in the tailings, so reducing the release of radon[6].

CONCLUSION

While not appearing at first to be a good cover material for Radon suppression, the smelter slag performed surprisingly well. The conclusion to be drawn from this experience: try what you have at hand, it might work!

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METHOD FOR ESTIMATING Rn-222 DAUGHTERS INTAKE FROM THE CONTENT OF Pb-210 IN BONE

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ABSTRACT

Underground uranium miners and other miners are exposed to Rn-222 and its daughters. It is of great concern that how the Rn-222 daughters intakt of miners will be estimated when the informtion of concetration of Rn-222 daughters in air is not available. In this paper a method for estimating the Rn-222 daughters intake from the content of Pb-210 in bone is presented.

The intake-retention function, i.e. the ratio of the content of Pb-210 in bone, Q(t,T), to the potential energy intake of Rn-222 daughters, Ip, is derived from the metabolic models of the ICRP Publication 30. At any time, T, after a constant exposure interval t, the function is given by

$$Q(t,T)/I_p = 0.59 A'_{D'} \sum_{i=1}^{3} D_i (1 - e^{-\lambda_i t}) e^{-\lambda_i T} / \lambda_i t F$$

$$A'_{D'} = (1 + \frac{\lambda_A}{\lambda_B} K_B + \frac{\lambda_A}{\lambda_C} K_B K_C) \lambda_D / [E_A + E_B (\frac{\lambda_A}{\lambda_B} K_B + \frac{\lambda_A}{\lambda_C} K_B K_C)]$$

$$K_j = 1/(1 + \lambda_v/\lambda_j) \qquad (j = A, B, C, D)$$

where, Di and λ_i are the uptake deposition fraction and the effective clearance constant of the bone compartments i, respectively; λ_A to λ_D are the radioactive decay constant of the RaA to RaD; λ_v is the air turnover rate; F is fractional contribution of the short-lived radon daughters inhaled to the content of Pb-210 in bone. Then Rn-222 daughters intake can be estimated from the calculated value of this function and measured content of Pb-210 in bone.

This method has been used to evaluate the Rn-222 daughters intake for 28 uranium miners. The results estimated in this way agreed with in-situ measured concentrations of Rn-222 daughters in air within a factor of 2 and 4 for 68% and 100% of cases, respectively.

A RESERCH IN THE CLEANER FOR REMOVAL OF DUST AND RADON DAUGHTERS

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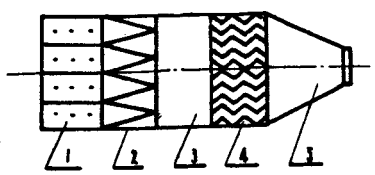
ABSTRACT. In polluted mines by radon and its daughters, the purification of mine air can be accomplished by the removal of dust particles and radon daughters using filters. A multi-stage filter cleaner was developed in Northeast University of Technology in People's Republic of China. Experimental data have indicated that removal efficiency is greater than 99% for dust and greater than 98% for radon daughters. Especially, it is able to withstand high humidity in mines. Its effectiveness has been proved by the data in-situ application in underground workings.

For the purpose of occupational radiation protection at uranium mines and other mines polluted by radon daughters, many researchers and engineers have been paying great attention to looking for a kind of high efficient cleaner. Because radon daughters are infinitesimal and positively charged particulates and can rapidly attach to dust, especially to submicron-dust, suppression of radon daughters is just as suppresion of dust in size of submicron. Obviously, it is impossible to collect such fine particulates by means of the program of common industrial collection dust. As well known, the glass fibre paper filter has a higher efficiency in the cleaning and has been used to protect radon radiation, but it is so difficult, even impossible to use it in underground because of high humidity in mines and low strength of glass-fiber paper. It is necessary that the cleaner has such an ability to withstand high humidity in underground, by all means, also high efficiency and economic cost are taken into account.

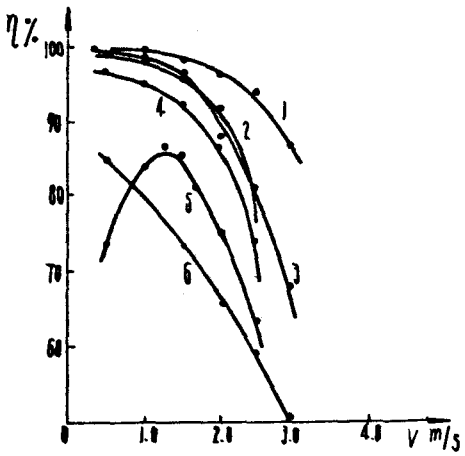
A multi-stage filter developed by Northeast University of Technology is composed of three parts: an electrostatic precharger (ESP), a fiber filter unit and a high efficient filter unit. Its structural scheme is shown in Fig.1. It owns such advantages as high collection efficiency, low noise level with 78db(A) and 75db(A) at the inlet and the outlet of cleaner respectively, high dust loading capacity (1500 g/m^2), long cleaning period of dust. The filter can be reused after being washed dust cake by water spray in assembly state. Its pressure drop is 60-100mmW.G at flow rate 3500-5400m^3/hr.



1. Electrostatic precharger
2. Fibre filter
3. Fan
4. High efficient filter unit
5. Collecting wind device

Fig.1 The structural scheme of multi-stage filter cleaner

Whether the last stage high efficient filter can be used successfully, it deponds on the effectiveness of the prefiltration for the particles in size of submicrons. Because a bit little dust loading will sharp increase the resistance of the high efficient filter, in order to prolong the filter's life, the prefilter should be designed as large dust loading capacity, high filtration efficiency, strong moisture resistance, simple maintenance and long period of dust cleaning. The PW fiberous filter medium is of advantages for the purpose of collecting dust in underground. It has been developed by Ventilation and Safety Engineering Division of NEUT and has been used effectively in industrial dust collection. Its efficiency is always more than 97%. The comparison of six kinds of filters are shown in Fig.2.



1. The filter with electrostatic precharger
2. The thick filter material
3. The series ESP
4. Common ESP
5. The ESP with cross collective plates
6. The thin filter material

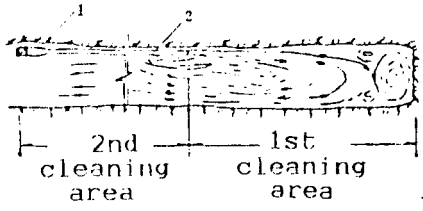
Fig.2 The comparison of the collection efficiency of six kind of filters.

As mentioned above, the PW fiberous filter medium are chosen as a prefilter and assembled in the cleaner to protect the last stage filter from excessive dust loading.

The compound filter's filtration effectiveness only acheived 80-90% because the efficiency of fiberous filter paper is 95% for particles larger than 0.3microns. It is neccessary for acheivement high efficiency that the submicron particles should be condensated as large as possible. As mentioned by Mr.R.L.Rock, the scholar of Denver Technical Support Center U.S.in 1972, "the most interesting aspect of ESP is that it appears that a bonus in air benefication is achieved by removal of condensation nuclei from mine air." So an electrostatic precharger have been assembled for condensation of particle in front of fiberous filter. Experimental data showed that the filtration efficiency for radon daughters was raised to 98%.

The test of the cleaner was carried out in the Heading 1730 at Malage Mine Yunnan Province China. Fig.3 shows the position of test site at the working face. The stope 1730 had an excessive concentration of dust and Radon daughters. Before the test, Radon daughters radioactive potential energy reached 6.4×10^{-5} JM^{-3}. It is 10times more than the National Health Standard Level. The experimental data are summarized in the following Table, which shows the concentration of Radon daughters and dust decrease quickly as long as the cleaner turns on 10 minutes later, the concentration can drop to the Nation-

al Health Standards. The cleaning efficiency in the working space achieved 89.6% and the total collection efficiency of the cleaner for Radon daughters reached 98.85%.



1. The cleaner
2. Ventilating tube

Fig. 3. The position of test site at working face.

sampling location	date time	process	concentration	
			Radon(JM^{-3}) daughters	dust mgm^{-3}
8.5m from face	Aug.7 8:55	loading	$5.6x10^{-3}$ (Initial)	2.28
	9:00		$0.42x10^{-5}$	2.23
	10:35	drilling (two drilling)	$0.37x10^{-3}$	0.75
	Aug.8 16:00		$0.25x10^{-5}$	0.72
15.5m from face	Aug.7 9:00	loading	$1.03x10^{-5}$	1.64
	Aug.8 8:30		$0.26x10^{-5}$	1.2
20m from face	Aug.8 16:00	drilling (two drillers)	$0.33X10^{-5}$	
cleaner inlet outlet	10:35		$1.97X10^{-5}$ $0.024X10^{-5}$	

The concentration of Radon daughters and dust in cleaning process . The outlet of ventilating tube was kept 15m from drilling face in-situ test.

CONCLUSION

Experimental results show that the method of local cleaning is available and reasonable technically for removal of dust and Radon daughters from the contaminated air in mines for the purpose of Alpha radiation protection. The cleaner can remove dust and Radon daughters effectively and make excessive concentrations of dust and Radon daughters decreasing to the National Standards during 3 to 10 minutes in underground atmosphere.Even though double drillers work at the same time, the cleaner can also make the concentration of dust decreasing to The National Standard, 2mg/m, at the working face. When the cleaner is connected with a duct of 90 meters long and the flow rate at the outlet drops to 0.8 m^3/s,the concentrations of both pollutants can still be drained off effectively. The cleaner can be run continuously. Because of its low noise, workers can talk each other freely.

RADON EMANATION MEASUREMENTS FROM URANIUM ORE TAILINGS IN ARGENTINA

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The uranium ore tailings represent important release sources of radon gas (Rn-222) into the atmosphere, being one of the main concerns regarding the environment in the vicinity of uranium mining and milling areas. The tailings area remaining after the mill has ceased operation becomes a long-term source of environmental radioactive contamination due to wind or water erosion and mainly to radon emanation.

Radon gas, with a radioactive half-life of 3.8 days, emanates from the ore tailings as a radioactive decay product of radium-226 (Ra-226, 1600 years), which is generated by the decay of thorium-230 (Th-230), with 8.10^4 years of half-life. Emanation from the tailings arises primarily from radon produced in the surface layer, and up to about 20 % diffuses out of the crystallized structure of the mineral (1). The Rn-222 emanation rate depends on many characteristics, including parents content in the tailings, ore particle size, moisture content, porosity and mineral species (2). Besides it, the gaseous diffusion may be influenced by meteorological factors such as atmospheric pressure, temperature and humidity.

Since 1983, a field team from the Environmental Radioactivity Section of CNEA, comprising the authors of this report, has carried out measurements of the emanation rates of Rn-222 from uranium mining and milling tailings located at different regions in Argentina, with the aim of estimating the resulting radiological impact.

RADON EMANATION RATE MEASUREMENTS

The Rn-222 emanation measurements were made by using the technique develloped by R.J. Countess (3). In this method, cilindric canisters containing activated charcoal are positioned on the tailing surface for a known period, to collect radon released from it. After collection, the canisters are stored for at least three hours to allow Bi-214, a gamma-ray emitting daughter, to grow towards equilibrium with the parent radon. The Bi-214 activity is counted through the 609 keV gamma emission because of its clear separation from other emission lines and the low background in such spectral area. The counting is carried out inside a lead-brick shield, with a 4 x 5-in. NaI crystal detector coupled to a multichannel pulse-height analyzer.
Rn-222 emanation rate is estimated taking into account the measured Bi-214 activity corrected for decay during the storage period, the sampling time and the exposed surface.
Assuming a charcoal adsorption efficiency similar to that reported by other authors,(4,5), a lower detection limit of 0.15 $Bq.m^{-2}.s^{-1}$ is obtained for the experimental conditions, with a statistical error of 10%.

RADIUM-226 CONTENT OF TAILINGS

The radium content of the sterile material beneath each site of emanation measurement was determined. Samples were taken from under the radon collectors located at each site, dried to determine the moisture content, homogenized and packed in small plastic bottles, sealed, and stored for allowing them to reach radioactive equilibrium. Afterwards, the samples were placed into a Ge(Li) spectrometer for analysis. Ra-226 content was estimated by using the data from Pb-214 and Bi-214 peaks in the resulting gamma spectra.

DOSE ASSESSMENTS

The uranium mining and milling sites are usually located at low population density areas. This is the case for all the reported areas but one, Malargue, which is a milling installation situated near from the homonymous city (about 500 m).
The individual and collective dose commitments from radon releases were estimated using the methodology presented by UNSCEAR (1). For the annual effective dose equivalent assessments to the most exposed members of the public, the model used assumes an atmospheric dilution factor of 5.10^{-6} s m^{-3} at 500 m from the sources, where the critical groups are supposed to live.
The collective effective dose equivalent commitments from these tailings were estimated by using a simplified model, taking into account the natural radon emanation from soil and the corresponding radon concentration in air. These collective doses were calculated over the mean life of Th-230 ($1.1\ 10^{5}$ y).

RESULTS

Measurements of radon emanation rate and radium content were made at four sites in Argentine uranium mining and milling areas. For each site, the corresponding distributions of Rn-222 emanation rates are shown in Figs. 1 to 4.
In Table 1, the Ra-226 concentrations are compared to the corresponding Rn-222 emanation rates.

TABLE 1. Rn-222 emanation rates

SITE	IMPOUNDMENT AREA (ha)	Rn-222 EMANATION RATE (Bq/m2 s)	Ra-226 CONC. (Bq/g)	RATIO
S.Rafael	5.0	7.7	8.1	0.95
Malargue	3.0	5.8	8.1	0.71
Salta	7.5	20.6	9.4	2.19
Chubut	7.0	2.7	4.9	0.56

The calculated doses to most exposed members of the public, expressed as annual effective dose equivalents, and the collective effective dose equivalent commitment for each site are listed in Table 2.

TABLE 2. Individual and collective dose commitments

SITE	Rn-222 RELEASE TBq/a	INDIV.DOSE uSv/a	COLL. DOSE (NORMALIZED) man Sv	man Sv/GW(e).a
S.Rafael	12.0	108.0	8.6E+3	3.0E+3
Malargue	5.5	49.6	4.0E+3	7.0E+2
Salta	49.0	441.2	3.5E+4	1.0E+4
Chubut	6.0	54.0	4.3E+3	3.6E+3

CONCLUDING REMARKS

The calculated specific radon emanation rate, the ratio of the radon emanation rate to Ra-226 concentration, varied from site to site, ranging from 0.56 to 2.19 Bq m^{-2} s^{-1} per Bq g^{-1} (Table 1). These values are compatible with those published by UNSCEAR, having into account that the values depend on meteorological conditions, such as wind speed, atmospheric stability and rainfall (1). Other factors that may affect this ratio are the spacial distribution of radium, and radon transport parameters, particularly moisture (5).
The above expressed explains the difference observed in the specific radon emanation rates for San Rafael and Malargue tailings where both radium concentrations are similar.
In the case of Salta, where the ratio is the highest, geographical and meteorological factors contribute to increase the radon emanation rate.

The estimated doses are in agreement with UNSCEAR reports. The individual doses to critical groups represent only a few percent of overexposure to natural radiation.
Collective dose commitments are extremely uncertain and highly speculative because of the assumptions of the duration of the constant release and of the fixed population density and habits. Furthermore, present day tailings managements may lead to radon emanation rates lower than the ambient levels for soils in the mill vicinity,so that almost no long term dose commitment arises.

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Fig. 1

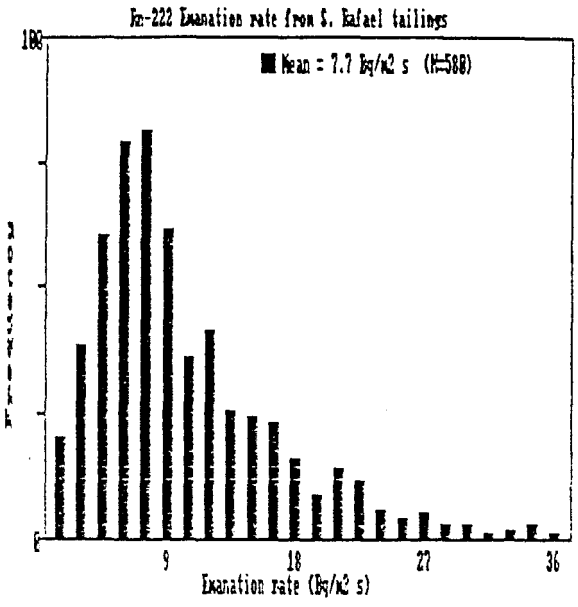


Fig. 2

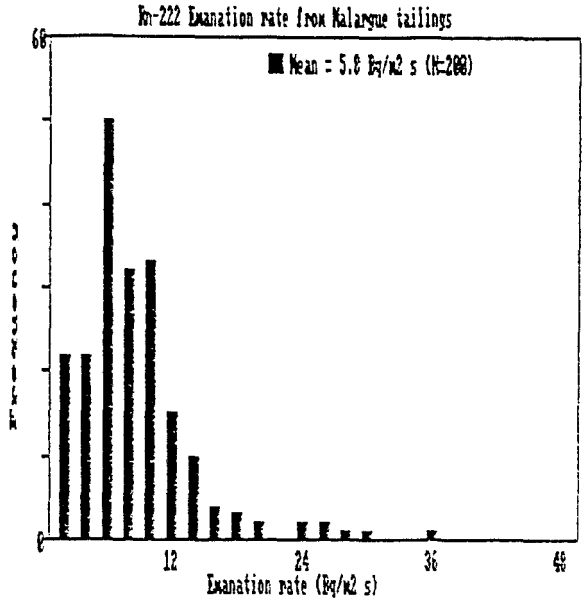


Fig. 3

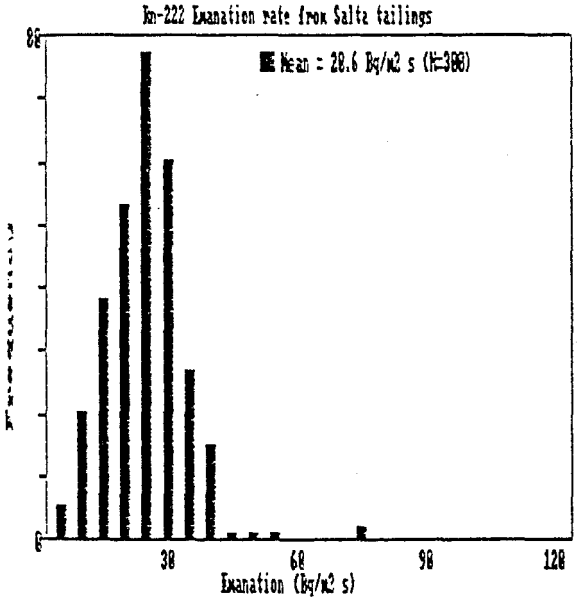
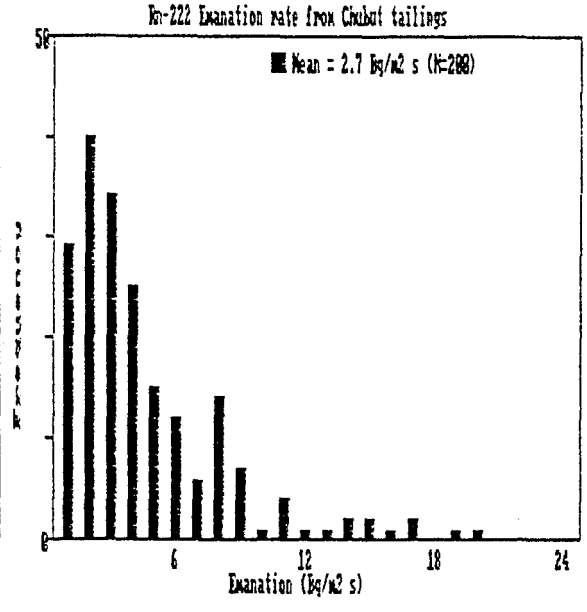


Fig. 4



IONS, ELECTRIC FIELDS AND RADON DAUGHTERS
EFFECTS OF FILTRATION AND ELECTROSTATIC PLATEOUT

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INTRODUCTION

Over the last decade several procedures have been suggested for reducing the exposure of the public to radon and more specifically to its short-lived daughter products.
Most of these procedures are aimed at lowering the radon level either by reducing the radon entry or by diluting the radon-rich indoor air with radon-free or radon-poor outdoor air.
A direct removal, by mechanical or chemical methods, of radon from the air, is so far not practical, although theoretically possible.
The radon daughters, on the other hand, being non-gaseous molecular clusters or particle-attached atoms, can be affected by such procedures as filtration and electrostatic plateout.

FILTRATION

The effect of filtration on airborne radon daughters has been described in several reports (1,2,3)
Normally the change in the absolute exposure rate, as expressed by the potential alpha energy concentration (PAEC), or, since the radon level may exhibit uncontrollable variations, the change in the normalized exposure rate, as expressed by the equilibrium factor, is used as a measure of the efficiency of the filtration process (4).
Such a procedure, however, does not take into account that the filtration process in itself by its very nature of removing particles from the air may change the partitioning between attached and unattached daughter products and thereby, even for constant PAEC, may change the radiological dose, which the exposure to the radon daughter bearing atmosphere will cause to the respiratory tract.
The relevant dose can be calculated from the individual daughter concentrations and their unattached fractions, using the James-Birchall-(5), the Harley-Pasternack-(6,7) or the Jacobi-Eisfeld-models (8) chosing a given age or activity group.
The common method today, however, is to calculate the doses from the PAEC of the attached and the unattached daughters respectively (9).

IONIZATION AND ELECTROSTATIC PLATEOUT

It has long been known (10,11) that airborne radon daughters to some degree are charged, and a suitable electric field in a room may thus remove radon daughters from the air, more or less in the same way as a filtering device.
The effect of electric fields on the concentration of airborne radon daughters depends primarily upon the field strength at the boundary of the room and the fraction and mobility distribution of the charged daughter products.
An electric field in a room is usually established between a fairly small electrode in the interior of the room with the walls as the grounded counter electrode. If the electrode is kept at a voltage V and the distance from the electrode to the walls is d, the mean field strength, E_m, in the room is $E_m = V/d$.
In practice, however, the voltage drops off rapidly in the immediate vicinity of the electrode with the result that the air in the major part of the room is exposed to a field of much smaller strength than that given by the mean value.
If, on the other hand, the electrode emits ions by a corona discharge a space charge will be set up in the room modifying the field in such a way that the field strength in general decreases near the electrode and consequently increases in the rest of the room (12).

The effect of the ionization, however, is not limited to increase the field strength at the walls. Experimental results (13) suggest that the charged fraction of the radon daughters also increases, when the atmosphere is exposed to a unipolar space charge. The same results also demonstrate that the plateout efficiency of a positive space charge, i.e. a field directed towards the walls, is considerably higher than that of a negative space charge, indicating that a larger fraction of the radon daughters is originally (without the space charge) positively charged.

EXPERIMENTAL RESULTS OF FILTRATION AND/OR ELECTROSTATIC PLATEOUT

In order to examine the remedial potentials of filtration and electrostatic plateout or possibly a combination of the two processes the results of several series of experiments are reviewed in the following.

The experiments were performed in a 150 m^3 room with a radon concentration of about 1000-1500 Bq/m^3. With the filtration experiments various types of electro filters were operated in the room and with the ionization/electric field experiments four corona emitters, spaced approximately 1.5 m apart, were mounted about 0.5 m from the ceiling. The emitters were through a high ohmic resistor connected to a high voltage supply, and the voltage drop across the resistor measured by a floating static voltmeter to give the corona current.

The following parameters were measured: radon concentration, by grab sampling and counting in scintillation cells, individual daughter activities and unattached fractions, by alpha spectroscopy of membrane filters and wire screens, aerosol concentration and AMD, by condensation nucleus counter and diffusion boxes, and corona current as described above. With the ionization experiments also plateout activities on grounded discs were measured (13). These results will not be dealt with here.

During the measurements an aerosol concentration of about $5 \cdot 10^{10}$ m^{-3} with an AMD of about 0.1 µm was maintained in the room.

In Figures 1 and 2 some of the results of the filtration and ionization/electrostatic plateout experiments, respectively, are shown. On the right hand side the equilibrium factor F_t, for the total daughter population, as well as the equilibrium factor F_u, for the unattached daughters, are plotted. On the left hand side the normalized mean bronchial doses from the total daughter population and from the unattached daughters are shown, calculated in two ways. The lower curves are derived from individual daughter relative activities and their unattached fractions, using the three dose models and averaging over six age and activity groups (4). The upper curves are calculated from the equilibrium factors (normalized PAEC) F_t and F_u. These doses correspond most closely to those calculated from the individual daughter relative activities by the use of the James-Birchall- and the Harley-Pasternack-models for adults with high breathing rates (9). It should be mentioned that the filtration rates plotted in Figure 1 are nominal factory settings of the filtration device.

With the experiments plotted in Figure 2 positive emitter voltages were used.

In Figure 3 finally a compilation of results of the effect on equilibrium factors and doses (total and unattached) of various filtration and/or ionization treatments is shown.

DISCUSSION OF THE RESULTS

It appears from the results shown in Figures 1, 2, and 3 that the effect of filtration and/or ionization assisted electrostatic plateout on the PAEC (as expressed by the equilibrium factors) and the bronchial doses is much more pronounced for the attached daughters than for the unattached ones.

As an example we see with the electrostatic plateout experiments, Figure 2, that the unattached equilibrium factor, F_u, stays almost constant about 0.03, while the total equilibrium factor, F_t, decreases from about 0.42 at an emitter voltage of 0 to approximately 0.07 at an emitter voltage of 30 kV. It also appears that simultaneously the total dose has decreased to about half of its original value, while the dose from the unattached daughters has only decreased slightly. At an emitter voltage of 30 kV about 87 % of the remaining dose is due to the unattached daughters.

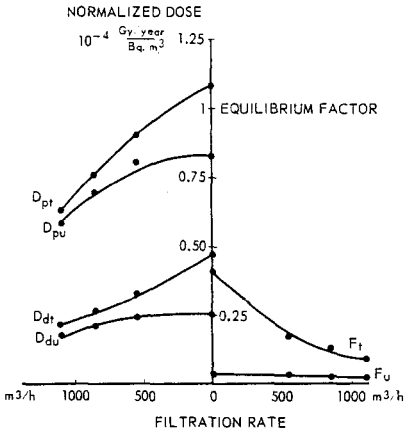

Figure 1. Equilibrium factor and normalized mean bronchial dose as a function of the filtration rate

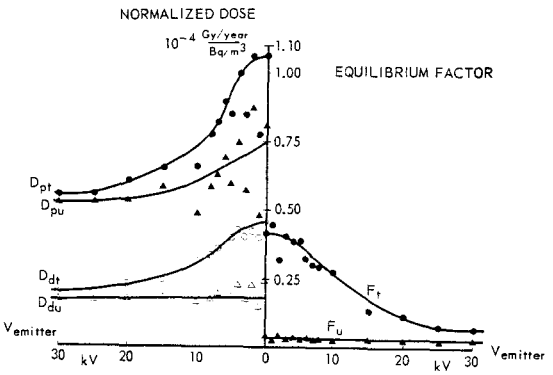

Figure 2. Equilibrium factor and normalized mean bronchial dose as a function of the emitter voltage (13)

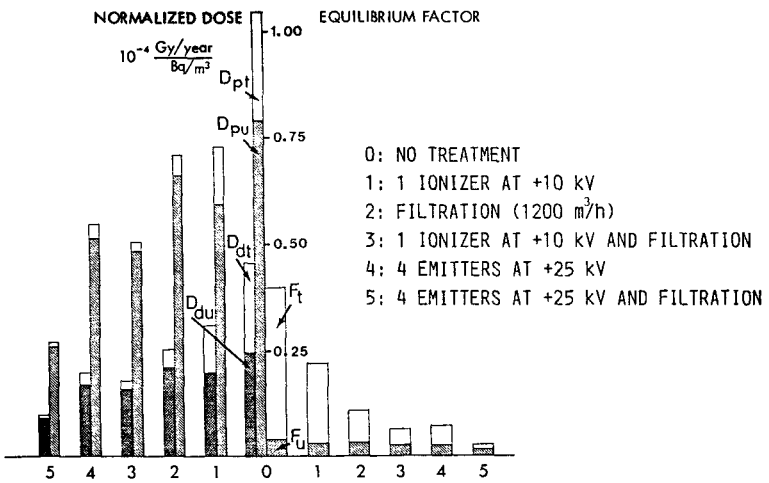

Figure 3. Equilibrium factor and normalized mean bronchial dose for various combinations of filtration and/or ionization.

F_t: equilibrium factor of attached + unattached radon daughters.
F_u: equilibrium factor of unattached radon daughters
D_{pt}: dose calculated from normalized total potential alpha energy concentration
D_{pu}: dose calculated from normalized unattached potential alpha energy concentration
D_{dt}: dose calculated from total individual radon daughter activities
D_{du}: dose calculated from individual unattached radon daughter activities

CONCLUSION

It has been demonstrated that it is possible by the use of filtration and/or electrostatic plateout to lower both the PAEC and the bronchial dose from airborne radon daughters. It appears that only the activity of the attached daughters are affected to any significant degree by the air treatment. This does not necessarily mean that unattached daughters are not being removed, but rather that the removal process itself, by removing aerosol particles from the air, causes more daughter products to stay unattached and thus compensates for the anattached removal.

By the experiments described the maximum reduction in PAEC was 78 % for filtration, 87 % for electrostatic plateout and 94 % for a combination of both processes. The corresponding reductions in doses were 48 % for filtration, 50 % for electrostatic plateout and 74 % for the combination.

It should be mentioned that the experiments were made under laboratory conditions, but that similar measurements under practical conditions seem to give similar results.

ACKNOWLEDGMENT
The support of this work by CEC under contract B16-F-113-DK and by Vattenfall under contract 60713 is gratefully acknowledged.

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COMBINED EFFECT OF FILTRATION AND UNIPOLAR AIR IONIZATION ON RADON PROGENY

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ABSTRACT

Unipolar air ionization has been used indoors mainly because of the claimed beneficial health effects. The biological and physiological importance of air ions is not generally accepted and it seems that air ionization has only minor, if any, direct effects on human beings. Unipolar ionization has, however, a clear effect on aerosol particles and — as been found out in recent studies — on radon decay products. Because of the attachement of radon daughters to aerosol particles the deposition of particles caused by the ionization results also in a decrease of the equilibrium factor. Depending on the ionization polarity also the unattached (electrically charged) daughters are removed to a great extent. These effects can be reinforced by mixing the air with a fan.

Air filtration has also been used to lower the radon progeny concentration. This reduces the attached fraction, which causes a decrease in the total equilibrium factor. As the amount of unattached daughters increases, the decrease in the dose is much smaller. Electrostatic precipitators commonly used for air filtration in dwellings can cause particle formation. This affects the radon progeny when no other particle sources are precent, because the free daughters are attached to these small particles.

This study is concentrated on the combined effect of filtration and unipolar ionization on radon progeny. This combination has several advantages:
-air cleaning minimizes the dirt accumulation on surfaces caused by ionization
-the charged fraction of free progeny is removed by ionization
-air mixing by filtration reinforces the effect of ionization.

Measurements made in a laboratory test room show that the combined effect of ionization and filtering is larger than from either of them alone. The difference of electrostatic precipitator and mechanical filter is studied by using two cleaners with identical flow rates. The effect caused by the particle formation by electrostatic precipitator is observed in clean air.

ON THE MULTIPOINT RADON MONITORING SYSTEM IN THE MINE

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ABSTRACT

The multipoint continuous radon monitoring system which is based on the two filter method has been developed to estimate the radon concentration at four points of working area in uranium mine of Japan. The system consists of the detecting device which is distributed to the four working or interesting places of the mine, double shielded coaxial cable, analysing device located at the ground surface of mine and control device. The detecting device is divided to the following parts : seven liter of cylindrical vessel including the removal filter and daughter collecting filter (millipore AA) placed at the downstream side of the vessel, diaphragm pump adjusted to the flow rate of 7.7 l/min., shielded alpha particle detecting unit including the silicon surface barrier detector which is located at the face of daughter collecting filter, booster amplifier, bias voltage supply and power supply. About 400 m in length of double shielded coaxial cables are provided to make the electrical connection between the output of booster amplifier to input of linear amplifier of analysing device. Input pulses for the analysing device are divided to the four groups by the mixer router which is provided in the 1K multichannel pulse height analyser and are accumulated to each 256 channels group. The accumulated information is transferred to the memory of micro computer and then calculated for each 50 min. accumulating period. The conversion factor of the system is estimated as about 0.23 pCi/l for 10 counts per 50 min. collecting time of RaA channel.

RESPONSE OF CONTINUOUS WORKING LEVEL MONITORS TO TRANSIENT RADON CONCENTRATIONS

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INTRODUCTION

The mathematical model developed earlier for a continuous working level monitor (1) (Kawaji, M., Pai, H.L. and Phillips, C.R., Use of gross filter activities in a continuous working level monitor, Health Phys. 40(4), 543-548 (1981)), assumes constant radon activity during the in-growth time (age of the air). In any real case, however, the radon activity may be expected to fluctuate over that time. Conversion factors from gross counts to working level (WL) may therefore change over time. Two simple cases are investigated here, a step change in radon concentration and a sine wave fluctuation (frequency ω and amplitude R). The two cases are examined in order to estimate the error in applying conversion factors derived for a constant radon concentration and in order to determine the best gross count mode, α, β or $\alpha + \beta$, for monitoring the working level of radon progeny.

RESULTS

Case I

The radon progeny activities are described by the superposition of solutions at constant radon activities at two different time scales, s and s - a. The activities depend on the relative change in radon activity, R, and the time of occurrence of the step change, a. (R varies from 0 to 1.) The α/WL, β/WL and $(\alpha + \beta)$/WL ratios are dependent slightly on both R and a at the time when the step change takes place; but after the time s > a these ratios are independent of R and a and approach those calculated for the case of a constant radon source. The variation of the three ratios about the ratios derived from the constant radon source (1) at the time of the step is about 1% for α/WL, 3% for β/WL and 2% about $(\alpha + \beta)$/WL (R = 0 to 1) (Table 1).

TABLE 1. Variation of Activity to WL Ratios

Ratio	Step Change (age of air = 5 min)	Sine Wave (ω, rad min^{-1}) (age of air > 20 min)	
		ω = 0.1	ω = 1
α/WL	~ 1%	3%	2%
β/WL	~ 3%	8%	7%
$(\alpha+\beta)$/WL	~ 2%	5%	5%

Case II

Radon progeny activities depend on two parameters, the amplitude, R, and the frequency of the sine wave function of the radon activity. The frequency, ω, ranges from 0 to 10 min^{-1} and the amplitude from 0 to 1. When ω or R is equal to zero, the activities of the radon progeny correspond to activities from a constant radon source with mean activity $\overline{\lambda_i C_i}$, where C_i is the atmospheric concentration of the ith member of the radon decay chain in atoms/cm^3; λ_i is the decay constant; i = 1 for ^{222}Rn (radon), 2 for ^{218}Po (RaA), 3 for ^{214}Pb (RaB), 4 for ^{214}Bi (RaC). The time constant, τ_i, for the decay and buildup process, is characterized by $1/\lambda_i$ which is defined as the mean life of the ith member of the decay chain. The radon progeny activities (^{218}Po, ^{214}Pb and ^{214}Bi) are shown in Figs. 1, 2, and 3 respectively for R = 1. The radon progeny activities describe a stable sine wave function for the period $T/2\pi > \tau_i$ which can be seen in the curves corresponding to ω = 0.1 min^{-1} in Fig. 1. The peak of the sine wave is greater than 1 since the period T is long enough for each progeny to respond. In the case of $T/2\pi < \tau_i$, which corresponds to ω = 1 min^{-1} in Fig. 1, and ω = 0.1 min^{-1} in Figs. 2 and 3, the effect of the transient persists over many cycles but its amplitude is small. For $T/2\pi << \tau_i$, which corresponds to the curves ω = 10 min^{-1} in Fig. 1, and ω = 1, 10 min^{-1} in Figs. 2 and 3, the curves approach the growth of activity of each progeny accumulating with a constant radon source. At high frequency, there is insufficient time for each progeny to decay. The effect of amplitude R is small. When R approaches zero, the curves resemble those for a constant radon source.

The ratios α/WL, β/WL and $(\alpha+\beta)$/WL are calculated for various values of in-growth time in Figs. 4, 5 and 6 respectively for R = 1. The ratios also exhibit behaviour with respect to ω similar to that of the radon progeny activities. However there is a cancelling effect of radon progeny activities in both the numerator and the denominator of the ratios. For ω = 10 min^{-1}, the curves are close to those given by Kawaji, Pai and Phillips (1) for the case of constant radon activity. For an ingrowth time greater than about 20 minutes, the α/WL, β/WL and $(\alpha+\beta)$/WL ratios do not fluctuate significantly. Table 1 shows the variation of the three ratios about the constant ratios α/WL = 17160, β/WL = 26000, $(\alpha+\beta)$/WL = 43100 (1).

The small variation of the α/WL ratio (less than 3%) with the sine wave fluctuation of radon activity makes the alpha monitoring mode the best for a direct reading continuous WL monitor. Even for an in-growth time of less than 20 minutes, the α/WL ratio is within 10% of the constant α/WL ratio of 17160. In the beta monitoring mode, the β/WL ratio varies by about 8% for ingrowth times greater than 20 minutes.

The α/WL, β/WL and $(\alpha+\beta)$/WL ratios are fairly insensitive to the amplitude, R. As R approaches zero, the three ratios approach the curves given for ω = 10 min^{-1} in Figs. 4, 5 and 6.

CONCLUSIONS

Count to Working Level ratios, or conversion factors, for a gross activity continuous WL monitor derived for the case of a constant radon gas concentration (1) can be applied to the situation in which the radon concentration undergoes a step change. When the radon gas concentration follows a sine wave fluctuation, the conversion factors α/WL, β/WL and $(\alpha+\beta)$/WL depend on ω. For a sine wave period $T/2\pi >$ the mean life, τ_i $(= 1/\lambda_i)$, of the radon progeny i, the transient activity becomes insignificant after only a fraction of a cycle of the sine wave. For $T/2\pi < \tau_i$, the transient remains after many cycles but its magnitude is small. For $T << \tau_i$, the three conversion factors approach values for a constant radon activity. The activity/WL ratios are fairly insensitive to the amplitude of the radon activity sine wave, R. As R approaches zero, the ratios, as expected, approach the values given by a constant radon activity. Of the counting modes examined, gross alpha counting is the best for both the sine wave and the step change in the radon activity, based on the relative constancy of α/WL over a long ingrowth time, assuming nearly equal counting efficiencies for ^{218}Po and ^{214}Po alpha particles. For an in-growth time greater than 20 min, the α/WL ratio varies by less than 3% from 17160 for both the sine wave and step change in the radon concentration.

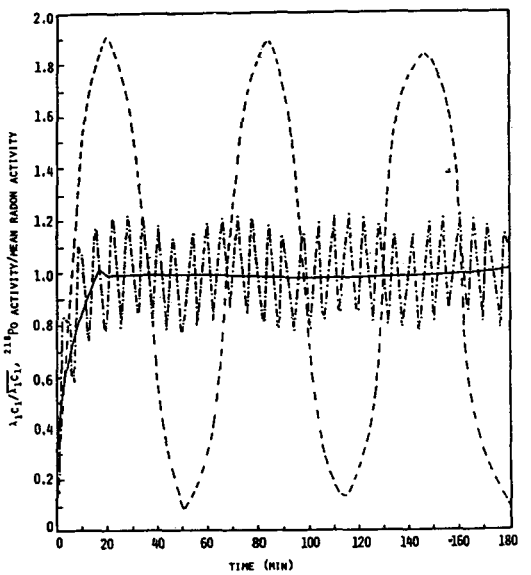

Fig. 1. Growth of ^{218}Po activity in the sine wave fluctuation of radon activity (R = 1).
----- ω = 0.1 min^{-1}
—.— ω = 1 min^{-1}
——— ω = 10 min^{-1}

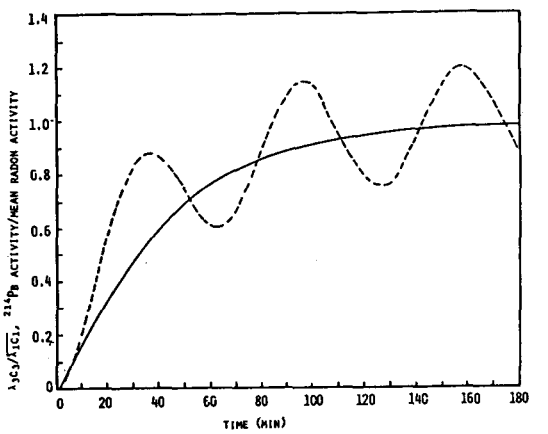

Fig. 2. Growth of ^{214}Pb activity in the sine wave fluctuation of radon activity (R = 1).
----- ω = 0.1 min^{-1}
——— ω = 1, 10 min^{-1}

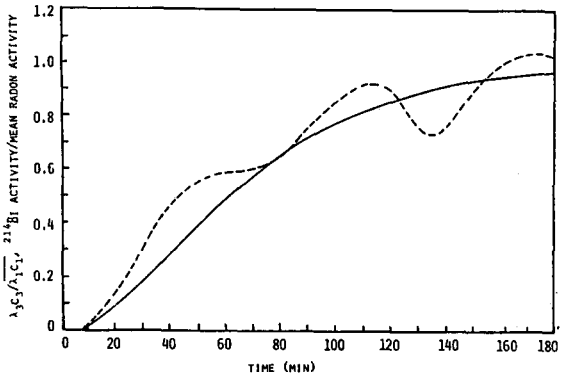

Fig. 3. Growth of ^{214}Bi (or ^{214}Po) activity in the sine wave fluctuation of radon activity (R = 1).
----- ω = 0.1 min^{-1}
——— ω = 1, 10 min^{-1}

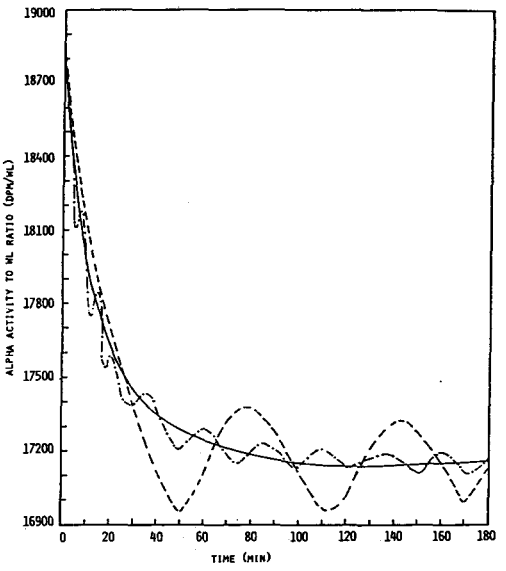

Fig. 4. Alpha activity to WL ratio as a function of the age of the air (R = 1, sine wave response).
----- ω = 0.1 min^{-1}
—·— ω = 1 min^{-1}
——— ω = 10 min^{-1}

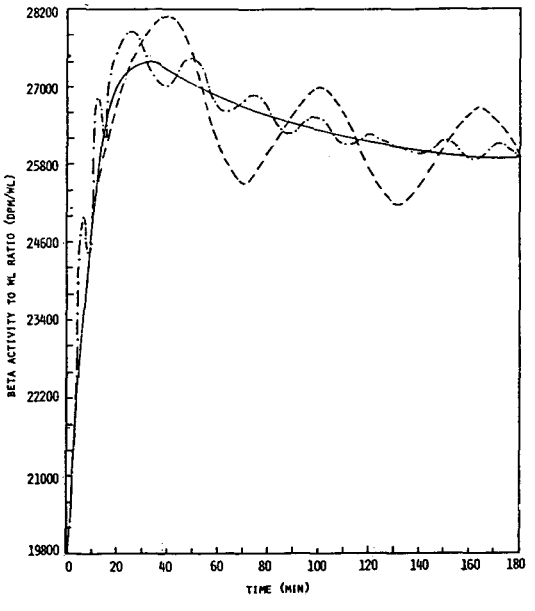

Fig. 5. Beta activity to WL ratio as a function of the age of the air (R = 1, sine wave response).
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—·— ω = 1 min^{-1}
——— ω = 10 min^{-1}

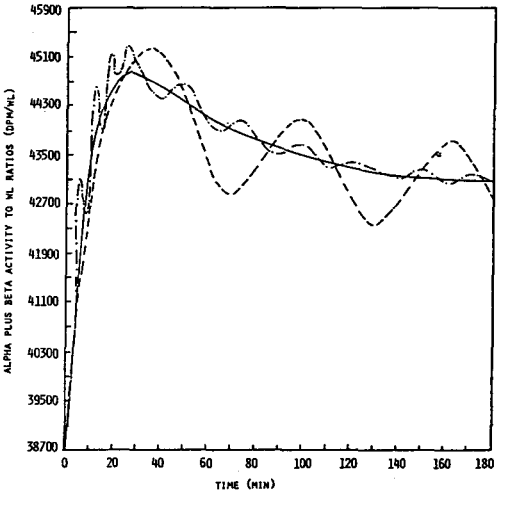

Fig. 6. Alpha plus beta activity to WL ratio as a function of the age of the air (R = 1, sine wave response).
----- ω = 0.1 min^{-1}
—·— ω = 1 min^{-1}
——— ω = 10 min^{-1}

RADON MEASUREMENTS IN GREECE

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INTRODUCTION

The naturally occuring radioactive gas Radon-222 is formed by the decay of Radium-226, which is present at low activities in most soils (about 20-30 Bq/Kg), with the exception of uraniferous deposits, where the concentration is much higher (UNSCEAR 1984). Radon usually enters the atmosphere by diffusion out of the soil, but its inhalation may have a transport component attributed to falling atmosphere pressure, which might occur either outdoors or indoors. Indoors concentrations are higher than outdoors. The main source of radon in houses appears to be the soil. Quantitavely, radon is the most significant source of collective dose to the earth's population (UNSCEAR 1982). At the same time it is a controllable natural source, unlike other natural sources, as e.g. potassium-40 which is under homeostatic control in the human body and its isotopic composition is constant.

For the above reasons, extensive surveys of radon concentration in spas, mines, caves and most important, in dwellings, have been carried out in many countries (Wrixon et al 1984, Swedjemark and Mjokes 1984, Letourneau et al 1984, Schmier and Wicke 1985, Green et al 1985, McAuley and McLaughlin 1985). A possible association between lung cancer and geological outcrop has been postulated (Fleischer 1986). In almost all the studies, a small percentage of dwellings were found to have a very high radon concentration.

Studies of radon concentration in greek spas (Danali et al 1986), in a cave (Papastefanou et al 1986), in constituents of the greek cement (Maraziotis 1985), in building materials in Greece (Siotis and Wrixon 1984, Papastefanou et al 1984) and in greek mines (Georgiou et al 1986) have been published. Some preliminary studies of radon concentration in greek dwellings have been published (Papastefanou er al 1984, Proukakis et al 1987).

MATERIALS AND METHODS

In order to get an idea of the problem in Greece we decided to carry out a national survey. Two different sites were selected: Athens, where domicile about 40% of the greek population and Domatia, a small village in northern Greece 600 Km from Athens, located in an area known to have soil with increased uranium concentration. Also, another series of measurements were carried out in ancient (5th century B.C.) and contemporary silver mines in Laurium, a town 50 Km south of Athens. It is from these mines that the ancient town of Athens was exploiting the silver. About 1000 ancient mining shafts and tunnels have been preserved until now. Some of them were extended last century and used until 1963.

Passive detectors measuring intergrated alpha sxposure from radon (Track etch, Terradex Co.) were used for our measurements.

In Laurium mines, 30 spots were selected, comparable in terms of depth and geological setting, divided in 3 groups (ancient, contemporary sites and modern tunnels pathed along ancient ones). Ventilation data for these spots were available. In Athens, modern apartments of the same economical and social level have been chosen. The detectors were installed in first floor apartments of building blocks most of which have been built in the seventies. We have chosen first floor apartments because the propability to find increased radon concentration was greater there than in higher levels of the same building, provided that the ventilation remains the same. So, the values that we measured ought to be the maximum for the given building under identical ventilation conditions. In Domatia, the houses are built in a traditional way, that is from local stones. In all houses, the detectors were installed in bedrooms, where people spend aproximately one third of their lifetime, for a period of six months (Spring - Summer - Autumn). During this hot period of the year, house's ventilation is high since people leave windows open.

RESULTS

Twelve of the dosimeters installed in Laurium have been lost (16% missing). The results of the measurements in the remaining are shown in Tables 1,2 and 3 for Laurium, Athens and Domatia respectively. Mean values ($\pm$SE) expressed in Bq/m^3 were: for Laurium 2828.4 $\pm$ 799, for Athens 24.0 $\pm$ 3.5 and for Domatia 135.7 $\pm$ 26.2 with ranges 8 - 10095.2 Bq/m^3, 3 - 136.2 Bq/m^3 and 7.4 - 492.1 Bq/m^3 respectively.

Table 1

Radon Concentration in Ancient and Contemporary Greek Silver Mines (Laurium, Attiki)

Spot	Bq/m^3	Spot	Bq/m^3
1	780.6	10	7186.7
2	222.6	11	8508.7
3	6922.2	12	4342.2
4	96.9	13	4927.1
5	10095.2	14	305.1
6	406.4	15	110.2
7	342.9	16	238.1
8	717.1	17	28.6
9	5672.3	18	8.0

DISCUSSION

The large differences of radon concentration found among the different shafts in Laurium are probably due to differences in ventilation, as there is a strong negative correlation (r= -0.98) between air flow and radiation exposure of the dosimeters. It is concluded that in ancient tunnels, due to their ventilation network still operating after 2500 years radiation exposure is maintained in comparable levels with that of contemporary mines.

Table 2

Radon Concentration in Athenian Dwellings

Spot	Bq/m^3	Spot	Bq/m^3	Spot	Bq/m^3
1	33.3	20	18.5	39	9.3
2	14.8	21	4.5	40	12.2
3	11.1	22	16.3	41	9.6
4	18.5	23	6.3	42	12.6
5	88.8	24	3.0	43	15.9
6	25.9	25	40.7	44	16.7
7	55.5	26	8.1	45	69.9
8	62.9	27	9.6	46	136.2
9	70.3	28	11.1	47	82.6
10	33.3	29	5.6	48	9.3
11	44.4	30	7.4	49	30.0
12	7.4	31	3.7	50	35.5
13	18.5	32	5.5	51	7.4
14	14.8	33	12.6	52	6.6
15	22.2	34	8.5	53	10.4
16	7.4	35	17.4	54	9.3
17	7.4	36	37.4	55	17.4
18	48.1	37	4.8		
19	14.8	38	8.9		

Table 3

Radon Concentration in Dwellings in the village Domatia (Macedonia)

Spot	Bq/m^3	Spot	Bq/m^3
1	88.8	10	7.4
2	492.1	11	162.8
3	111.0	12	122.1
4	37.0	13	122.1
5	318.2	14	129.5
6	144.3	15	22.2
7	85.1	16	66.6
8	162.8	17	136.9
9	48.1	18	185.0

In all but one (No.46) cases from measurements held in Athens, radon levels were lower than those reported in other countries (Gessel 1983, Schmier and Wicke 1985), a finding partially explained from the high ventilation during the time of measurements. Our values were 3 fold higher than the annual average of the population - weighted radon concentration in outdoor air (UNSCEAR 1984). Our results are in agreement with those of Proukakis et al (1987). Papastefanou et al (1984), reported radon measurements performed by solid-state nuclear track detectors in 15 dwellings made of bricks and concrete located in northern Greece and a mean concentration of 148 Bq/m^3 with a range between 104-381 Bq/m^3 was reported.

As far as Domatia measurements are concerned, our results clearly indicate that there is a large difference between Athens and Domatia and that in Domatia there are some dwellings exceeding the action level for remedial action of EPA (Hanson 1987) and of the Nordic countries (Nordic countries 1986).

Our future plans include surveys of other areas in Greece and studies correlating the indoor radon concentration with the uranium concentration of the soil around the house.

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ENVIRONMENTAL IMPACT OF URANIUM MINING IN THE VICINITY OF THE GRAND CANYON NATIONAL PARK

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INTRODUCTION

The area around the Grand Canyon is rich in uranium mineralization, with many of the deposits existing as unique breccia pipe formations located about 300 meters below the surface. The pipe deposits average 0.75% uranium, but concentrations in excess of 4% to 20% are not uncommon. Each deposit is capable of providing a 30 year supply of electricity to a city of one million people. Figure 1 is a diagram of the Colorado River watershed and identifies the important areas where uranium may be found.

At present there are three active underground mines, another four breccia pipes in various stages of development, and thousands of registered claims. One of the mine sites is located on the south rim of the canyon, while the others are located on the north rim in the vicinity of Kanab Creek.

The Grand Canyon National Park borders the Colorado River. Although no mining is currently allowed inside the Park boundaries, there is one abandoned uranium-copper mine located on the south rim near the headwaters of the normally dry Horn Creek wash. Erosion caused by wind and the Colorado River and its tributaries has exposed several breccia pipe deposits inside the canyon.

There are numerous surface and near surface deposits of lower grade uranium along the Little Colorado River. They are not breccia pipe formations.

The Colorado River is a major source of water for a variety of public uses. The water is used for irrigation in Arizona and southeastern California. The recently completed Central Arizona Project canal diverts a considerable amount of the water to the Phoenix metropolitan area. Finally, the river itself and Lake Mead, which is fed by the Colorado River, are both major recreational areas.

In order to assess the impact from existing exposed deposits and the increased uranium mining of breccia pipe formations around the Park, a rigorous environmental assessment program is being conducted by the ASU Radiation

Measurements Facility. Assessments include routine sampling of air, water, and soil, and measurements of direct radiation and radon releases from the mine vents and ore piles. The monitoring will be used to quantify local changes in the natural radiation environment, to quantify any cumulative effects of radon releases from multiple mines, and to provide baseline measurements for use in the final decontamination and decommissioning of the mine site. Each mine has an individualized radiological assessment program which is reviewed and approved by the U.S. Forest Service or the U.S. Bureau of Land Management and other appropriate state and federal agencies. (Li 86)

MEASUREMENT, FINDINGS AND CONCLUSION

Monitoring stations which measure the background gamma radiation are established at a minimum of four locations around each mine site. Each station is about 500 meters from the center of the mine yard. Additional stations are located along the ore haulage routes. All information is being collected using identical detection methods and since the entire region has similar characteristics, any changes from existing background will be obvious.

Data will be collected quarterly using a suite of radiation detectors. Panasonic UD-802 passive thermoluminescent radiation dosimeters are placed at each site and measure the cumulative exposure. When dosimeters are exchanged, additional measurements are obtained using two Ludlum micro-R scintillometers. Annually, data is collected using a Reuter-Stokes RSS-111 Pressurized Ion Chamber. Background gamma radiation at all sites is on the order of 75 to 110 mR/yr.

During mine operation the radiation levels in the vicinity of the uranium ore stockpile will be on the order of 1 to 3 mR/hr. Levels decrease to background within a few hundred meters from the pile. It is anticipated that gamma radiation will remain unchanged at the monitoring stations during mine operation.

Radon gas will diffuse from the ore piles and be exhausted from the mine vent. Terradex track etch detectors are placed at most environmental stations and continue to measure normal background radon concentrations. Background radon gas concentrations are on the order of 0.2 pCi/L in the environs surrounding the Grand Canyon. The radon cups are exchanged quarterly. MILDOS code calculations indicate that increases will be minimal at distances in excess of about 2 km from the mine vent and the uranium ore piles. (St 85)

The breccia deposits are located about 500 meters above the water table so the mines are very dry. Nevertheless, a well is drilled at each mine site and samples are collected to obtain baseline data and to monitor for changes in trace element concentrations which might be the result of mining operations.

Average rainfall in the areas where the mines are located is generally less than about 38 cm per year. The amount of rainfall is about the same in the summer and winter, while the spring and fall are relatively dry. Summer precipitation usually is the result of thunderstorms which form over the heated canyon walls almost every afternoon from early July until the end of August. Since the thunderstorms may produce local sheet-flooding, each mine site is surrounded by a berm which is designed to protect against a potential 500 year flood event.

In order to assess the potential impact on the Colorado River from previous and present uranium mining activities and erosion from exposed deposits, routine radiological assays are performed on samples taken from the river and its tributaries as well as from several springs inside the canyon itself. Table I summarizes the significant results and reveals the definite increase in both uranium and Ra-226 in the Colorado River downstream of the confluence with the Little Colorado River. The radioactivity in the Little Colorado River is from exposed deposits which are not breccia pipe formations and from abandoned uranium concentrators located along the river. There is no impact on the waters of the Colorado River from current mining operations. A complete discussion of the water survey is published elsewhere. (St 87)

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Table I. Radionuclide Concentrations in the Colorado River and its Tributaries

mBq/L (+-2sigma)

Location	Date	U-238	U-235	U-234	Ra-226
1km upstream of Little Colorado	8/85	59 (15)	7 (4)	85 (22)	9 (5)
	7/86	67 (13)	12 (5)	140 (20)	13 (2)
Little Colorado (Cameron)	*7/86*	*890 (10)*	*33 (14)*	*1150 (150)*	*366(15)*
Little Colorado at confluence with Colorado	*9/86*	*560 (70)*	*48 (15)*	*460 (60)*	*67 (4)*
1km downstream of Little Colorado	8/85	85 (2)	6 (4)	110 (20)	36(10)
	9/86	96 (19)	<4	130 (20)	63 (4)
100km downstream of Little Colorado	8/85	74 (15)	2 (2)	96 (15)	35(10)
Kanab Creek	7/86	67 (15)	9 (6)	85 (22)	41 (6)

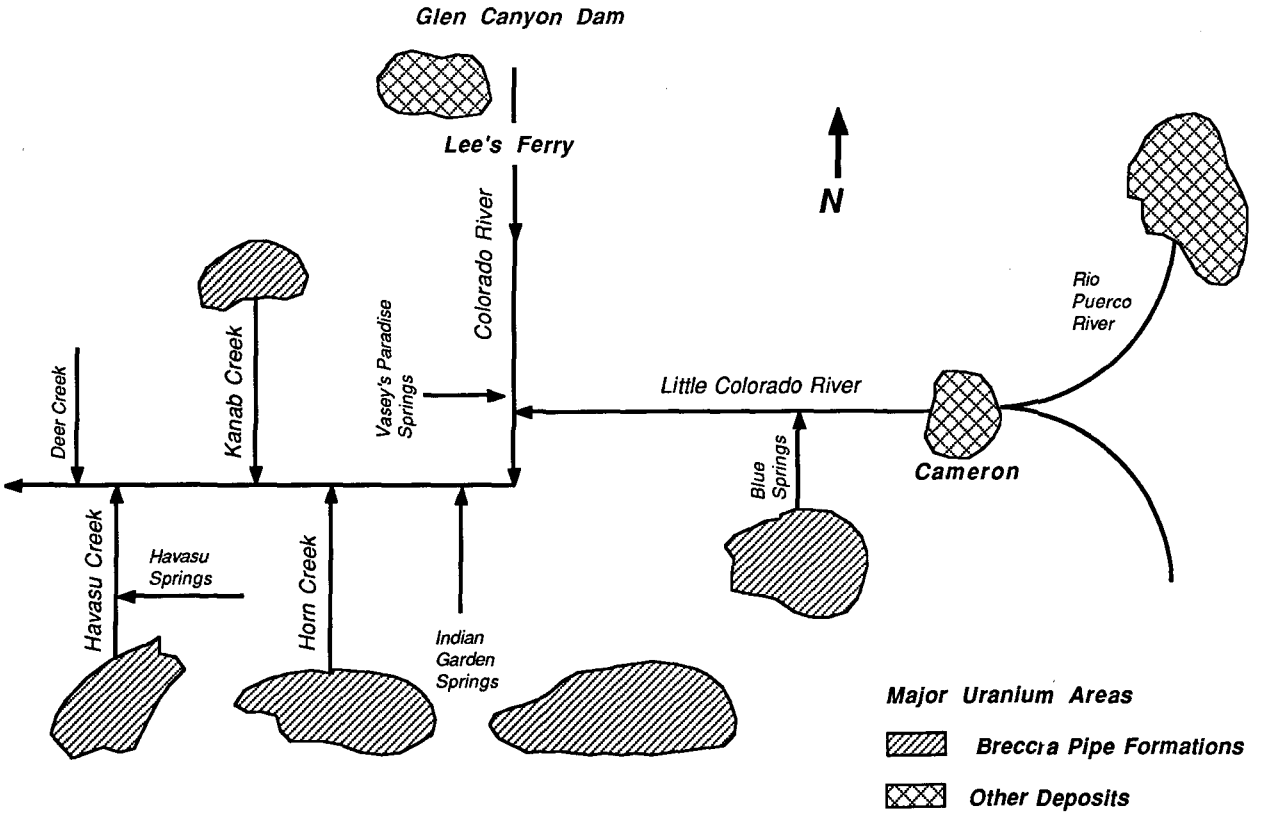

Figure 1 Uranium Mineralization Around the Grand Canyon

1.THE COMPUTER SYSTEM OF AUTOMATICAL MICROSCOPE ANALYSIS OF MINERS' INDIVIDUAL DOSIMETERS.*)

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The Institute of Occupational Medicine (IOM) carries on routine investigations on miners' individual exposure to radon and its alpha radioactive daughters in Polish mines [1]. Evaluation of miners' exposure is based on automatic analysis of track detectors by computer SYSTEM RADON. The IOM used detectors of size 2x3 cm cut from Kodak LT115 or LR115 dosimetry foil. The scheme of the system is presented in Fig.1 whereas Table 1 includes specification of its elements.

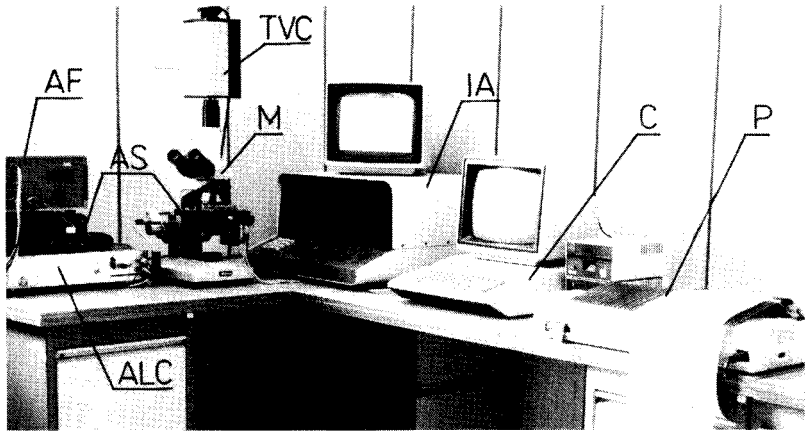

Fig.1. General scheme of SYSTEM RADON (abbreviations - see Table 1).

Table 1. Specification of main elements of SYSTEM RADON.

Abbreviations used in Fig.1 and text	Element	Producer
IA	Image Analyser	Analytical Measuring Systems Ltd England
AS	Auto Stepping Stage	
AF	Auto Focus	
C	APPLE IIe computer	Apple Computer Inc. USA
ALC	Auto Light Control	IOM Poland
M	Nikon Labophot Microscope	Nippon Kogaku K.K. Japan
P	Epson FX100 Printer	Epson Corporation Japan
TVC	TV Camera	Robert Bosch GMB Germany

*) Completed under project CPBR 11.5.65.2.

The detectors are subject to chemical etching [2] and next placed under the microscope M and viewed at magnification 20x through TVC. The image generated by TVC is analysed by IA with resolution of 280x384 piksels. The tracks on the detector's surface are counted by IA when the degree of 'greyness' is higher than the 'greyness' corresponding to the determined by the user detection level and when the surface area is within the determined limit of areas. Control range of detection level ranges from 0 to 99, whereas the limit of areas may be changed within the range from 0 to 99999 piksels. The analysed image is projected on IAM monitor. The automatic detector's stepping under the microscoppe M objective is controlled by AS and the picture sharpness is corrected by AF. The precision of detector's framing under the M objective amounts to 0.1mm. During the analysis, the intensity of light passing through the analysed field is measured by ALC. Hence, an information on the detector's thickness is obtained which is next used, after completing the analysis, to calculate the miner's exposure on the basis of the tracks density. The operations of the system's elements: IA, AS, AF, ALC are controlled by the measuring program introduced into the computer C. The program has been created by the Institute of Occupational Medicine in two languages: UCSD Pascal 1.2 and Assembler 6502. During detector's analysis the data from IA such as: the number of tracks and light intensity for particular analysed fields are passed into the computer's C memory. These data are not accepted by C when summary tracks area exceeds the limit of areas determined by the user and when light intensity passing through analysed foil exceeds the predetermined limit of intensities. To eliminate the influence of artefacts being an effect of e.g. detector's mechanical abrasions of size comparable to the tracks size on the determination of tracks density, a statistical verification of the collected data is performed. This verification is being made after completing the detector's read out. Due to verification all the fields on which tracks density differs from the mean tracks density, calculated for all analysed fields, by a defined by user number of standard deviations, are rejected. After completing statistical verification, the miner's exposure to radon and its progeny is calculated. These results, together with the basic data on the analysis process that has been carried out are printed on the printer P. The measurement program allows also to obtain the distribution of tracks density according to their diameters and to test the correctness of setting the principal parameters of the detector's analysis, such as detection level and intensity of light falling on the detector. The analyser IA used in SYSTEM RADON cannot distinguish the shapes of the analysed tracks, however, the presented above way of read-out causes that this option is not necessary for automatic reading-out the track detectors. The relative error does not exceed 30% , including the inherent error of the method (inaccuracy caused by the radon/daughters equilibrium). Duration of detector's analysis has been decrease in this system to 2 minutes. The results of the analyses are sent from computer C through IEEE 488 interface to IBM PC/XT computer and collected in the data base on hard disk. This data base compries the data on all miners covered by the radiological protection system implemented in Poland.

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2.METHOD OF CALIBRATION & ANALYSIS OF TRACK DETECTORS.*)

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Determination of track detector's exposure to radon and its daughters has been based on the experimentally chosen calibration coefficient K_{RnDP} correlating the measured surface density of tracks with the exposure to radon progeny. K_{RnDP} coefficient is defined as follows:

$$K_{RnDP} = \frac{G - G_T}{E_{RnDP}} \qquad (1)$$

where: G is the measured surface track density of detector after exposure expressed as [track*mm^{-2}], G_T is the measured surface track density of detector's background expressed as [track*mm^{-2}] , E_{RnDP} is the exposure to radon progeny of detector expressed as [mJ*h*m^{-3}].

The surface track density G depends, for a given exposure E_{RnDP} , on the state of equilibrium F [1] between radon and its daughters during the time of exposure, and also on the depth of detector's chemical etching after the exposure. The depth of detector's etching can be determined on the basis of the measurement of its thickness before and afer chemical etching. The system of automatical microscope analysis of track detectors enables to correlate the depth of detector's etching with the light intensity passing through the detector during the read-out of tracks, assuming that the intensity of light falling on the detector and the thickness of detectors before etching are constant. The former assumption is conditioned by the equipment used for the analysis of detectors, while the latter is true for detectors cut out of the detecting material produced in the same production cycle, e.g.: the material sold in rolls. It is also important here that the detecting material should be coloured by the producer since it determines the application of this method with regard to visible light. The red-coloured dosimetric foil LT115 produced by Kodak fulfills these conditions. It can be thus written for LT115 detectors that:

$$K_{RnDP} = f(F,I) \qquad (2)$$

In order to find the apparent functional form of K_{RnDP} coefficient, LT115 detectors have been exposed to radon in the experimental chamber [2] at various states of equilibrium F. Five measurement series, 30 measurements each, have been made. Each series has been exposed at different equilibrium F, for 8 hours and at radon concentration of 150 kBq*m^{-3}. Examinations have been made for F value in the range F=0.04-0.8. After the exposure detectors have been divided into 10 classes etched for various periods from 120 to 210 minutes. Various depths of etching have been thus obtained, as well as various light intensities I during the read-out for particular detectors. NaOH solution at the concentration of 6N and temperature of 55°C has been used for the etching of the detector while the reading has been made by means of SYSTEM RADON described in the first part of the paper. The detector's background depends on the depth of etching, so $G_T=f(I)$. For LT115 track detectors that dependence is as follows:

$$G_T = 0.0145*I - 0.884 \quad [track*mm^{-2}] \qquad (3)$$

This dependence was determined in the following procedure. The non-exposed detectors have been etched for various periods from 120 to 210 minutes and next the surface track density and light intensity was determined for these detectors. The linear regression has been applied for obtained data.

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*) Completed under project CPBR 11.5.65.2

K_{RnDP} value was calculated for each detector according to dependence (1). The regression curves made from obtained data are shown in Fig.1.

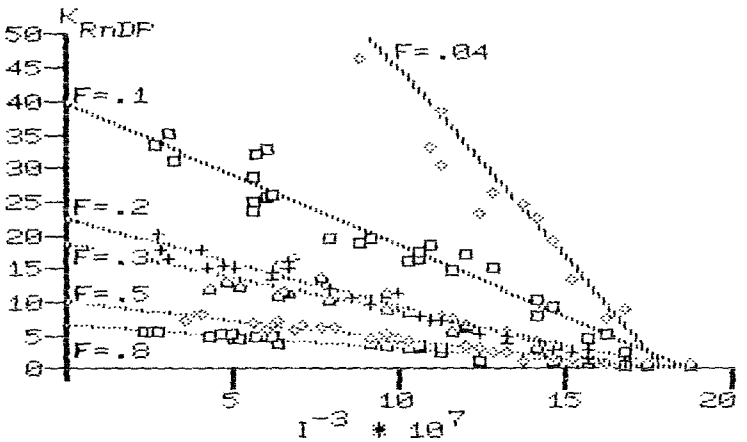

Fig.1. Diagram of the dependence of K_{RnDP} coefficient in the function of light intensity I for various states of equilibrium F.

On the basis of the computer analysis of experimental data the following dependence for K_{RnDP} has been obtained:

$$K_{RnDP} = (5.94 - 3.23*10^{6}*I^{-3}) * F^{-0.851} \qquad \left[\frac{track*mm^{-2}}{mJ*h*m^{-3}}\right] \qquad (4)$$

Exposure to radon daughter products, E_{RnDP} is calculated from correlation (5) resulting from equations (1),(2),(3). Thus:

$$E_{RnDP} = \frac{G - (0.0145*I - 0.884)}{(5.94 - 3.23*10^{6}*I^{-3})*F^{-0.851}} \qquad [mJ*h*m^{-3}] \qquad (5)$$

Exposure to radon E_{Rn} is correlated with exposure to radon daughters E_{RnDP} by equilibrium coefficient F :

$$F = \frac{178*E_{RnDP}}{E_{Rn}} \qquad (6)$$

where E_{RnDP} is measured in $mJ*h*m^{-3}$, while E_{Rn} in $kBq*h*m^{-3}$. Exposure to radon E_{Rn} can be calculated using correlation (6).

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THE IMPLICATIONS OF ICRU RECOMMENDED QUANTITIES FOR INDIVIDUAL MONITORING

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THE NEW SITUATION

In Report 39/1/, ICRU has recommended new quantities to be used in radiation protection measurements for exposures from external sources (cf also ref. /2/). These are

- for environmental monitoring
  - of strongly penetrating radiation, the ambient dose equivalent, $H^*(10)$,
  - of weakly penetrating radiation, the directional dose equivalent, $H'(0.07)$,
- for individual monitoring of exposure of the trunk of the body
  - to strongly penetrating radiation, the individual dose equivalent, penetrating, $H_p(10)$,
  - to weakly penetrating radiation, the individual equivalent superficial, $H(0.07)$.

Whereas the application of the quantities for environmental monitoring in the 30 cm diameter ICRU sphere does not pose serious conceptual problems and their implementation will probably be only a matter of time, the quantities for individual monitoring are the subject of lively international discussion. The Problems arise because $H_p(10)$ and $H_s(0.07)$ are defined

- at a specified position <u>within</u> the exposed individual,
- in the ICRU standard soft tissue, and <u>not</u> in the tissue actually existing at the specified position in the individual.

$H_p(10)$ and $H_s(0.07)$ therefore neither actually exist, nor are they measurable in the strict sense. However, besides these rather formal objections, there are other reasons for questioning the expedience of their choice. In particular, there is no generally valid unique relation of $H_p(10)$ [1] or a dose equivalent actually existing at the specified position of the individual to the quantities used in calibration, i. e. air kerma free-in-air (formerly exposure) for photon radiation and particle fluence for neutron radiation. The reason for this is that this relation depends on the geometrical peculiarities of the individual under consideration, the more so when strongly penetrating secondary radiations originating in the exposed person contribute substantially to the dose equivalent at the point of interest. Moreover, there will be no primary standards for dose equivalent in the forseeable future, i. e. for dose equivalent, irrespective of radiation type and quality.

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1) The following considerations will be restricted to $H_p(10)$, as $H_s(0.07)$ is less affected by variations in individual geometry due to the short ranges of weakly penetrating radiation.

RETROSPECT

Individual monitoring has been successfully performed for many decades. For photon radiation the procedure is as follows. Materials for dosemeter probes are selected to meet the requirement that their response be proportional to the air kerma (or exposure) independent of incident photon energy, if possible over the whole range of energies of interest, at least within admitted uncertainties. Alternatively, the readings of several probes with different energy responses are skillfully combined in order to achieve an energy-independent performance (e.g. photographic film with different filters). The expediently designed monitor (or a selected representative specimen of a bunch) is then calibrated free-in-air in units of exposure (or air kerma), thereby obtaining the calibration factor N (the quotient of air kerma K_a or exposure X and reading M).

After exposure of the monitor on the individual's body, the observed reading M is multiplied by the calibration factor N_K (or N_X), thereby obtaining the air kerma (or exposure) at the monitor's position on the body where it was exposed:

$$K_a = N_K M \quad \text{or} \quad X = N_X M$$

In order to obtain the individual's dose equivalent, K_a or X are multiplied by the conversion factor 1.15 Sv/Gy (for air kerma) or 0.01 Sv/R (for exposure). This result is routinely recorded as "personal dose":

$H_{ind} = 1.15\ K_a$ Sv/Gy or $H_{ind} = 0.01\ X$ Sv/R

H_{ind} is a good approximation of the dose equivalent at or near the surface of the exposed person and thereby in general, a safe estimate of the whole body dose or the effective dose equivalent, if the monitor's position at the body surface is representative of the person's radiation exposure.

For neutron radiation the procedure is different as the individual monitor is usually calibrated on a phantom approximating the trunk of the body (cf. ref. /3/), this requirement being a necessary consequence of the energy dependence of the dose equivalent response of most monitors, the varying Q factor being the chief reason.

THE NEW SITUATION

What then, is changed in practical procedures with the implementation of the new ICRU recommended quantity for individual monitoring, $H_p(10)$, for strongly penetrating radiation? Most important, there is now a uniform unique quantity for all kinds of strongly penetrating radiation by its definition in the ICRU sphere as an approximation of the human trunk. ICRU /1/ states,

> "The calibration of dosemeters is done under simplified conventional conditions at the depth d in an appropriate phantom. For dosemeters worn on the trunk a suitable phantom is the ICRU sphere."

From this statement, it can be conclusively inferred that the calibration is to be done in terms of the directional dose equivalent H'(10) defined at 10 mm depth in the ICRU sphere. This, then, is a prescription for the monitor's spectral and angular response. It should be made clear, however, that this requirement does not necessarily call for the calibration to be performed on the prescribed phantom. For photon radiation, the dose equivalent is numerically equal to the tissue-absorbed dose. A monitor's probe which is designed in such a way that its reading is proportional to tissue kerma at its location may therefore be calibrated by any procedure securing this. When the probe then, is embodied in an individual monitor simulating the geometric conditions of the H'(10) definition, it will sense the radiation field actually existing at its position and give the correct reading in terms of tissue kerma, which under the prevailing conditions is numerically equal to H'(10).

More frequently, however, the complete monitor (probe with casing) is calibrated, and especially in the case of neutron exposures, it is neither expedient nor even possible to design the probe's response to be proportional to the dose equivalent at its position, irrespective of the energy of the incident radiation. Instead, the probe's response is skillfully shaped to meet the required response over at least a limited range of energies when sensing the radiation field at the surface of the wearer's body, as for example with albedo monitors: then, of course, calibration on a phantom in terms of H'(10) is necessary. When using for the sake of convenience a phantom other than the ICRU sphere, any noticeable deviation from the required H'(10) response introduced thereby should be corrected.

Conversion factors relating H'(10) to the calibration quantities air kerma K_a or exposure X and neutron fluence as a function of incident energy and direction can be found in the literature /4,5,6/.

EXPERIENCE WITH EXISTING MONITORS

Tests with existing individual monitors for photon radiation demonstrate that the required spectral and angular responses can be met to a sufficient approximation. However, up to the present there has been no generally required and clearly defined quantity for individual monitoring, still less for its dependence on the direction of radiation incidence, except that resulting automatically and involuntarily when the monitor is exposed on the surface of the wearer's body.

As the accuracy requirements for individual monitoring are not very stringent (cf. /8/), there appears to be no serious obstacle to accepting the recommended H'(10) response for individual monitors, the required angular response to be verified for the front hemisphere at the most, or more realistically, to a cone with an angle of aperture of 60° (or less) around the direction of perpendicular frontal incidence onto the surface of the sphere.

EVALUATION OF MONITORS IN PRACTICE

As pointed out at the beginning, the ICRU recommended individual dose equivalent $H_p(10)$ neither actually exists nor is it directly measurable or limited by any regulation. Why then strive for its determination? It would be better to retain the well established and approved procedure of taking the product of the reading of a properly designed monitor and its calibration factor as the "individual dose equivalent for recording". The only material difference to the traditional procedure is the substitution of the multiplicity of reference quantities (air kerma or exposure for photons, tissue absorbed dose for beta radiation, maximum dose equivalent for neutrons) by a single one, namely the directional dose equivalent H'(10), (or H'(0.07) for weakly penetrating radiation). But we should not fail to acknowledge that this is important progress.

There is really no danger in following the procedure advocated here and omitting the explicit determination of $H_p(10)$, especially as doses in particular organs and tissues or the effective dose equivalent are the quantities to be determined or estimated in those cases where stated investigation levels of the "individual dose equivalent for recording" are exceeded. The proposed agreement would be a clarification and recognition of well-established procedures.

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RESULTS OF THE EUROPEAN COMMUNITY'S BETA INTERCOMPARISON PROGRAMME OF INDIVIDUAL DOSEMETERS IN 1986

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INTRODUCTION

The expanding use of radioactive sources in industry, medicine and research has led to a growth in the number of persons exposed to beta radiation and has highlighted the need to assess beta doses accurately. The difficulties experienced in the aftermath of the Three Mile Island accident owing to the inadequacies of beta dosimetry, had prompted a fresh look at the state of beta dosimetry within the European Community.

For over 20 years, the Commission of the European Communities (CEC), in collaboration with competent laboratories in the member states, has been conducting intercomparisons of individual dosemeters with the objective of improving technique for monitoring ionizing radiation and establishing a common basis for dose assessment within the Community. These programmes not only serve the participants with an opportunity to validate their calibration and measuring procedures but also help to create a forum in which to exchange information and discuss experience with other participants. The performance and results of such an intercomparison exercise conducted in 1986 are reported here.

Five European laboratories (Numbers 2 to 6 of author affiliations) performed the irradiations of the dosemeters *). Thirty-two laboratories, mainly drawn from the member states, took part in the exercise with 47 dosimetry systems, 31 were equipped with thermoluminescent detectors, 14 with films, 1 with a combination of a thermoluminescent detector and a film, and 1 with exoelectron emission detectors. The dosemeters were irradiated at different orientations with respect to the radiation field with beta rays from $^{90}Sr/^{90}Y$, ^{238}U, ^{204}Tl and ^{147}Pm sources and with gamma rays from a ^{137}Cs source. Neither the radiation quality nor the dose (nominally 5 mSv) was disclosed to the participants.

*) Under contract of CEC

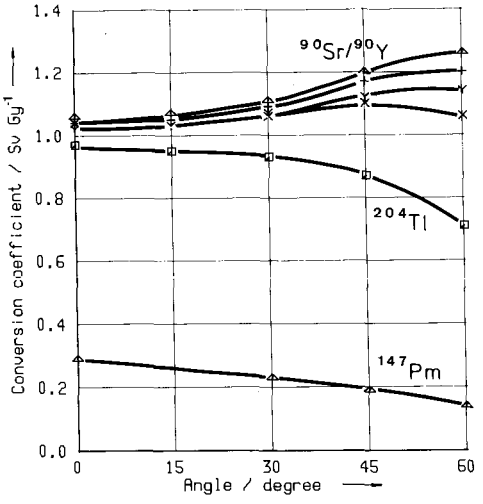


Figure 1: Conversion coefficient as a function of the angle of incidence for beta sources used during the intercomparison programme (four types of $^{90}Sr/^{90}$ sources, one ^{147}Pm and ^{204}Tl source type). The conversion coefficient is defined as the dose equivalent at a depth of 0.07 mm from the surface of a semi-infinite slab of tissue-equivalent material for a particular angle of incidence divided by the absorbed dose to tissue at the surface of a semi-infinite slab of tissue-equivalent material for an angle of incidence of 0^{o}. The angle of incidence is defined as zero if the axis of the beta particle field is perpendicular to the slab.

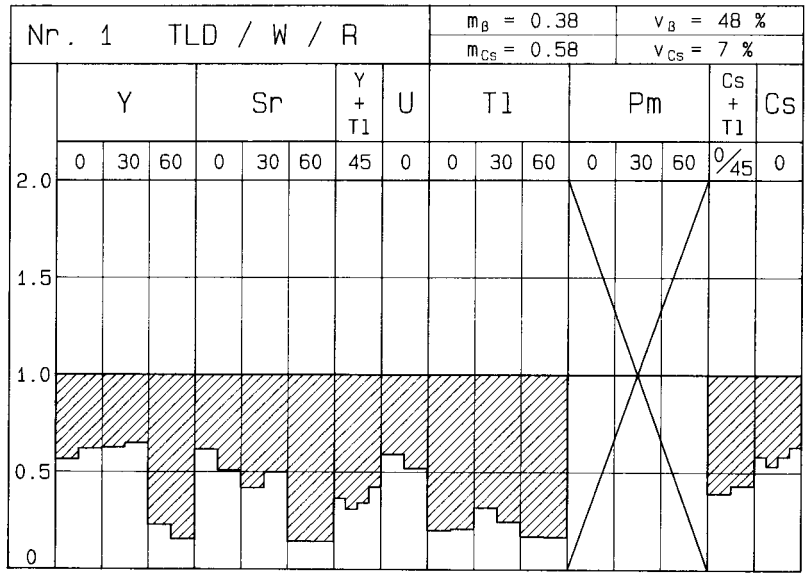
QUANTITIES AND IRRADIATION PROCEDURES

The quantity $H_s(0.07)$ recommended by the ICRU[1] was chosen as the operational quantity. In the absence of a universally accepted calibration quantity or irradiation procedure it was decided to expose the dosemeters to photon radiation positioned on a spherical perspex phantom of 30 cm diameter and to convert air kerma free in air measured at the location of interest, to $H'(0.07)$ by using coefficients derived for this purpose[2].

For beta radiation, however, it was judged that the use of a spherical phantom was not necessary because of the limited ranges of beta particles in dense media, or practicable because of the difficulty of achieving the expanded field condition over such a large phantom at the short distances at which these irradiations are carried out. Instead, the following phantoms were considered appropriate: a perspex slab of 20 cm sides and 1 cm thick for calibrating planar dosemeters and a perspex rod 20 cm long and with appropriate diameter for calibrating finger dosemeters. The calibration quantity in these cases is defined as the dose equivalent at a depth of 0.07 mm from the surface of semi-infinite slab of tissue-equivalent material. It became necessary to determine this calibration quantity for all the beta sources by additional measurements using extrapolation chambers. The conversion coefficients obtained from these measurements (figure 1) allowed the dose equivalent for various angles of incidence to be calculated from the absorbed dose to tissue at the surface of a semi-infinite slab of tissue-equivalent material for an angle of incidence of 0^{o} which is given in calibration certificates[3].

RESULTS

For every dosimetry system the results were shown in a diagram so that the overall performance, as well as the energy and angular dependence of the response, could be seen. The method of



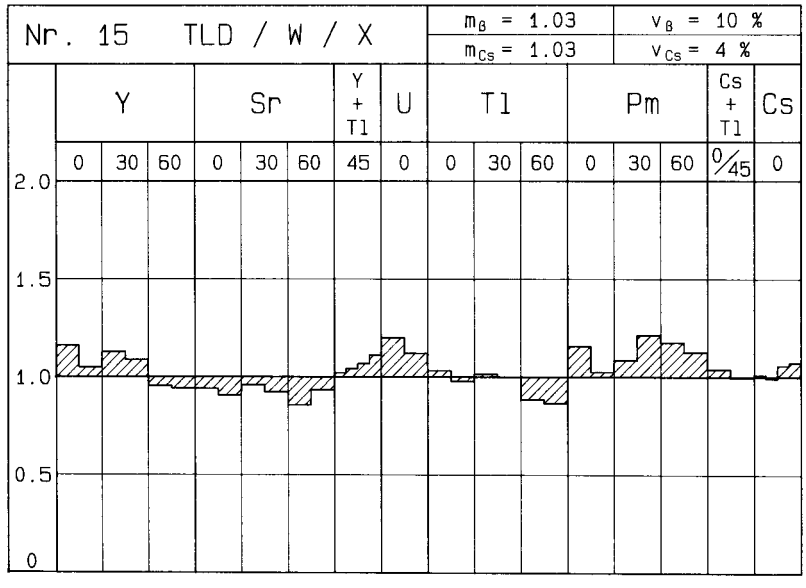


Figure 2: Two examples of results obtained for thermoluminescence dosimetry systems during the intercomparison programme.

presentation is illustrated in figure 2. The left-hand side of the top row contains the number and type of the dosimetry system. The result was calculated as a quotient of dose assessed by the participant by the dose to which the dosemeter was exposed by the irradiating laboratory (taken as conventional true value). For each dosemeter, the value of the quotient falling between 0 and 2 (first column) is represented by a horizontal bar under the appropriate radiation quality (second row) and the angle of incidence (third row). The area between this bar and the line representing quotient value equal to unity is shaded providing a visual indication as to whether the assessment by the dosimetry system is an overestimate or an underestimate. The letter "Y" in the second row indicates the use for irradiation of a $^{90}Sr/^{90}Y$ source with a relatively thick encapsulation whereas "Sr" indicates that of a source with thinner encapsulation, but of the same nuclide. On the right-hand side of the first row, mean values of the quotient for all the beta irradiations (m_β) and that for all the ^{137}Cs irradiations (m_{Cs}) together with corresponding coefficients of variation, v_β and v_{Cs}, are given.

The results of the dosimetry systems show substantial variability. System No. 15 showed the best overall performance for beta radiation (m_β = 1.03; v_β = 10 %). 35 of the 47 systems were unable to measure the low penetrating beta radiation from ^{147}Pm. Only 2 TLD systems using thin or effectively thin detectors were capable of assessing all doses of this intercomparison within $\pm$ 40 %. A considerable number of systems showed significant orientation dependence of response even for beta rays of higher energies (from ^{204}Tl and $^{90}Sr/^{90}Y$ sources). Therefore, even if the results obtained for the ^{147}Pm irradiations are excluded, for only 4 systems (3 TLD, 1 film) all the results lay within $\pm$ 40 %. This number of systems increases to 16 (9 TLD, 7 film) if $\overline{10}$ % outliers are regarded as being admissible.

A number of participants seemed to have problems when calibrating (at normal incidence) their dosemeters with ^{137}Cs gamma radiation and with $^{90}Sr/^{90}Y$ beta radiation. The m_{Cs} values ranged from 0.58 to 1.42, the corresponding values for $^{90}Sr/^{90}Y$ sources from 0.45 to 1.70. These ranges of values are not considered to be satisfactory. It is noted that some participants had problems with outliers, this being a strong argument for improving quality assurance.

More details of the intercomparison can be found elsewhere[4]. Participants found it a very useful exercise in improving the practice of radiation protection and felt that the intercomparison programme should continue. It was felt that there was a need for meetings on the topic of practical beta dosimetry, perhaps to be held on a regular basis.

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U.S. DEPARTMENT OF ENERGY LABORATORY ACCREDITATION PROGRAM FOR PERSONNEL DOSIMETRY SYSTEMS (DOELAP)

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The U.S. Department of Energy (DOE) Office of Nuclear Safety has developed and initiated the DOE Laboratory Accreditation Program (DOELAP) for personnel dosimetry systems to assure and improve the quality of personnel dosimetry at DOE and DOE contractor facilities. It consists of a performance evaluation program that measures current performance and an applied research program that evaluates and recommends additional or improved test and performance criteria. It also provides guidance to DOE, identifying areas where technological improvements are needed.

The two performance evaluation elements in the accreditation process are performance testing and onsite assessment by technical experts. Performance testing evaluates the participant's ability to accurately and reproducibly measure dose equivalent. Tests are conducted in accident level categories for low- and high-energy photons as well as protection level categories for low- and high-energy photons, beta particles, neutrons and mixtures of these. The choice of categories depends on the unique radiation protection needs at the participant's facility. Low-energy photon sources include 4 different X-ray bremsstrahlung spectra and 2 nearly-monoenergetic K-fluorescence X-ray spectra. A Cs-137 source is used for high-energy photons. Beta sources include Tl-204, Sr-90/Y-90 and a uranium slab. The neutron sources are bare and D_2O-moderated Cf-252. The performance test for each category requires the participant to evaluate and report the dose equivalent for 15 dosimeters irradiated at the performance testing laboratory. They are irradiated in 3 groups of 5 dosimeters over a period of about 4 months. The participant generally does not know what category a particular dosimeter belongs to.

The DOELAP performance test criteria are described in the Department of Energy Standard for Performance Testing of Personnel Dosimetry Systems (DOE/EH-0027, 1986). This DOE Standard is based on the American National Standards Institute's Criteria for Testing Personnel Dosimetry Performance, ANSI N13.11-1983, but it has been written with the additional dosimetry needs of DOE facilities in mind. These include multipurpose national laboratories, high-energy accelerators and facilities that process special nuclear materials. Additional beta, X-ray, and neutron sources and additional test categories have been added to better relate test and field conditions. The DOELAP performance criteria are similar

to those specified in ANSI N13.11 but they are generally more stringent.

After a participant successfully completes performance testing in the appropriate categories, an onsite assessment of the dosimetry system is conducted. Two assessors are assigned to visit the facility. They evaluate the dosimetry program to insure that it includes a quality assurance program; up-to-date documentation describing significant procedures and practices; a program for training and retraining staff members; adequate equipment and facilities; and appropriate procedures to insure the proper use, calibration and maintenance of that equipment. This process is described in the Handbook for the Department of Energy Laboratory Accreditation Program for Personnel Dosimetry Systems (DOE/EH-0026, 1986). At the end of the onsite visit, the assessors discuss program deficiencies with appropriate members of management. Accreditation can be granted for a period of 2 years after the deficiences are corrected.

A pilot DOELAP performance test session was conducted in 1985 and a report was published in 1986 (DOE/ID-12104). Six DOE facilities voluntarily participated. The results of this pilot study were used to finalize the performance test criteria and procedures. By the fall of 1987, four additional voluntary test sessions were completed. A total of 19 dosimetry systems were tested at least once (separate neutron and beta/photon dosimeters are listed as one system), and 7 passed the performance tests in all categories they required. Principal reasons for failure included incorrect calibration methods, inadequate dose calculation algorithms, inherent limitations in the dosimeter, lack of preparation and clerical errors. The performance test results for Sessions 1-5 are summarized in Table 1.

Table 1. DOELAP Performance Test Results Through Session 5

	Category	% Passed
I.	Low-Energy Photons, High Dose	57
II.	High-Energy Photons, High Dose	82
III.	Low-Energy Photons	26
IV.	High-Energy Photons	80
V.	Beta Particles	62
VI.	Neutrons	54
VII.	Mixtures	
	III + IV	85
	III + V	22
	IV + V	74
	III + V	78
	IV + VI	85

Clearly most dosimetry systems can pass the categories involving high-energy photons. The most difficult categories have been the the low-energy photon and low-energy photon + beta mixture categories. Many dosimetry systems have difficulties in accurately measuring low-energy photons. Energy dependence problems can be very significant over this range of photon energies. These tests are designed to identify such a problem by changing the X-ray spectrum for each group of 5 dosimeters. This is a more satisfactory approach than selecting only one spectrum.

To date, about one-half of the DOE dosimetry programs have participated in the voluntary DOELAP performance tests. As a result, many are making improvements that should enable them to successfully complete the performance testing when DOELAP accreditation becomes mandatory.

The DOELAP research program identifies potential improvements in DOELAP testing categories and provides guidance for implementing these in the Standard. Some of this work is conducted in response to DOELAP participants who feel that existing sources are an inadequate test of their dosimeter for particular field conditions. Other efforts focus on evaluating calibration methods, factors for assigning delivered dose or dose equivalent, and methods used in testing dosimeter response.

The response of a personnel neutron dosimeter depends on the energy distribution of the incident neutron field. Although the DOELAP test is not designed to evaluate the participants' field calibration, it is desirable to use a test source which is roughly appropriate for the neutron energies encountered in the workplace. At present, participants choose between an unmoderated Cf-252 fission spectrum and that of a Cf-252 source moderated by a sphere of heavy water 15 cm in radius and covered by a cadmium shell 0.051 cm in thickness. The addition of an Americum-Beryllium (Am-Be) neutron source is being evaluated at the request of accelerator facilities that have higher energy neutron fields. The Cf-252 spectrum is peaked near 2 MeV with a dose equivalent average energy of 2.4 MeV. An Am-Be source has a broad spectrum maximized between 1 and 11 MeV and a dose equivalent average energy of 4.4 MeV. While an Am-Be source is not an ideal representation of neutron sprectra at high energy accelerators, it provides a better test of dosimetry systems at those facilities than does Cf-252. An Am-Be source may be added to the DOE Standard as early as 1988.

Addition of a thermal neutron category was also considered. In a feasibility study, dosimeters from five DOE participants were irradiated to thermal neutrons from a sigma pile, to thermal neutrons from a research reactor, to Cf-252 neutrons, and to selected neutron-photon mixtures. The dosimeter response for the two thermal neutron sources differed by a factor of three, because of the source geometries and differences in the neutron energy distributions. The sigma pile presented an essentially isotropic field, while the reactor configuration was a collimated beam

irradiating dosimeters mounted on a phantom. After a review of the data and an informal survey of DOE facilities, it was concluded that the demand for a thermal neutron category is low because the thermal neutron dose contribution is small; most dosimeters respond adequately to thermal neutrons; the worst problem associated with thermal neutron dosimetry is a tendency for some dosimeters to falsely indicate a fast-neutron irradiation; and no masking effects were seen that would hide a significant photon or fast neutron irradiation in the presence of thermal neutrons. An additional thermal neutron category was therefore not recommended.

It has also been proposed that performance criteria should be added for dosimeter angular response. This proposal is now being evaluated. The DOELAP standard currently requires participants to measure and document the dosimeter response in two axes of rotation for angles between 0 and 85 degrees. However, the standard does not contain performance criteria for angular response. To add these, appropriate quantities must first be defined for assigning dose equivalent at nonperpendicular incidence. Then the existing recommendations for allowable dosimeter performance must be expanded to include uncertainties due to angular response. Existing models for assigning dose equivalent were reviewed. The directional dose equivalent, H', as defined in ICRU 39, was found to be an appropriate quantity for irradiations at nonperpendicular incidence, since it is additive for a multidirectional field. Performance criteria were developed based on recommendations in ICRP 35, which defines acceptable performance as accuracy within a factor of 1.5, including uncertainties from angular response. If recommended, the test of dosimeter angular response would be performed on a one-time basis, separated from the tests at perpendicular incidence.

There is considerable diversity among published conversion factors for determining dose equivalent for photon exposure or neutron fluence. Existing conversion factors for photons were reviewed in the hopes of alleviating some of the confusion caused by this diversity. Disagreement in the conversion factors is apparent, not only for the different torso models, but also for calculations using the same torso shape and tissue composition. These disagreements may be explained, in part, by the assumptions of parallel or divergent beams. The photon conversion factors used in DOELAP are based on a ten-element tissue slab model, and differ by up to 20% from the sphere calculations. These factors were verified by repeating extrapolation chamber measurements for k-fluorescence and filtered bremsstrahlung beams. Additional studies are planned for verification of conversion factors for the ICRU sphere, and a review of the impact of ICRU 39 is planned.

Other planned research includes evaluating appropriate performance criteria for CR-39 track etch dosimeters, considering the possible addition of a higher energy photon source and a low-energy beta slab source, and identifying standard methods for instrument calibrations.

DOSIMETRIE PERSONNELLE EN SUISSE
RETROSPECTIVE DES DIX DERNIERES ANNEES

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INTRODUCTION

La dosimétrie individuelle est asssurée en Suisse par des services privés et publics. Afin de coordonner ces activités, la Commission fédérale de la protection contre les radiations a constitué dès 1976 un Groupe d'experts pour la dosimétrie individuelle comprenant des spécialistes appartenant aux différentes instances de contrôle, ainsi qu'aux services dosimétriques. Les tâches de ce groupe sont les suivantes :

- réunion, interprétation et publication des valeurs de doses des personnes exposées professionnellement aux radiations ionisantes;
- élaboration de critères en vue de la reconnaissance des services de dosimétrie;
- élaboration des documents de base et de recommandations pour l'exécution de la dosimétrie de l'irradiation externe et de l'incorporation;
- contrôle de la qualité des résultats de mesure des services dosimétriques à l'aide d'intercomparaisons;
- analyse de l'état de la science et de la technique dans le domaine de la dosimétrie individuelle, activité de consultation pour les instances de contrôle.

Les activités de ce groupe et les résultats obtenus au cours des dix années écoulées font l'objet du présent rapport.

RESULTATS DE LA DOSIMETRIE INDIVIDUELLE

L'ensemble des résultats de la surveillance dosimétrique individuelle est publié chaque année sous forme de rapport (1). Les données rassemblées concernent aussi bien les mesures de l'irradiation externe, dosimétrie du corps entier et des extrémités, que celles touchant à l'incorporation. Ce rapport comprend, outre une ventilation des personnes surveillées en fonction de leur activité et de la dose reçue, une indication des doses collectives dans les différentes catégories professionnelles. L'évolution de ces grandeurs en fonction du temps est donnée à la figure 1. Le nombre de personnes surveillées augmente régulièrement depuis 1976; cet accroissement correspond d'une part à une augmentation réelle du collectif et d'autre part à une surveillance plus étroite des personnes professionnellement exposées aux radiations, en particulier dans le domaine médical. On remarque parallèlement une réduction de la dose collective dans la majeure partie des secteurs d'activité. En ce qui concerne les centrales nucléaires, l'augmentation observée est liée à la mise en service de deux nouvelles centrales et à divers travaux de rééquipement effectués dans les centrales plus anciennes.

Figure 1: Evolution des doses collectives et du nombre de personnes surveillées de 1977 à 1986

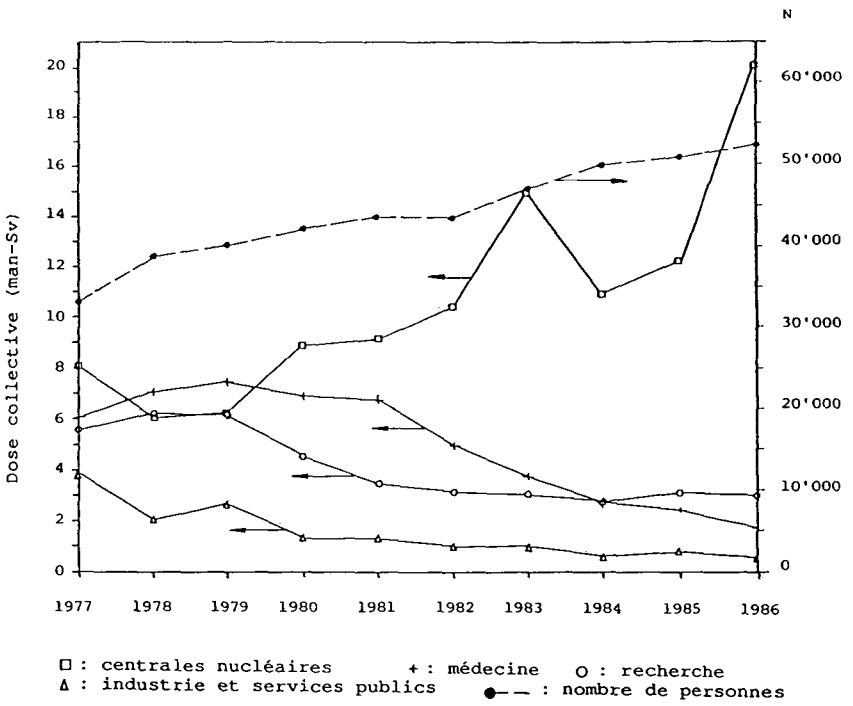
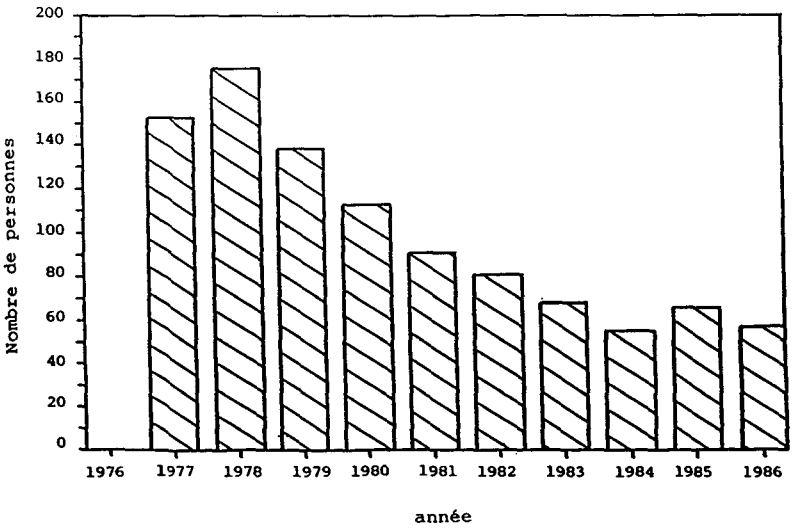


Figure 2: Nombre de personnes ayant incorporé une quantité supérieure à 10 % de la limite annuelle



Les incorporations impliquent un nombre de personnes et des doses annuelles relativement limités. A titre d'illustration, le nombre de personnes ayant incorporé une quantité supérieure à 10% de la limite annuelle est donné à la figure 2 pour les dix dernières années. Dans la plupart des cas, il s'agit d'incorporation de tritium, radioélément utilisé pour la fabrication de peintures et matériaux luminescents.

Tous les dépassements des limites annuelles (irradiation externe et incorporation) font l'objet d'un examen particulier, qui est documenté dans le rapport annuel. Le nombre annuel de dépassements ayant conduit à une irradiation réelle de la personne oscille entre 0 et 5. Dans un seul cas, au cours de ces dix dernières années, un dommage a été constaté; il s'agissait d'une brûlure à une main.

ASPECTS LEGISLATIFS ET RECOMMANDATIONS

Une ordonnance sur l'homologation et l'exploitation des services de dosimétrie individuelle, mise en vigueur en 1981, fixe les exigences applicables aux services de dosimétrie (2). Celles-ci concernent aussi bien les aspects administratifs que techniques. Ainsi les services dosimétriques ont l'obligation de mesurer les doses d'irradiation situées entre 500µSv et 5 Sv avec une précision située entre plus 50% et moins 30% en routine. En outre, les performances minimales des divers systèmes dosimétriques sont prescrites; les exigences dans le cas de détecteurs basés sur la thermoluminescence sont données au tableau 1 à titre d'exemple.

Tableau 1: Exigences minimales applicables au système de dosimétrie individuelle par thermoluminescence

Caractéristiques	Exigences	
	photons	électrons
Dose minimale détectable	0,5 mSv	2,5 mSv
Dose maximale détectable	5 Sv	5 Sv
Domaine d'énergie	15 - 3000 keV	0,5 MeV
Dépendance énergétique	+/- 40 %	- 50%;+100%
Dépendance directionnelle	+/- 40 %	-
Reproductibilité	+/- 20 %	+/- 20 %
Erreurs diverses	+/- 10 %	+/- 10 %

Le groupe d'experts a en outre émis des recommandations touchant à des aspects particuliers de la surveillance dosimétrique. Ainsi dans le cas du contrôle d'incorporation, la fréquence, le mode des contrôles et les méthodes d'interprétation ont fait l'objet d'une étude. Pour les nucléides les plus couramment utilisés une procédure de contrôle a été fixée (3).

Dans le domaine de la dosimétrie de l'irradiation externe, la méthodologie métrologique et les procédures de calibration ont été précisées, dans le but d'une uniformisation de la dosimétrie (4).

INTERCOMPARAISONS DOSIMETRIQUES

Le groupe d'experts organise chaque année une intercomparaison dosimétrique dans le domaine de l'irradiation externe; les performances des systèmes y sont testées quant aux critères fixés par l'ordonnance. Les aspects suivants ont été analysés :

- contrôle de la calibration;
- linéarité des dosimètres dans le domaine de dose correspondant aux mesures en routine;
- dépendance en fonction de l'énergie des rayonnements X;
- linéarité dans le domaine des doses accidentelles; vérification du temps nécessaire à l'évaluation du dosimètre en cas d'accident;
- précision des valeurs de mesure lors d'une irradiation avec plusieurs types de rayonnements.

Les résultats obtenus sont indiqués dans les rapports annuels correspondants. Ces intercomparaisons ont contribué à améliorer la qualité de la dosimétrie individuelle en Suisse.

Deux intercomparaisons ont en outre été effectuées pour les services chargés de la suveillance de l'incorporation de tritium; les résultats obtenus ont indiqué qu'un effort complémentaire doit être consenti dans ce domaine.

CONCLUSIONS

Sur la base des résultats de la dosimétrie individuelle, l'évolution de la radioprotection en Suisse au cours de ces dix dernières années peut être considérée comme réjouissante.

Bien que le nombre des personnes professionnellement exposées aux radiations ait augmenté de manière continue, la dose collective indique une tendance à la baisse, à l'exception du domaine des centrales nucléaires. Ceci est particulièrement significatif, car le nombre d'installations produisant des rayonnements ionisants a sensiblement augmenté aux cours de ces dix années.

Ce succès n'aurait certainement pas été obtenu, si l'on n'avait pas attaché l'importance qu'elle mérite à la dosimétrie des personnes professionnellement exposées aux radiations. A ce jour, les instances de contrôle tiennent la surveillance bien en main.

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THE ADEQUACY OF CURRENT METHODS OF SKIN DOSE ASSESSMENT

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ABSTRACT

Irradiation of the skin is often a limiting factor in routine and accidental radiation exposures. In most situations the exposure is spatially non-uniform and in the case of subjection to radioactive participation it may be highly non-uniform. In these cases the choice of a depth and area over which to measure or calculate the skin dose is open to some controversy and is currently the topic of consideration by the NCRP and ICRP. A range of animal experiments have previously been described which shed light on these questions and this paper will present detailed dosimetry for some of the radiation sources used. A novel automated extrapolation ionisation chamber will be described which has enabled detailed isodose data to be generated for uniform and non-uniform Tm-170 beta sources which have been used to study the 'hot particle effect'. Comparisons with thermoluminescence dosimetry data and calculations will be given. The radiobiological data show no evidence of any enhanced carcinogenic response for non-uniform exposures and support the use of mean dose as a relevant parameter for evaluating the risk of late stochastic effects. Data on non-stochastic effects indicate that current procedures for estimating skin dose are likely to considerably over-estimate the risk of detrimental effects. The adequacy of currently employed skin dosimeters will be discussed in the light of these biological and dosimetric data.

STUDY OF DOSEMETER PARAMETERS FOR THE MEASUREMENT OF $H_s(0.07)$ FOR PERSONAL BETA DOSIMETRY

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INTRODUCTION

The basic requirement of a dosemeter designed for individual monitoring of beta radiation is the capability of measuring the skin dose (in a few cases also the dose to the lens of the eyes) independently of energy and incident angle of the radiation and with the accuracy prescribed by national and international bodies. Other important design criteria are concerned with cost and such practical aspects as dosemeter size, identification possibilities and fitness for automated processing. According to ICRU the skin dose should be measured at a depth of 0.07 mm below the surface of the skin and a dosemeter with an energy and angular response equal to that of $H'(0.07)$ is considered to satisfy this requirement[1]. In nearly all practical situations a dosemeter intended for personal beta dosimetry must also be able to measure the dose contribution to $H_s(0.07)$ from strongly penetrating gamma radiation.

The performance of a dosemeter is determined from a combination of detector characteristics and badge design. If a tissue-equivalent detector is used, $H_s(0.07)$ can be measured with a simple dosemeter design. Where non-tissue-equivalent detectors are used it is necessary to evaluate beta ray and low energy photon doses separately due to the energy dependence of these detectors to low-energy photon radiation, implying the requirement of using different thin filters. The most common non-tissue-equivalent detector type, the film, is wrapped in a light-tight packet, typically about 0.25 mm thick, permitting this dosemeter to be used only for monitoring beta rays with energies above approximately 0.5 MeV.

A thin tissue-equivalent detector of a few $mgcm^{-2}$ and covered with a similar tissue-equivalent filter is an appropriate dosemeter for skin dose measurements. During the last few years different types of tissue-equivalent detectors with a thin effective detector thickness have been developed[2] and this study in addition comprises a new graphite-mixed sintered LiF detector recently developed by the Alnor Oy. For a dosemeter intended for automatic readout it may be difficult to obtain a position of the skin dose detector in the badge exhibiting an optimal dosemeter performance.

In this investigation the importance of different dosemeter parameters, e.g. detector thickness, filter thickness, and detector/filter geometry and material, for obtaining an optimal energy and angular dosemeter response for beta dose measurements has been studied.

EXPERIMENTAL METHODS

Five types of TL detectors were included in this study, namely LiF, $Li_2B_4O_7$:Mn, and LiF with graphite produced by the Alnor Company, LiF TLD-700 produced by the Harshaw Chemical Company and MgB_4O_7:Dy, with graphite produced by the Boris Kidrić Institute, Vinča[3]. Dimensions and dosimetric properties for irradiation with ^{60}Co photons are presented in Table 1. The detectors were read-out in a hot nitrogen stream for 10 s at 270°C. As pre-irradiation annealing was used a second read-out in the reader and all detectors were given a post-irradiation annealing of 15 min. at 100°C.

Table 1. Dimensions and dosimetric characteristics of different types of TL detectors.

Detector	Dimensions (mm)	Relative γ-ray(^{60}Co) sensitivity	Detection threshold (3σ) (eqv. μGy ^{60}Co)
LiF-N, Alnor	4.5(diam.)x0.8	100	8
LiF TLD-700, Harshaw	4.5(diam.)x0.9	86	8
$Li_2B_4O_7$:Mn, Alnor	4.5(diam.)x0.85	21	20
LiF-N, (2% C), Alnor	4.5(diam.)x0.7	9.5	110
LiF-N, (4% C), Alnor	"	4.5	110
LiF-N, (8% C), Alnor	"	2.0	200
MgB_4O_7:Dy(3%C),Vinča	4 (diam.)x0.9	4.5	120

Beta-ray doses from ^{90}Sr-^{90}Y (E_{max} = 2.27 MeV), ^{204}Tl (E_{max} = 0.76 MeV) and ^{147}Pm (E_{max} = 0.225 MeV) were obtained from the PTB-Büchler secondary standard beta calibration unit. Furthermore, two stronger ^{204}Tl- and ^{147}Pm-sources were used: a 42-mm diameter ^{204}Tl source covered with 20 mgcm^{-2} thick silver foil and a 5-mm diameter ^{147}Pm source with a 5-μm titanium face layer. The distances used between source and detector were 30 cm for the ^{204}Tl and ^{90}Sr-^{90}Y sources and 20 cm for the ^{147}Pm source. As calibration quantity for the measurement of $H_s(0.07)$ for beta radiation was used, the dose equivalent at a depth of 0.07 mm in a semi-infinite extended slab of tissue-equivalent material. Values of this quantity for the standard beta sources were obtained from the calibration certificates for normal radiation incidence and from Böhm[4] for other angles of radiation incidence. The values for the two non-standard sources were obtained from measurements with an extrapolation chamber designed by Böhm[5] and purchased from PTW, Freiburg, FRG. A 1-cm thick perspex plate that can be turned to obtain different irradiation angles was used as phantom for the irradiations. ^{60}Co gamma radiation was used as reference radiation with the detectors placed between two 4-mm thick perspex plates during the irradiations.

RESULTS AND DISCUSSION

Detector/filter thickness

The energy and angular response of a dosemeter to beta radiation strongly depends on the thickness of the detector and/or filter as illustrated in Table 2 where responses of dosemeters with different detector and filter thicknesses are given for different angles of radiation incidence and beta particle energies. The standard ^{90}Sr-^{90}Y source and the two non-standard ^{204}Tl and ^{147}Pm sources were used for these measurements. It can be seen from the table that if appropriate, thin filters are used the graphite-mixed detectors, all having a thin effective detector thickness, are well-suited to the measurement of $H_s(0.07)$ for beta radiation with energies above approximately 220 keV. In contrast, the use of a dosemeter with a relatively thick detector, e.g. the normal 0.9 mm thick LiF detector, for beta dosimetry may imply a significant underestimation of the dose, even for a beta dose from a high-energy beta emitter like the ^{90}Sr-^{90}Y source.

Table 2. Beta ray response of TL detectors placed on a perspex phantom. Results are given for measurements with and without the use of a 7 $mgcm^{-2}$ tissue-equivalent filter and for 0° and 45° angle of radiation incidence.

Detector type	Filter ($mgcm^{-2}$)	TL reading per unit beta dose at 0.07 mm tissue and at angle β° / TL reading per R ^{60}Co					
		^{90}Sr-^{90}Y		^{204}Tl		^{147}Pm	
		0°	45°	0°	45°	0°	45°
LiF-N	0	1.05	0.84	0.49	0.38	0.12	0.13
Alnor	7	1.02	0.80	0.42	0.32	0.03	0.03
LiF, TLD-700	0	1.00	0.80	0.29	0.22	0.12	0.13
Harshaw	7	1.00	0.80	0.27	0.22	0.03	0.03
$Li_2B_4O_7$:Mn	0	1.06	0.89	0.62	0.54	0.21	0.27
Alnor	7	1.03	0.83	0.58	0.47	0.04	0.04
LiF-N (2%C)	0	1.01	0.92	0.93	0.96	0.84	1.02
Alnor	7	1.02	0.95	0.87	0.85	0.16	0.16
LiF-N (4%C)	0	1.02	0.94	1.02	1.00	1.10	1.37
Alnor	7	1.02	0.96	0.93	0.90	0.21	0.24
LiF-N (8%C)	0	0.98	0.96	1.02	1.13	1.50	1.94
Alnor	7	1.03	0.96	0.96	0.96	0.30	0.29
MgB_4O_7:Dy(3%C)	0	0.95	0.94	1.05	1.14	2.90	3.80
Vinca	7	1.02	0.98	0.96	0.90	0.70	0.73

Detector/filter geometry

In most personal dosemeters the detector is positioned at a certain distance from the filter beneath a so-called beta window that has a significant influence on the energy and angular response of the dosemeter to beta radiation[2]. In the Alnor and Risø badge the skin dose detector is placed behind holes in 1 mm thick plastic and aluminum shields, respectively[6]. Beta-ray

responses of the two types of badges were measured by use of graphite-mixed LiF and MgB_4O_7:Dy detectors and by use of both cylindrical- and cone-shaped beta windows (see Table 3). By comparing the data of Table 3 with those of Table 2 it can be seen that to obtain an optimal energy and angular response of the dosemeter a detector/filter geometry as close as possible to that of a simple dosemeter design with the detector positioned in contact with the filter and with no disturbing beta windows should be aimed at. The results furthermore show that improvements can be obtained by optimising the shape and size of the beta window as well as the type of construction material.

Table 3. Beta ray response of TL detectors placed in the beta window position of the Alnor and Risø badge. Results are given for a 4.5 $mgcm^{-2}$ tissue-equivalent filter, for 0° and 45° angle of radiation incidence, and for different beta window shapes (see text).

Detector type	Badge type	Diam.(mm) and shape of beta window	TL reading per unit beta dose at 0.07 mm tissue and at angle β° / TL reading per R ^{60}Co					
			^{90}Sr-^{90}Y		^{204}Tl		^{147}Pm	
			0°	45°	0°	45°	0°	45°
LiF-N (4% C) Alnor	Alnor	3.5 Cyl.	1.08	0.66	0.50	0.34	0.14	0.16
		3.5 Con.	1.13	0.68	0.55	0.44	0.16	0.17
	Risø	3.5 Cyl.	0.83	0.48	0.44	0.32	0.15	0.15
		4.0 Con.	1.07	0.65	0.68	0.53	0.20	0.20
MgB_4O_7:Dy (3% C) Vinča	Alnor	3.5 Cyl.	1.13	0.74	0.72	0.60	0.51	0.50
		3.5 Con.	1.15	0.81	0.81	0.69	0.62	0.66
	Risø	3.5 Cyl.	0.85	0.50	0.60	0.47	0.48	0.48
		4.0 Con.	1.14	0.74	0.81	0.85	0.70	0.70

CONCLUSIONS

The results obtained from this study emphasise the importance of using thin detectors for skin dose assessment. Furthermore a badge design with a minimal shadow effect caused by the beta window should be aimed at.

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THE FADING OF LiF CHIPS IRRADIATED BY UV-PHOTONS

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1. INTRODUCTION

The thermoluminescent dosimeter is becoming increasingly important in all aspects of personnel and environmental monitoring. The main limiting factor that has prevented even wider usage is the inability to allow reassessment of radiation doses after the initial read-out.

The most useful dosimeter, especially in personnel monitoring, is the LiF:Mg,Ti (TLD-100). The energy, absorbed from ionizing radiation, produces mobile charge-carriers, electrons or holes, in dopped LiF. Some of the charge carriers are trapped by impurity complexes, and are released from the traps by heating the irradiated LiF. Its normal read-out is a 10-60 seconds period of heating to 250-300 °C, which is sufficient to remove the electrons (or holes) from all but the deepest traps. The thermally released carriers migrate to luminescent sites, where they recombine with (other) holes (or electrons). Light is released as this recombination occurs. The signal is recorded as a glow curve of thermoluminescence versus time or temperature.

The high, first dose can be reassessed by a second reading (residual dose) or by evaluating the dosimeter after exposing it to UV photons. In a previous work[1], we have studied the re-assessment of radiation doses measured by TLD; we have found that the dose obtained by the latter technique is greater than the residual dose, by one order of magnitude. It is known as 'UV light bleaching' or 'photo-transfered thermoluminescent' dosimetry or PTTL. It is well used for reassessing high doses in personnel and environmental dosimetry[2,3].

One of the inaccuracies of thermoluminescent dosimeters is the fading of the dose between the exposure time and the evaluation. If no annealings are performed, it can be as high as 20-30% per month for LiF chips[4,5]. The fading of LiF (TLD-100) can be decreased by different pre- and post-irradiation annealings: for a pre-irradiation annealing of 400 °C/hr or a post-irradiation annealing of 100 °C/10 min, the fading of TLD-100 is less than 4% per month[6,7].

In this work, the fading of PTTL in LiF:Mg,Ti chips during a 31-day period, was measured.

2. Experimental and results.

The measurements were performed with the standard (Harshaw) TLD-100 chips (3x3x0.9 mm^3) which were all annealed for an hour at 400 °C, prior to irradiation. The samples were read-out by a manual Harshaw thermoluminescent detector, model 2000C, connected to an automatic integrating picoammeter, model 2000B and a Yokogawa recorder for glow curves.

Each chip was irradiated with a ^{60}Co source up to a dose of 80 mGy and evaluated after different periods of time, from 1 to 31 days. The dosimeters were read-out in an atmosphere of dry nitrogen to minimize the effect of tribo-thermoluminescence. The heating was about 12 °C/sec and a maximum temperature of 280 °C was reached in each cycle. Immediately after the first reading, the chips were irradiated by a 15 watt UV lamp (254 nm) at an 11 cm distance and then a second reading was performed (PTTL). In a previous work[3], we have concluded that the optimal irradiation-time for TLD-100 chips by a UV lamp is 15 minutes.

The 'second' readings, which were performed after the UV-PTTL-irradiation at different times following the ^{60}Co-irradiation, were compared to those readings that were evaluated immediately. The fading of PTTL was thus calculated for different periods of time, from 1 to 31 days. The results are given in Table 1.

Table 1. The fading of PTTL.

Time (days)	Fading (in %)
1	2.6 ± 0.1
4	2.7 ± 1.5
8	2.5 ± 1.3
15	2.8 ± 0.9
22	4.7 ± 1.2
31	4.7 ± 2.6

The glow curves of PTTL after 1, 4, 15 and 31 days are shown in figure 1.

3. Conclusions

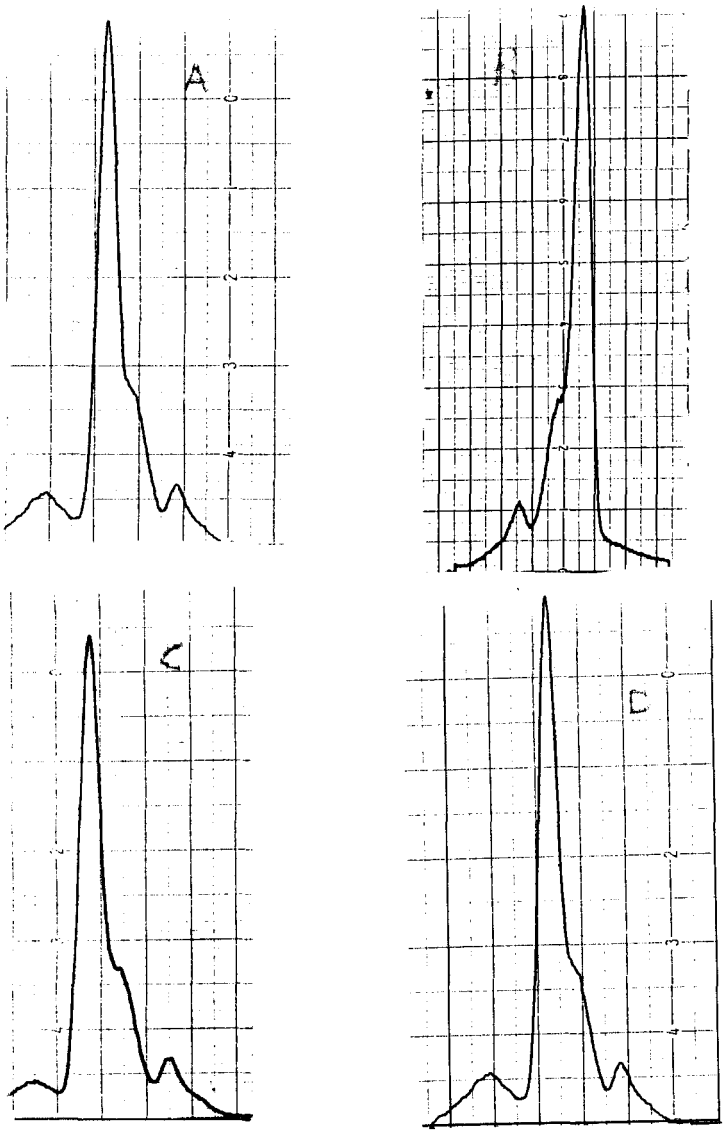
A. The fading of PTTL for TLD-100 chips is changing from 2.6% after 24 hours to 4.7% after 31 days.

B. The relatively low fading is to be expected as the second reading is due to electrons in the deep (high temperature) traps.

C. The glow curves of PTTL exhibite the same glow peaks as the curves from the γ-irradiated TLD-100; the main peaks are 4 and 5, while the lower ones are 2 and 7.

D. There is no significant difference between the glow curves of PTTL after 1 day or 31 days (see figures 1a - 1d).

Fig. 1 - The glow curve of PTTL:
a) after 1 day, b) after 4 d, c) after 15 d, d) after 31 d.



E. High doses in personnel and environmetal monitoring can be reassessed accurately, without the need for any significant fading corrections.

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DOSEMETER FOR MEASURING PARTIAL BODY DOSE AND FOR ADDITIONALLY DETERMINING THE TYPE OF RADIATION

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While handling unshielded, radioactive substances the radiation dose to the nand frequently has to be measured. Close to the radiation source the greatest fraction of the dose may originate from weakly penetrating ß-radiation. However, part of the dose also originates from strongly penetrating radiation. In general, the skin is the critical organ and a dosemeter which can measure the individual dose equivalent, superficial, is sufficient to fulfil the legal regulations. However, information about the type of radiation is additionally required in order to evaluate the working conditions and radiation hazards. This is particularly important in evaluating possible radiation injuries after an accident. If the radiation type is known then the dose can be more accurately determined from the dosemeter reading with the aid of appropriate correction factors.

PREVIOUS EXTREMITY DOSEMETER

A few years ago, an extremity dosemeter was developed at the Nuclear Research Centre Jülich (KFA) enabling a precise determination of the skin dose to be made (1). This dosemeter also provides additional information about the radiation quality. The dosemeter contains three thin TLD's of LiF in Teflon arranged in series and covered with a very thin, aluminized plastic foil of 0.9 mg/cm² mass per area. The first TLD measures the directional dose equivalent H'(0.07) approximately independent of energy. The second and third TLD's in the dosemeter provide information about the depth dose distribution and thus about the radiation quality.

The dosemeter is excellently suited for the determination of H'(0.07). This was demonstrated during participation in the 1986 international intercomparison programme of the Commission of the European Communities. In this intercomparison programme 36 dosemeters from each participant were irradiated with ß-radiation of various energies from E_{max} = 0.23 MeV to E_{max} = 2.2 MeV, with Cs-137 Y-radiation and with ß-Y mixed radiation. Irrespective of the radiation type, radiation energy and direction of incidence, the results of the KFA extremity dosemeter were in agreement with the reference values within ±23 % for all 36 dosemeters (2).

However, a disadvantage of this dosemeter is that the very thin window in front of the TLD is not sturdy enough and is occasionally destroyed.

MODIFIED EXTREMITY DOSEMETER

Since the previous dosemeter was not sturdy enough attempts were made to develop a more suitable extremity dosemeter. The new dosemeter should be more sturdy and moreover provide even more information about the radiation field. The new dosemeter

should also be able to differentiate between high-energy and low-energy Y-radiation (3).

In the modified extremity dosemeter (Fig. 1), three TLD's are arranged one behind the other in a finger-ring and the three TLD's are enclosed by a heat-shrinkable sleeve. Two ultra-thin TLD's of $CaSO_4$ in Teflon are used, between which a 0.4 mm thick TLD of LiF in Teflon is arranged. The thinnest opaque heat-shrinkable foil on the market is used with an unshrunk thickness of 0.05 mm.

Different radiation fields can be distinguished using the readings of the three TLD's in the dosemeter:

In the case of high-energy ß-radiation and Y-radiation, all three TLD's show the same value which is equivalent to the value of the skin dose.

For low-energy ß-radiation, the first, ultra-thin TLD gives a high value. The other TLD's show considerably lower values.

For low-energy Y-radiation, the first and third TLD's show the same, too high values. The dose is determined from the reading of the middle TLD. The presence of low-energy Y-radiation and thus the oversensitivity of the $CaSO_4$ detectors may be recognized by the fact that the first and third detectors show the same reading and that in addition both readings are higher than that of the middle TLD. The energy of the Y-radiation can be assessed from the ratio of the reading of the first TLD to the reading of the second TLD.

The response R of the first TLD and the ratio of the readings of the three TLD's are given for ß-radiation in Table 1. The response is too low for low-energy ß-radiation. It can be seen from the ratios $D_1:D_2$, the reading of the first TLD to the reading of the second TLD, and $D_1:D_3$, the reading of the first TLD to the reading of the third TLD, that the dosemeter was irradiated by low-energy ß-radiation. It is then possible to correct the too low response of the dosemeter.

Fig. 2 shows the response for Y-radiation.

The lower detection limit is equal to 0.3 mSv for the reading of the first and third TLD's and equal to 0.8 mSv for the reading of the second TLD.

The readings of the dosemeter in mixed ß-Y-radiation fields were calculated from the response of the three TLD's to ß-radiation and Y-radiation of various energies. The calculations were carried out for over 400 different radiation fields. It was established that the dose may be determined precisely in the vast majority of cases. However, there are radiation fields in which the dose is incorrectly determined by up to a factor of two. These are radiation fields with low-energy ß-radiation (Pm-147) and Y-radiation. Apart from radiation fields with low energy ß-radiation, the dose may be determined with an accuracy of ± 25 % in all other radiation fields.

The new extremity dosemeter is sturdy and is particularly suitable for dose determination in pure β- and γ-radiation. The dosemeter is also suitable for mixed β-γ-radiation fields if no low-energy β-radiation is present. The dose may be incorrectly determined by up to a factor of two in mixed radiation fields with low-energy β-radiation.

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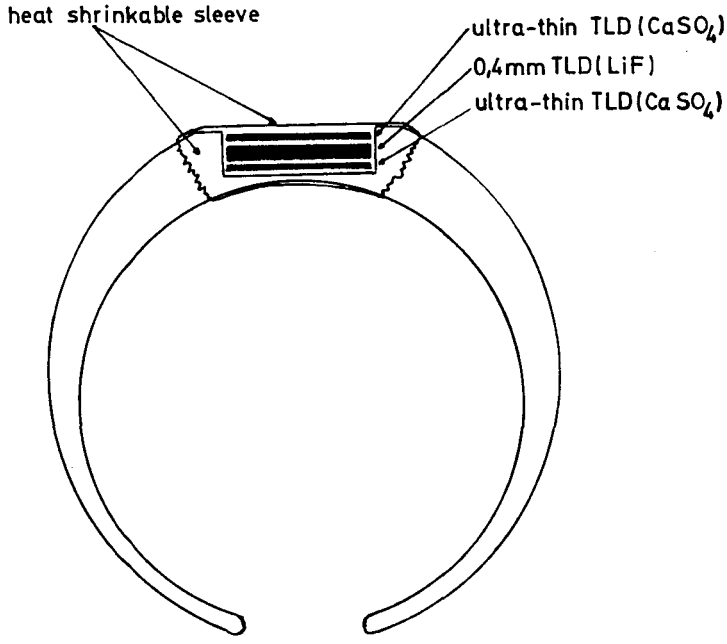

Fig. 1: Diagram of the modified extremity dosemeter

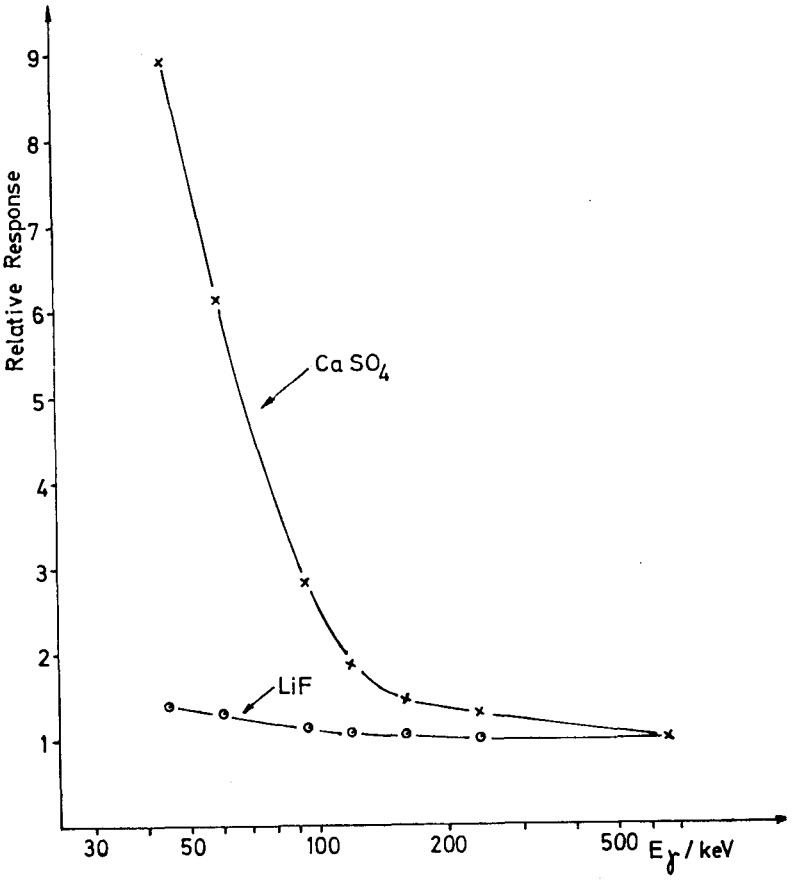

Fig. 2: Energy dependence of the response for Y-radiation of the first TLD ($CaSO_4$) and the second TLD (LiF) of the dosemeter

Table 1: Response, R, of the first TLD, the ratio $D_1:D_2$ of the reading of the first TLD to the reading of the second TLD and $D_1:D_3$ of the reading of the first TLD to the reading of the third TLD for irradiation with the PTB secondary standard

Nuclide	R	$D_1 : D_2$	$D_1 : D_3$
Pm-147	0.55	≈ 15	3.3
Tl-204	0.92	1.8	6.5
Sr-90/Y-90	0.98	1.0	1.1

A FEASIBILITY STUDY FOR DOSIMETRY WITH THERMALLY STIMULATED EXO-ELECTRON EMISSION (TSEE) IN BeO

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INTRODUCTION

The objective of this work was to determine the feasibility of thermally stimulated exoelectron emission (TSEE) from BeO thin film detectors as a method for low energy beta dosimetry. The research included the summarization of related papers and experimental work to verify the claimed characteristics of these detectors.

Experiments were performed to determine:

a) necessity for Geiger counter cooling
b) the effect of counter gas flow-rate variations
c) correct anode voltage for exoelectron measurement
d) annealing temperatures required for maximum sensitivity
e) fading of signal and activation of the detectors by fluorescent tube lighting
f) response of unirradiated detectors (noise)
g) lower detection limit
h) reproducibility
i) fading and activation at 50°C
j) linearity of detector response to radiation dose
k) sensitivity to beta and gamma radiations

DESCRIPTION OF THE MEASUREMENT TECHNIQUES

The detectors used here were developed at the Battelle Institute, Frankfurt and the University of Giessen, both of the Federal Republic of Germany (Kottler and Lerch, 1980). The preparation of the detectors consists of the deposition of a thin Be film on a graphite disc, followed by oxidation in a wet nitrogen atmosphere at 1300°C.

The exoelectron reader was a DIGITEC (Lesz and Holzapfel 1985) methane gas-flow, mono-point Geiger counter with a built-in heater. Readouts averaged 7 minutes per detector and a calibration cycle required 5 minutes.

Thorough flushing of air from the Geiger chamber was very important. Air contamination shortened the Geiger plateau length, changed the counting characteristics and increased the chance of sparking damage to the anode.

On start-up, with the ^{14}C calibration source and guarded anode, the Geiger plateau was 400-500 V in length and had a slope of less than 2 %/100 V. The plateau did slightly shorten and the slope slightly increase after several heating cycles.

The anode voltage for exoelectron readouts was chosen to be 3400 V (with 99.995 % methane and a guarded anode). This voltage gave a well defined Geiger peak in the pulse height distribution (PHD) and did not cause counter saturation at higher pulse rates. This voltage was 200 V higher than that used for the ^{14}C calibrations. This discrepancy is explained by the fact that the Geiger plateau for ^{14}C is not transferable to exoelectrons.

The discriminator lower level cutoff voltage was set to equal the position of the minimum between noise and Geiger peak in the pulse height distribution, rather than the electronic noise level (0.2 V). This choice of threshold eliminated less than 10 % of discernable counts, was very easy to determine and was stable. Counting rates resulting from 4,3 mGy doses (less than 2000 s^{-1}) did not shift the PHD enough to warrant changing the threshold level.

The counting cycle was made almost completely automatic by interfacing with the heater control. Some of the detectors had their second emission peak extending to 520°C. However, since the noise counts increased significantly at the higher temperatures, 500°C was used as the upper bound on peak 2.

RESULTS AND DISCUSSION

Cooling of the counter was found to be necessary after it was noticed that the calibration source pulse rates were varying as a function of the number of heating cycles that had been performed. Experimental heating of the counter increased pulse rates. This effect was probably due to an increase in the gas amplification factor caused by methane expansion as it passed by the heater assembly.

Gas flow-rate did not affect the ^{14}C calibration source pulse rate, with an anode voltage of 3100 V. This was in agreement with the DIGITEC counter manual. However, Geiger plateau length was shortened by as much as 100 V with maximum gas flow-rates. The effect of gas flow-rates on exoelectron counting characteristics was not determined.

PHD's from TSEE readouts did not have well defined Geiger peaks when readouts were performed with an anode voltage chosen with the ^{14}C calibration source. The incorrect anode voltage resulted in an apparent poor detector sensitivity and stability. An attempt was made to define the exoelectron Geiger plateau by plotting total counts (at constant dose) vs anode voltage. The Geiger plateau was not obvious enough to determine an anode working voltage. Instead, an exoelectron working voltage was found by observing the TSEE PHD's as anode voltage was increased. The optimal 3400 V (with 99.995 % methane and guarded anode) was cho-

sen because it was high enough to provide a well defined Geiger peak and not so high that it caused counter saturation at higher count rates.

For complete emptying of the electron traps, annealing by heating at 2°/s, to 550°C, was found to be satisfactory. This compares to a final temperature of 450°C (reading Peak 1 only) by Kottler and Lerch (1980) and 580°C by Lesz et al (1985). Higher anneal temperatures did not reduce the residual counts measured when a second readout was performed immediately after the anneal.

Exposure to fluorescent tube light at a distance of 30 cm caused both TSEE signal fading and activation. Fading in Peak 2 was negligible but Peak 1 counts had decreased by 15-40 % after 30 minutes. Activation occured in Peak 1 with an apparent dose of 1-7 μGy/hour and in Peak 2 at 13-23 μGy/hour. These effects would be decreased by about a factor of 10 in a room with ceiling mounted lights.

Unirradiated detectors had a noise level of 5-12 counts in Peak 1 and 50-130 counts in Peak 2 (guarded anode), independent of detector sensitivity variations. Sensitivities were 7-14 exoelectrons/μGy · cm^2 in Peak 1 and 20-40 exoelectrons/ μGy · cm^2 in Peak 2. These sensitivities compare with total sensitivities of 30-50 exoelectrons/μGy · cm^2, reported by Kottler and Lerch (1980).

Requiring that the signal-to-noise ratio be at least 3, the minimum detectable dose was 5-10 μGy in Peak 1 (detector area = 0.4 cm^2).

The uncovered detector sensitivities were very similar for beta and gamma radiation from Cs-137, Am-241, Pm-147, Tl-204 and Sr/Y-90 radionuclides, as shown in Table 1.

TABLE 1 Sensitivities to Gamma and Beta Radiations for the uncovered detector (for ^{137}Cs a perspex buildup cap was used): The sensitivities were calculated for constant dose at a tissue depth, d=0. Estimated uncertainty in parentheses.

ISOTOPE	PHOTON ENERGY (keV)	AVERAGE β-ENERGY (keV)	SENSITIVITY RELATIVE TO CS-137	
			PEAK 1	PEAK 2
^{137}Cs	662	-	1.0 (6%)	1.0 (10%)
^{241}Am*)	59	-	1.5 (25%)	1.7 (26%)
^{147}Pm	-	62	1.1 (12%)	1.1 (14%)
^{204}Tl	-	244	1.15 (8%)	1.15 (8%)
$^{90}Sr+^{90}Y$	-	570	1.1 (12%)	1.2 (11%)

*) The Am-241 source dose rate was not known accurately

Reproducibility averaged 95 % and was never worse than 93 % for a series of 12 consecutive readouts on each of four detectors (4.3 mGy doses).

A temperature of 50°C did not cause rapid fading or activation in the detectors. Measurements over a period of 17 days indicate a possible 5-10 % fading per 10 days.

Linearity of dose response was verified in the range of 5 - 570 μGy. This is in agreement with all reports to date.

CONCLUSION

Experiments performed for this report were generally in agreement with available literature. Exoelectrons and ^{14}C betas were found to have very different counting characteristics, with no Geiger plateau to be found for exoelectrons. Fluorescent lighting of the detectors caused significant fading and activation but exposures while performing a readout were acceptable. Storage in darkness was necessary. Total sensitivities were 34-50 exoelectrons/μGy · cm^2, giving lower detection limits of 5-10 μGy in Peak 1 and 20-30 μGy in Peak 2. Measurements had better than 94 % reproducibility with consecutive 4,3 mGy doses, however, sensitivity varied from week to week. Linearity of detector response to dose was shown within a range of 5 to 750 μGy. Counter cooling was necessary and gas flow-rates did not affect the calibration source count rates. A final anneal temperature of 550°C was found to be sufficient to restore maximum detector sensitivity. The uncovered detectors showed similar sensitivity to gamma and beta radiation from Cs-137, Am-241, Pm-147, Tl-204 and Sr/Y-90.

The completed experiments indicate that BeO TSEE dosimetry can be useful for measuring low energy beta or mixed field radiations. The stability of detectors and counting technique needs to be investigated further. BeO TSEE dosimetry may still be considered feasible but more testing should be performed before practical application of the technique.

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AUTOMATIC PHOSPHATE GLASS DOSIMETRY SYSTEM USING PULSED UV LASER EVALUATION

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INTRODUCTION

Photoluminescent detectors became once more attractive in routine photon dosimetry after N_2 lasers of high intensity and reproducibility are commercially available. The UV excitation with pulsed laser, in particular, reduces the pre-dose reading by two orders of magnitude and makes the previously used glass cleaning procedure unnessecary. Flat phosphate glass dosemeters with energy compensation filters have now the advantages of measuring low doses in the 10 µSv range and being practically independent of energy and the direction of the radiation incidence down to photon energies of 10 keV.

On the basis of long-term routine experience with photoluminescence dosimetry (PLD) systems (1,2) and previously performed investigations using flat glass dosemeters (3) in combination with the Toshiba reader FGD-8 with pulsed UV-laser excitation (4), Toshiba Glass and KfK recently agreed in a joint cooperation program to design and build-up a modern automatic read-out system using pulsed UV laser excitation. A prototype of the commercial automatic read-out system FGD-10 was built-up at the end of 1987. The full read-out system makes use of a dosemeter encapsulation unlocked and opened automatically within the reader. A microprocessor controlled evaluation technique allows the flexible change of the read-out procedure by the choice of five different read-out modes, the simultaneous indication of different dose quantities and the automatic exchange of high dosed by annealed glass cards as well as a long-term data storage of the accumulated dose.

DOSEMETER AND READ-OUT SYSTEM

The slide type flat glass dosemeter capsule (size 4x3x0.8 cm^3) has been developed for the automatic reader using UV laser pulse excitation (Fig. 1). Toshiba FD-P16-7 glass element (size 16x16x1.5 mm^3) is fixed in a stainless steel card bearing the card No. (hole code). The glass card is housed in the plastic capsule having the energy compensation filters consisting of perforated tin filters of 0.75 mm thickness. The capsule is locked by a magnetic latch and can be released only in the read-out instrument. The capsule is marked with ID No. (bar code).

Toshiba FGD-10 is a full automatic PLD read-out system. The mechanical unit (Fig. 2, A) allows by magazine supply the continuous reading of up to 500 dosemeters as well as the automatic exchange of high dosed by annealed glass cards, the opening and closing of dosemeters, an auto-reading of ID No. for capsules and cards. The optical part (Fig. 2, B) consists mainly of the pulsed nitrogen gas laser, optical filters and diaphragm in front of the glass and the photomultiplier (PM), respcetively. The size of the last one is automatically exchangable to indicate different dose quantities. Eu-doped calibration glasses and dosed reference glasses are used for the continuous calibration of the read-out system and the laser pulse intensity, respectively.

In order to suppress the pre-dose reading caused by primary fluorescence in the glass and dirtyness on the glass surfaces the measurement unit (Fig. 2, C) makes use of the following technique: After the pulsed UV excitation of 5 ns and a delay time of about 2 µs the microprocessor supported evaluation separates a short-term and long-term luminescent component which are integrated in the periods (2-22) µs (area F_1) and in the period (40-60) µs (area F_2), respectively. The actual F_2 value serves for the correction of pre-dose. The difference $(F_1\text{-}a\text{-}F_2) = M$, indicated by the display, is then proportional to the dose. During read-out of annealed glasses the batch dependent factor „a" is set automatically that

just positive pre-dose values are indicated. The reader allows the choice of 10 to 50 pulses per read-out, resulting for the higher pulse frequency in a lower random error of measurement in particular in the dose range below 0.5 mSv.

All readings are carried out full automatically by control with a computer (Fig. 2, E), which fulfills any functions in 5 different modes of read-out: data processing, drawing up of documents, data filing and transmitting to a host computer. The on-line desk computer/printer indicates card and capsule number, the pre-dose, the dose, the daily accumulated dose between entrance/exit control and the total accumulated dose for a dosemeter card and capsule. Personal data can be stored for a group of 10 000 dosemeters/persons.

ENERGY AND ANGULAR RESPONSE

The size of the diaphragm in front of the PM and thus the ratio of the tin and plastic covered glass volume may be changed. By this technique different dose quantities may be indicated, for instance the directional independent dose quantity „exposure" X free in air and the new directional dependent dose quantity H'(10) on the phantom (Fig. 3). Using for calibration a water-filled sphere phantom of 30 cm diameter which simulates the ICRU sphere, the energy dependence has been found to be within ±10 % in the photon energy range 10 keV - 1.3 MeV. On the basis of a free air calibration the same glass dosemeter indicates the dose quantity exposure within ±20 % above a photon energy of 25 keV. For these dose quantities and energy ranges, in personnel dosimetry the angular reponse has been found to be within ± 30 % in the angular range 0° to 60°. For an application in environmental monitoring the angular response in isotrop fields is simulated by rotation in two axis.

Furthermore, the ratio of the PL intensity measured in the glass volumes behind the plastic and tin filter, respectively, may be used as an estimate of radiation quality in the photon energy range below 100 keV.

DOSIMETRIC PROPERTIES

The lower detection limit H_{LDL} defined as two times the standard deviation of the pre-dose of glasses unirradiated was found to be 30 µSv and 7 µSv for 10 laser pulses and 200 laser pulses per read-out, respectively. Assuming a monitoring period of 1 month, for instance, the dosemeter may thus indicate the natural radiation background H_{nat} = 60 µSv with an random uncertainty (1 σ value) of ± 15 % and together with an additional occupational dose of 50 µSv within ±4 %.

After a total accumulated dose of 3 mSv the lower detection limit for the estimation of additional doses is H_{LDL} = 0.1 mSv. On this basis the dosemeter may accumulate H_{nat} over a period of 4 years before an annealing procedure is necessary. For occupational annual doses below the maximum permissible annual dose limit a one-year dose accumulation in personnel dosimetry seems to be appropriate to estimate the annual dose of a person with the lowest random uncertainty of better than ± 5 % for the total dose.

Tab. 1 compares the basic dosimetric properties of the PLD system with those of a TLD system. Assuming also for the TLD system an oven annealing and an individual pre-dose subtraction the pre-dose values, the lower detection limit and the random uncertainty in the linear dose range are comparable for both systems. The PLD system, however, offers batch uniformity in pre-dose and response and thus corrections of the individual response are not nessecary). The read-out of glasses can be repeated. A high energy threshold, one of the main drawback of phosphate glasses in the past, turned out to be superior to that of LiF, above all with respect to the indication of the new dose quantities. Temperature during exposure and storage does not seriously affect the reading and the dose stability with respect to environmental effects is higher than those for LiF.

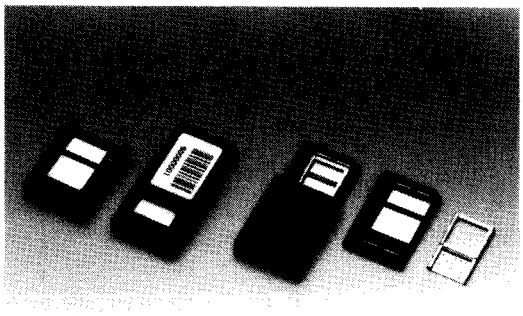


Fig. 1 SC-1 flat glass dosemeter

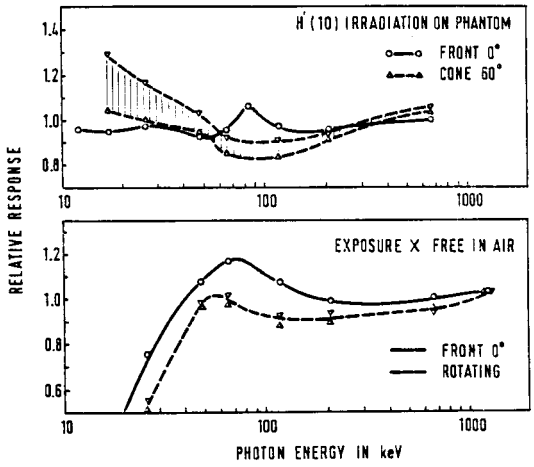
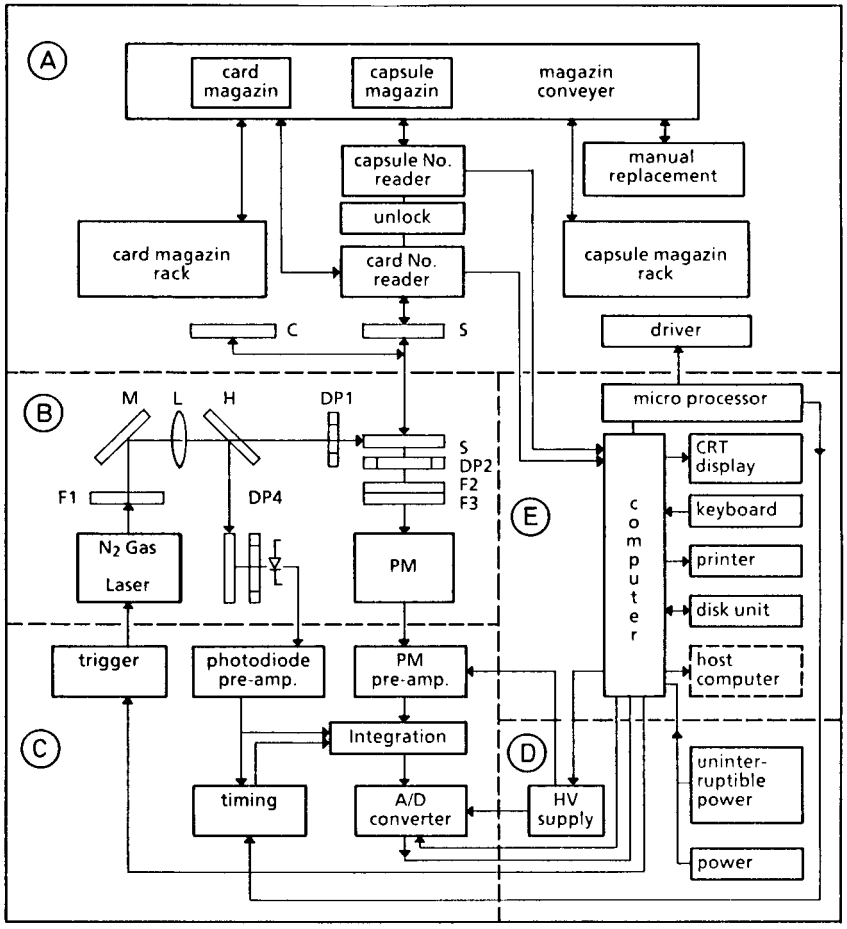


Fig. 3 Energy dependence of the glass dosemeter



M: mirror
L: lens
H: half mirror
DP: diaphragm
PM: photomultiplier
D: photodiode
F1: UV pass filter
F2: UV cut-off filter
F3: orange filter
S: sample glass
R: reference glass
C: calibration glass

Fig. 2 Diagram of the TOSHIBA FGD-10 glass dosimetry read-out system

Tab. 1 Dosimetric Properties of Phosphate glass and TL Dosemeters Using Automatic Readout in the Toshiba FGD-10 and Alnor Dosacus reader

	PLD System (FGD-10)	TLD System (LiF:Mg,Ti)
Dose range	0.03-10 Sv	0.03 - 1 Sv
Batch uniformity		
response	± 2 %	± 10 %
pre-dose	(0.03 ± 0.02) mSv	(0.06 ± 0.03) mSv
Random uncertainty (H_{nat} + 50 µSv – 110 µSv)	⊥ 5 %	⊥ 10 %
Energy dependence		
Exposure free in air	25 keV - 1.2 MeV ± 20 %	15 keV - 1.2 MeV ± 20 %
H'(10) on phantom	10 keV - 1.2 MeV ± 10 %	20 keV - 1.2 MeV factor 2[1]
Fading at 20° C	- 10 %/10 years	-20 %/1 year

1) Without energy compensation filter

CONCLUSION

In conclusion the new automatic dosimetry system for a large-scale application of personnel dosemeters offers the following advanteages:

- The technique of the automatic read-out namely the microprocessor supported evaluation, the reliability of measurement, the read-out time for one dosemeter and the capacity of magazins and data storage as well as the dosimetric properties of the system such as detector sensitivity, lower detection limit and random uncertainty of dose measurement are comparable with those of modern automatic TLD systems.
- The excellent batch uniformity in response and pre-dose, the relatively low energy dependence of dose measurement for the simultaneous indication of different dose quantities and the long-term stability of dose information with respect to repeated read-outs and environmental effects are, however, properties so far not offered by TLD systems.
- Phosphate glass dosemeters are still superior to TLD systems with respect to the simplicity of the read-out procedure, the permanent availability of dose information, in particular the capability to repeat the read-out procedure and to indicate the accumulated dose between daily, monthly and annual read-outs. In the case of accidents high dose values can be measured and the radiation induced effect can be confirmed by an annealing of the reading followed by a recalibration of the actual glass with respect to response and pre-dose.

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WHAT IS THE BEST ANNEALING TREATMENT FOR LiF DOSEMETERS? SOME EXPERIMENTAL RESULTS

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1. INTRODUCTION

Many references in the Literature, when dealing with the problem of TLD annealing, have recommended various solutions, according to the material to be treated. For lithium fluoride, one of the most widely recommended pre-irradiation annealing procedures consists of a two-phase cycle: 1 hour at 400°C (higher termperature phase) and 2 hours at 100°C (lower temperature phase).

The aim of this work is to assess whether this method is really the optimum or some variations in termperature or time should be introduced. The paramenters considered to evaluate the effectiveness of the annealing were sensitivity (as signal-to-dose ratio) and reproducibility.

2. MATERIALS AND METHODS

The dosemeters used for the present study were sintered LiF:Mg,Ti (TLD-100) chips, individually calibrated. The experimental procedure can be summarised as follows:
1 - Pre-irradiation annealing
2 - Irradiation
3 - Post-irradiation annealing
4 - Readout
5 - Data processing

As regards the pre-irradiation annealing, the lower temperature treatment has been kept constant at 100°C for 2 hours, whereas the higher temperature treatment was varied in the range between 300 and 500°C. Great care was exercised to keep constant both annealing time (+/- 10 seconds) and location of dosemeters (+/- 2 mm) into the furnaces. During this annealing, a series of five Chromel-Alumel thermocouples inserted into the furnaces were used to measure the actual temperature of the TLDs; gradients and time-dependent fluctuations of these temperatures were also checked.

The irradiations were carried out by a 60-Co Gammabeam 150-C facility.

The post-irradiation annealing was performed at 100°C for 15 minutes.

The TLDs were read out on a semi-authomatical TL analyser, Harshaw 2000-D, with a capacity of 50 dosemeters. A 300°C nitrogen flux was used in this reader. In order to allow for the long-term instability of the TL analyser, all the readouts of each reading session were normalised to the mean response of 25 additional TLD-100 chips annealed and irradiated according to a standard procedure and read during the same session.

3. EXPERIMENTAL TECHNIQUE

The parameters considered as variable in the present experiment were:
1 - Temperature of the first phase of pre-irradiation annealing;
2 - Irradiation level.

The temperature was varied in the range between 300 and 500°C, in 50°C steps. Two levels of dose were selected: 50 and 500 mGy.

Fifty dosemeters were randomly selected from the same batch and used throughout the experiment. Each temperature level was investigated by carrying out 4 annealing-irradiation-readout cycles.

4. RESULTS

A decrease in sensitivity was systematically obse ved in t e TLDs after the 500 C cycles. In order to allow for this phenomenon, the evaluation of

sensitivity at the reference temperature (400°C) was repeated before each variation of the annealing temperature.

Fig. 1 shows the reproducibility, in terms of variation coefficients, of the sensitivity obtained in the 4 cycles performed at each temperature. The experimental points are scattered in the range from 0.7 to 3.7%; the statistical analysis (F-test) of the data does not reveal any significant trend.

Figs. 2 and 3 show the mean relative sensitivity of LiF as a function of annealing temperature. The agreement of the experimental points with the decreasing exponential functions reported in the inserts is within 2%. The deviations are still lower for the 5 mGy dose level.

These figures also compare the present results with the ones published by Regulla (1981).

5. DISCUSSION

The experimental results point out that the relative sensitivity monotonically decreases in the range 300 to 500°C, with a slight negative slope below 400°C and a steeper one for higher temperatures. This behaviour is somewhat different from that found by Regulla (see Figs. 2 and 3) who observed a positive slope below 400°C, with a maximum around this temperature. However, the comparison of these two sets of data is difficult as some basic experimental parameters (e.g. dose level, lower temperature pre-irradiation annealing, etc.) cannot be found in Regulla's paper.

The decrease of sensitivity with annealing temperature can be connected with the existence of trapping levels at different energies below the conduction band. A Phenomenon of thermal destruction of centres corresponding to shallow levels is likely to occur for very high temperatures. On the contrary, for temperatures lower than 400°C, the trap concentration does not undergo a remarkable reduction, so that the slight increase in sensitivitiy for decreasing temperatures might be explained in terms of charge and centre kinetics between levels situated at different energy depths

Temperatures higher than 400°C should therefore be avoided for LiF annealing treatments because the TLDs may be permanently damaged. On the contrary, when carried out at a temperature level less than 400°C, the annealing does not lead to a relevant increase of sensitivity and, moreover, the trap emptying may be incomplete.

In conclusion, the results of the present experiment show that the optimum temperature for the first phase of the annealing cycle of LiF is not far from the universally accepted value of 400°C. Further measurements are now in progress in order to improve the resolution of the experiment, taking into account not only sensitivity and reproducibility of dosemeters but also possible 'memory' effects due to previous irradiations.

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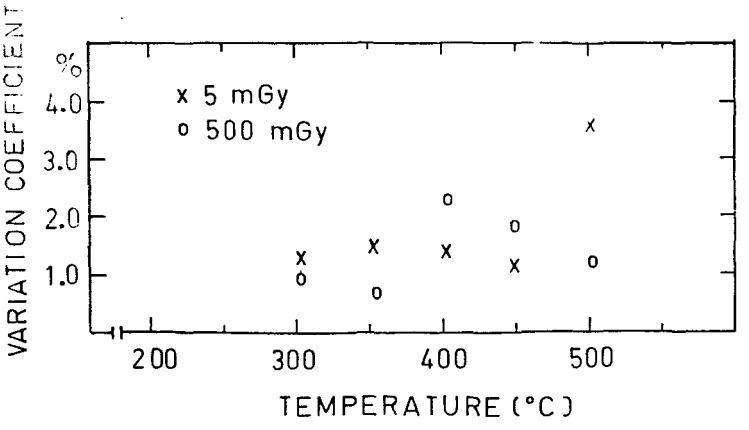

Fig. 1: Reproducibility of LiF as a function of the temperature of the pre-irradiation annealing

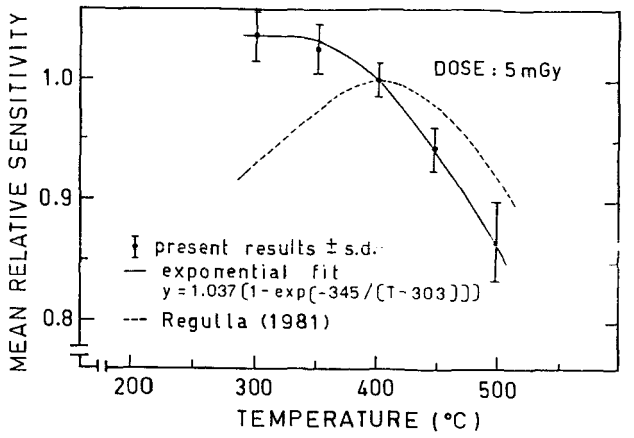

Fig. 2: Mean relative sensitivity of LiF as a function of the temperature of the pre-irradiation annealing.
Dose level : 5 mGy

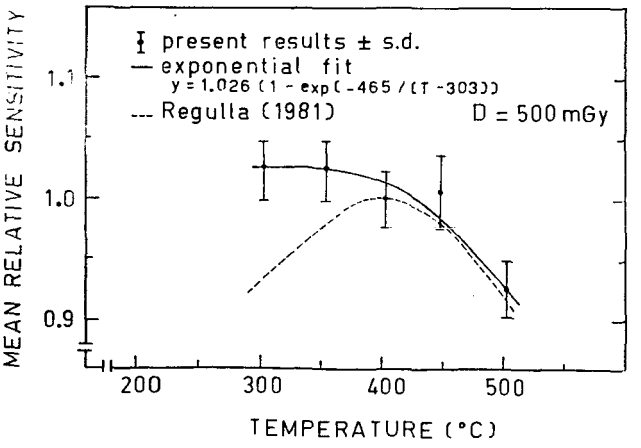

Fig. 3: Mean relative sensitivity of LiF as a function of the temperature of the pre-irradiation annealing.
Dose level : 500 mGy

EXPERIENCE IN PERSONNEL BETA DOSIMETRY IN AN ARGENTINE CANDU REACTOR

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1.- INTRODUCTION

This paper describes the present difficulties existing through out the world for the execution of a correct Beta personnel dosimetry together with the method implemented at this Nuclear Power Station -(600 MWe CANDU REACTOR)-

This method intends to use conservative hypothesis so as to overestimate the measured doses. At site, examples of Beta radiation fields are as follows: the air of rooms of the Reactor Building in wich Noble Gases would exist (see Ref. N°1) and zones or equipments with high values of external superficial contamination.

Beta radiation is formed by electrons having an associated spectrum of energy. This spectrum, which is typical for each isotope, has a medium and a maximum // energy .This arises one of the main problems regarding Beta Dosimetry as we do not work with monoenergetic radiation as in the case of Gamma radiation.

Due to the ranges of the most ordinary isotopes in human tissue, those that will be affected by the external Beta radiation are as follows: The skin that covers the subcutaneous tissue, more precisely the ones called basal cells wich are placed at the limit of the dermis and the epidermis, the crystalline of the eye and the gonads (see Ref. N°2).

The depht of these radiosensitive cells vary according to the part of the human body being considered, for example: 7 mg/cm2 average in the skin of the head and trunk, 30 mg/cm2 in the hand palm and 300 mg/cm2 in the case of the crystalline. To solve this second inconvenient the value of 7 mg/cm2 is taken as reference for the specification of the Beta dose.

The methods used worldwide for the execution of the dosimetry based on TLD can be divided in Three main groups : a) according to TLD's crystals, which are quite thin, in order that they have an independent response of the energy of Beta radiation (see Ref. N°3), b) based on badges with two TLD's crystals and a calibration factor,c) using multielement badges to determine the maximum effective energy .

Regarding Embalse Nuclear Power Station,(see Ref.N°4,5,6), the dosimeters used / are those grouped in the last two methods. At present, the group of workers -20- more exposed to high Beta fields, use multielement type dosimeters and rings with two TLD's crystals for the hands. The rest of the personnel -390- authorized to enter the controlled zone use bages with two TLD's crystals.

It should be noticed the remarkable importance of the Beta dose magnitud being / dealt with at a Nuclear Power Station. The average obtained values of Beta dose at site are examples of the above importance. They are as follows :

NORMAL OPERATION : (12 months)	COLLECTIVE BETA DOSE : 0,22 man-SIEVERT
	PERSONAL MAXIMUM BETA DOSE : 5,9 mSv
SCHEDULED SHUT-DOWN : (1 month)	COLLECTIVE BETA DOSE : 0,70 man-SIEVERT
	PERSONAL MAXIMUN BETA DOSE : 146,6 mSv

2.- EXPERIMENTAL DETAILS

2.1. CURRENT USE BADGE : They have a card Trade HARSHAW Model 4BG11 containing / two crystals of lithium fluoride (TLD-100) of 0,9 and 0,38 mm thick respectively. The corresponding readings to Beta and Gamma radiation are obtained by using a / badge containing an aluminium filter and a window of 8 mg/cm2. The Gamma component is evaluated with the usual equations.

Regarding the calculation of Beta dose, it is considered that, as these dosimeters do not have the possibility that could allow us to know the effective maximum energy of the incident Beta radiation, it is necessary to make certain suppositions to obtain the reading conversion factor into absorbed dose at a depth of 7 mg/cm2.

Accordingly, it is assumed that these dosimeters are used only by workers that / are not so much exposed to irradiations with Beta particles coming from the decayment of Noble Gases and airbornes (Rb-88).

Experimentally,it was proved that, by means of the multielement dosimeters, the corresponding effective maximum energy could be approximately 0,9 Mev, consequently, this value is used for the calculation of the above mentioned factor.

2.2. CASE OF DOSIMETERS OF RING TYPE : They are used as dosimeters of extremities in those tasks involving important Beta fields, i.e. greater than 3 mGy/hour in contact. Each worker uses in the same hand a pair of receptacles trade HARSHAW , ring type and each ones contains a TLD-100 chip of 235 mg/cm2 thick.

One of the crystals is covered by a thin filter of 7 mg/cm2 thick, and the other is covered by an aluminium sheet of 1 mm thick.

2.3 CASE OF MULTIELEMNT DOSIMETERS : They are used in the same way the rings are,

in those tasks involving important Beta fields. These dosimeters, which were developed at Site, consist in one card -Trade HARSHAW-Model 4G1111 containing four TLD's crystals 0,9 mm thick each.They are covered in their front part by filters of different thickness: 540, 63, 8 and 1,6 mg/cm2.

Based on the measurement of the transmission factors of the different filters for the incident Beta radiation , it is possible, by means of the corresponding equations,to derivate the maximum energy value and dose value being searched. Sometimes,these multielement dosimeters are used as Area Monitors. The following experimental comparison details one example of that use.

This comparison consisted in placing multielement dosimeters on the wall of different rooms containing Noble Gases distributed in the air of the Reactor Building. Air samples from different rooms were taken every 8 hours, during 7 days,and the concentration of every radionucleid was determined by Gamma spectrometry.
The results are detailed in the following tables:

ISOTOPE CONCENTRATIONS (Bq/m3)

ROOM	Xe-133	Xe-135	Kr-85m	Xe-133m	Kr/Rb-88
R-501	5,2 E 7	8,5 E 5	1,2 E 5	2,3 E 6	9,2 E 4
R-007	1,2 E 7	3,2 E 5	5,2 E 4	5,2 E 5	4,8 E 4
R-008	1,0 E 7	3,0 E 5	3,5 E 4	4,4 E 5	4,8 E 4

MEASURED AND CALCULATED DOSE RATE (mSv/Hour)

ROOM	CALCULATED WHOLE BODY GAMMA DOSE	CALCULATED SKIN GAMMA DOSE	MEASURED GAMMA DOSE (TLD)	CALCULATED BETA DOSE	MEASURED BETA DOSE (TLD)
R-501	1,0 E-2	1,4 E-2	1,3 E-2	0,68	0,55
R-007	2,7 E-3	3,7 E-3	3,5 E-3	0,18	0,08
R-008	2,4 E-3	3,2 E-3	4,7 E-3	0,14	0,09

If we compare the average differences between the measured Beta dose rate values and those calculated, a value of 38% is obtained. Anyway,we can observe, within

the experimental errors, a very good correlation between the Beta-Gamma measured values and the calculated ones.

3.-CONCLUSIONS

The above described method always uses conservative hypothesis that allow us to calculate absorbed Beta doses greater than the actual ones.Example of the above is the fact that, regarding rings the depth of 7 mg/cm2 is considered, while the actual average depth of the basal cells in the palm of the hand is 30 mg/cm2.This makes the actual dose they absorb to be less than the calculated,i.e (80% less).

Also,in order to prevent the hands from getting contaminated, they are covered / with gloves, which is not taken into consideration either. Similarly,the clothes worn within the controlled zone,which cover all the body excepting the face, have an average thickness of 50 mg/cm2.

For this reason, this value should be taken in mind when calculating the dose / factor,as any Beta radiation with an energy less than 0,6 Mev will be attenuated by the above said clothes. The need to use conservative methods is based on the fact of compensating, among other things, the decrease of the Beta reading component due to the angular dependence of the most of the badges existing at present.

The routinary use of multielement dosimeters in high Beta fields has proved to be practically an indispensable tool in the calculation of the involved Beta dose. Unfortunately, the above mentioned use is not worlwide spread at present. Any possibility regarding either exchange of information or dosimeters will be highly appreciated at the Embalse Nuclear Power Station - Argentine.-

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THE U.S. DEPARTMENT OF ENERGY PERSONNEL DOSIMETRY EVALUATION AND UPGRADE PROGRAM

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INTRODUCTION

The U.S. Department of Energy (DOE) Personnel Dosimetry Evaluation and Upgrade Program is designed to identify and evaluate dosimetry deficiencies and to conduct innovative research and development programs that will improve overall capabilities, thus ensuring that DOE can comply with applicable standards and regulations for dose measurement. To achieve these goals, two programs were initiated to evaluate and upgrade beta measurement and neutron dosimetry.

DISCUSSION

Beta dosimetry has had radiation protection problems for many years, compounded by a need to improve the accuracy and precision of beta measurements. Generally, devices designed to measure other radiations (usually gamma) are adapted for beta measurement. These devices can underestimate beta dose by a factor of 10 or more because of the complex spectra and low-penetrating ability of beta radiation, which produce significant spatial and spectral variations in the radiation fields.

DOE facilities use a variety of instruments, calibration sources and procedures, and personnel dosimeters to assess beta exposure. In most cases, facilities cannot meet the accuracy criteria in national or international standards [1].

The measurement of neutron dose has similar problems. For the same dose, neutrons can produce many times the biological damage of gamma rays, indicating that the degree of hazard from neutron radiation may be higher than previously assumed. Previously, concerns about neutron dose were less because it was only a small fraction of the total dose. Anticipated changes in neutron quality factors increase that fraction and put greater emphasis on the need for improved measurements.

It is necessary to estimate the neutron dose at various depths in the body and on the surface. In January 1987, President Reagan signed the "Radiation Protection Guidance to Federal Agencies for Occupational Exposure; Approval of Environmental Protection Agency Recommendations," [2] which adopts the ICRP-26 [3] methods. The ICRP-26 recommends that in non-uniform fields the dose equivalent be determined by summing the dose equivalent to various specified tissues multiplied by a set of weighting factors. Almost all DOE radiation workers exposed to neutron radiations are in non-uniform fields. ICRP-26 and the proposed

changes in quality factors require an assessment of neutron organ dose and improvement in neutron dosimetry capabilities.

The Neutron and Beta Programs are structured by tasks that address specific problem areas. Work is accomplished by a unique combination of contracts and subcontracts which focus the best possible expertise on the task. National laboratories, universities, and private companies act as subcontractors to PNL to accomplish this research. Consulting committees are used to guide program developments, and interactions with the National Bureau of Standards assure that measurements and results are consistent with national standards.

DOSIMETER DEVELOPMENT

Beta Dosimeters

New dosimeter configurations and materials are being developed as well as innovative techniques for evaluating dosimeter results. These include:

- A laser readout for conventional and experimental TLDs was developed, a prototype built, and problems limiting the usefulness of the concept identified. TLDs are being evaluated.

- A thin LiF dosimeter bonded to a graphite backing was developed to provide superior beta response. The units are being evaluated in badge configurations.

- Optically-stimulated luminescence dosimeters use light to de-excite trapped electrons in solid-state materials which then emit fluorescence radiation that is detected by a photomultiplier. The technique is used to anneal and read out dosimeters without increasing the temperatures.

- A re-evaluation of exoelectron dosimeters has occurred because of the recent development of new materials that are more sensitive than TLD materials and do not exhibit an energy dependence. Their superior performance has been demonstrated, resulting in their incorporation into a dosimeter for evaluation.

Neutron Dosimeters

In response to trends toward revising dose equivalents, standards, and limits, new dosimeters being developed include:

- An interim dosimeter was developed consisting of thermoluminescent (TL) and track-etch (TE) elements in a combination TL/TE dosimeter (TLD/TED). Neutrons above about 100 keV are detected by the TED plastic, CR-39. Lower-energy neutrons are detected by the TL component. The combination TLD/TED can nearly match the existing fluence-to-dose equivalent conversion curve.

- A newly-proposed personnel neutron dosimeter counts the noble gas atoms released from a solid matrix material by collision events that can be related to neutron dose. A laboratory

reader counts the noble gas atoms released by using a modified laser-induced fluorescence technique that shelves the noble gas atoms in a metastable excited state. Krypton and xenon atoms will be the first evaluated.

- Recent findings regarding the sensitivity of optical fibers to neutron radiation show promise in detecting fast and thermal neutrons. Previous research was directed toward increasing radiation resistance of optical fibers. Optimizing their sensitivity will be key to their dosimetry application.

- Neutron dosimetry devices using semiconductor detectors being developed include: diode devices that collect electron hole pairs as a result of proton adsorption, semi-conductor devices for measuring neutron dose using deep-level transient spectroscopy measurement techniques, and a spectrometer to determine the energy distribution of charged particles.

- Superheated droplets are suspended in a solid-elastic or high-viscous liquid medium. When a neutron strikes a droplet, the energy from the recoil charged particle triggers the explosion of the droplet resulting in an acoustical signal which can be counted. In a solid medium, the gas bubbles are trapped and can be counted after an exposure as passive measure of the neutron dose.

INSTRUMENT DEVELOPMENT

Beta Instruments

Instruments must provide a correct reading of absorbed dose in a mixed beta-gamma radiation field and be able to differentiate between penetrating and nonpenetrating components of the radiation field. Reduction in geometry dependence of survey meters must also be accomplished.

- A thin scintillator (~5 mg/cm^2) was developed that shows excellent beta response; however, current cumbersome electronics make it impractical. Research is underway to develop a practical system using special scintillators to avoid the electronic complexity.

- A combined thin and thick scintillator can discriminate against the signal from gamma events occurring in the thick scintillator. Only the beta particles will deposit energy in both scintillators which can be identified by pulse shape discrimination techniques. A laboratory model of this technique has been demonstrated (phoswich counter).

- A thin proportional counter can be used to assist in separating the beta and gamma signals in a thick scintillator. The technique shows superior beta-gamma discrimination and less complexity than other techniques (phoswich).

- Evaluation of survey meters has shown that a thin (1 to 2 cm) ionization chamber with a large area (to compensate for loss

of volume) will be superior for beta measurements. Prototype chambers are currently under evaluation.

Neutron Instruments

Instruments are being developed and sought that satisfy new measurement criteria; that are energy independent, economical, rugged, and easy to use; and that measure all types of radiation.

- The total dose meter is a small personnel monitor that simultaneously measures the dose equivalent from photon and neutron radiation with a single tissue equivalent proportional counter (TEPC). A unique determination of total dose equivalent from all penetrating radiations is possible.

- A prototype field neutron spectrometer has been built to determine neutron quality factor and dose equivalent in the workplace to determine calibration factors for dosimeters and other instruments. The system consists of ^{3}He and TEPC detectors, a modular multichannel analyzer, and a microcomputer. Components are packaged into a unit the size of a small suitcase. Programs were developed for the ^{3}He and TEPC data analyses, including display of the raw data, analyzed spectra, average neutron energy, average neutron quality factor, and dose equivalent rates with an error estimate of measurement accuracy.

CONCLUSIONS

With changing regulations, the most accurate and sensitive measurement techniques must be available for protecting workers in the nuclear industry. The Neutron and Beta Programs provide an effective mechanism for evaluating and developing the latest measurement techniques. The techniques described span the range of efforts from laboratory evaluation to field applications. As concepts are generated for improving dosimetry, they will continue to be evaluated and developed with the objective of providing better radiation protection in the workplace.

Acknowledgment

This work was performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

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INTERCOMPARISON OF FILM BADGE AND POCKET DOSIMETER FOR Tc-99m X-RAYS

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ABSTRACT

Intercomparison of film badge and pocket dosimeter for Tc-99m x-rays

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An intercomparison was made between the doses measured by film badges and pocket dosimeters for the personnel in a nuclear medicine department. The data was taken over a three year period (1984, 1985, 1986). The film badge data was processed by a commercial supplier (Siemens) and the pocket dosimeters were read and recharged weekly by the Radiation Safety office.

The dose measured by the pocket dosimeter was in very good agreement with the film badge reports. The correlation coefficient was 0.95 and the degression equation was:

$$\text{Dosimeter} = .884\ \text{BADGE} + 61.9$$

The measured mean annual dose was 226 mR for the film badge and 262 for the pocket dosimeter and the total dose range was 20-500 mR.

This data shows that the dose measured by the pocket dosimeters is an accurate representation of the dose recieved by the nuclear medicine personnel when compared to the commercially processed film badges.

Previous intercomparison of these two personnel dosimetry devices was done only for x-ray technologists and the correlation was not as meaningful at the lower x-ray energies.

This study demonstrates that the pocket dosimeters are an accurate and reliable device for personnel monitoring in nuclear medicine.

SKIN DOSE ESTIMATES FROM RADIOACTIVE SKIN CONTAMINATION

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ABSTRACT

ICRP has established a secondary dose equivalent limits based on the deep dose equivalent index for the stochastic effects and the shallow dose equivalent index for the non-stochastic effects. One important contribution to the shallow dose equivalent is that from the radioactive skin contamination. Although skin dose estimates from such contamination are only regarded as qualitative procedures due to the imprecise measures of the percutaneous absorption of contaminants, the assessed result should be included in the individual's personal record if it exceeds one-tenth of the appropriate dose equivalent limits. In this work, we have examined the effects of the percutaneous absorption of pig's skin for several radioactive compounds both in the dry state and the aqueous state of different pH values. A practical methodology has been developed for determining the maximum beta-ray energy and the skin dose equivalents at various depths based on the Loevinger formula and the depth-activity distribution according to the pertaneous absorption data.

RELATIONSHIP BETWEEN $\bar{Q}$ DEFINED IN TERMS OF y FOR 1 μm SITES AND INITIAL RADIATION DAMAGE*

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ABSTRACT

The recent suggestion that quality factor (Q) be defined in terms of lineal energy (y) rather than LET (IN 86) requires specification of the diameter of the site to be used. Since present knowledge of the mechanisms leading to effects of radiation is insufficient to specify the exact site size, an arbitrary choice of site size must be made. A site size of one or two micrometers has been recommended because it can be simulated by tissue equivalent proportional counters, making Q a measurable quantity. However, it is not immediately obvious that y measured in a 1 μm diameter site is relevant to the biological effectiveness of different radiations. Monte Carlo track structure calculations have been used to determine the distribution of energy deposition for protons and electrons at a variety of energies in 10 nm and 1000 nm diameter sites. The results indicate that measurements using large sites can be used to provide a satisfactory estimate of energy deposition in small sites and can thus be used to predict biological effectiveness, even if the effect depends on the concentration of damage in very small sites.

INTRODUCTION

Many years ago an experimental technique for measuring the energy deposited in microscopic volumes was introduced (Ro55). This technique was originally intended as a method for measuring LET, and thus a way to determine the mean quality factor of unknown radiation fields. Tissue equivalent proportional counters are still used in this way occasionally, but certain fundamental characteristics of radiation interactions place limitations on the relationship between LET and energy deposition in small sites. The study of these energy transfer processes and the resulting stochastics of dose in small regions has come to be known as microdosimetry. It has been proposed that the microdosimetric quantity lineal energy (y) be substituted for LET in a new definition of Q (IN 86). The advantages to such a system include the fact that y can be measured directly (while LET must be calculated from the mass, charge and velocity of the ionizing particle), that reasonably simple relationships between y and RBE have been found for many biological systems (Bo83, Ph87) and that y is more nearly related to the actual energy deposition in cell nuclei than is LET for those radiations where energy loss, straggling and delta ray escape may be important. These include most radiations of practical importance in protection from external sources. However, there are also disadvantages to a definition of Q based on y. Such a definition requires choosing the diameter of the site for which y will be specified. It is unlikely that a single site size is relevant to all biological systems and endpoints. Furthermore, data on the biological

*Work supported by U.S. Department of Energy Contract DE-AC06-76RLO 1830

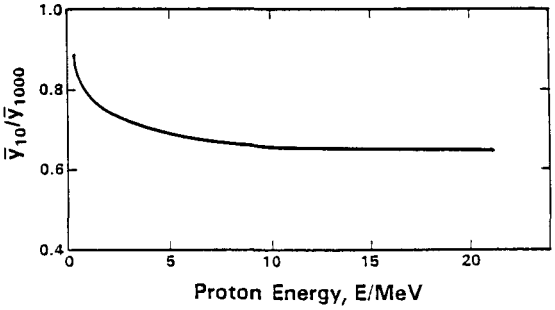

FIGURE 1. The ratio of lineal energy in 10 nanometer and 1000 nanometer diameter sites is nearly constant for protons above 5 MeV.

effectiveness of unusual radiations such as very soft x-rays (Go77) and molecular ions (Ro84) show that the initial damage responsible for common biological effects is governed by energy deposition in very small sites, on the order of 10 nm.

Current experimental techniques are limited to a minimum simulated site diameter of about 300 nm by the requirements of the gas gain mechanism. For low energy charged particles (range comparable to or less than the simulated site diameter), the energy deposited approaches the energy of the charged particle, and y depends directly on the site size. However, these short track radiations are relatively minor problems in radiation protection; x-rays below a few keV do not penetrate the body and neutrons below 100 keV are important primarily when the source is well shielded. This discussion will deal only with radiations which transfer energy via charged particles with range larger than the diameter simulated by the detector. For these long track radiations y is a slowly varying function of the site size.

ENERGY DEPOSITION IN SMALL SIGHTS

The primary tool for studying the energy deposition in small sites is track structure simulation by Monte Carlo techniques. Using suitable atomic cross section data the position of each ionization produced by a charged particle and its secondaries can be calculated (Wi80). The most noticeable characteristics of such tracks are the occurrence of occasional clumps of ionizations, and delta rays which often carry significant quantities of energy away from the path of the primary ion. The results of soft x-ray and molecular ion experiments suggest that the concentration of several ions in a small cluster may be responsible for the biochemical changes which initiate the biological effect. These clusters can be caused by the random occurrence of ionization along an ion or delta ray track, by the overlap of two or more elements of a track (for example,a delta ray track turning and crossing the primary ion track), or from the decreasing mean free path for ionization rear the end of an electron's range. Thus the question of the relationship between measurements in a relatively large site and effects in sites a few nanometers in diameter can be divided into two parts; the ionization produced by the primary ion and that produced by delta ray events.

Figure 1 shows the ratio of the mean lineal energy in a 10 nm site to the mean in a 1000 nm site as a function of proton energy.

It is evident that the lineal energy in the small site is less than in the large site. This is due primarily to transport of energy outside the small site by delta rays. Furthermore, the ratio is relatively constant down to a few MeV, indicating that energy deposited in μm diameter sites can be used as a reasonable indicator of mean energy deposited in much smaller sites. At the lower energies the range of delta rays is short and less energy is transported outside a small site. In this region the overlap of ionizations produced by the primary and those produced by the delta rays becomes more significant, but even in the extreme case the ratio of means changes by only about 50% while the stopping power changes by approximately a factor of ten.

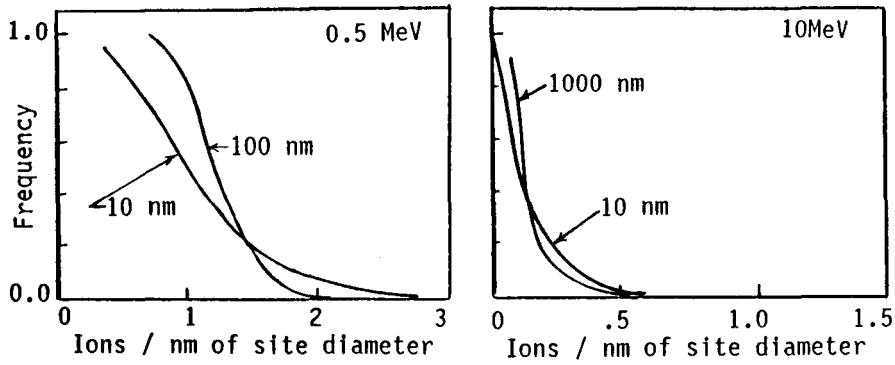

FIGURE 2. The distribution functions for protons crossing through the centers of 10 nm and 1000 nm sites.

If it is the relatively large energy deposition events which are relevant biologically, then one must consider the shape of the distribution of events in addition to their means. Distribution functions shown in figure 2, give the probability that an event will deposit more energy per unit path length than $\bar{y}$. For large site sizes this distribution is a steep curve. For small sites the curve is less steep and crosses the large site curve at about 0.7. Thus for protons from 0.5 to 20 MeV crossing 10 nm sites, the effect of increasing the site size is simply to narrow the distribution of energy deposition events, and increase the mean lineal energy by including some delta ray ionizations which would occur outside a smaller site. The difference between those curves is less for event sizes greater than the median than it is for the smaller events. Thus, for the portion of the energy deposition distribution assumed to be most relevant to the production of biological damage, the large events in small volumes, the energy deposition measured in larger sites would provide a better indication of biological effectiveness than would be suggested by a simple comparison of mean values.

Approximately one third of the energy deposited in matter by protons with initial energy greater than 3 MeV is carried beyond a 5 nm radius around its track by secondary electrons (delta rays). Experimental work using relatively large simulated site sizes (G172) suggest that if only delta rays interact with a site the energy deposition is nearly independent of the primary ion energy as well as the distance from the site to the track. This can be

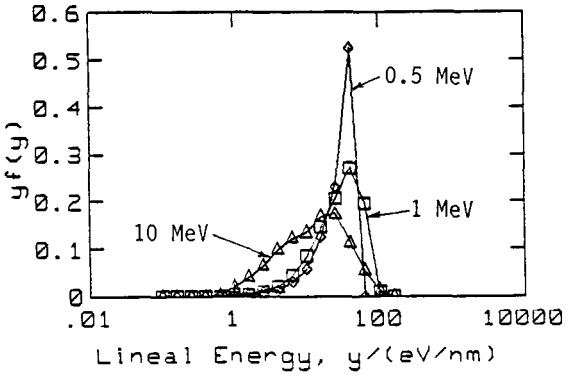

FIGURE 3. Calculated lineal energy distributions in small sites are similar for a wide range of initial electron energies.

understood based on the characteristics of electron tracks. An electron can transfer any fraction of its energy to another electron in a collision so a characteristic distribution of electron energies develops after monoenergetic electrons have undergone only a few collisions. Thus electron spectra differ only with respect to the frequency of the highest energy electrons, and since these have low stopping power they contribute very little to the energy deposition distribution. To illustrate this for very small sites the lineal energy in random 10 nm diameter sites irradiated by electrons of different initial energies is illustrated in figure 3. It is evident that the distribution of delta ray events is essentially independent of the initial particle energy. Thus the energy deposition in large sites, which is a good indication of the total amount of energy transferred by delta rays, can be used to estimate the amount of biological damage caused through this mechanism as well as that caused by the primary ionizations.

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DEPTH DOSE PROFILES RESULTING FROM BETA EMITTERS ON THE SKIN SURFACE

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INTRODUCTION

Contamination of the skin by beta emitters is a common health physics problem for workers at nuclear facilities. Several different methods are in use for evaluating dose from skin contamination, and the different methods often give conflicting results. The depth dose profile is critical to understanding the stochastic and nonstochastic damage to skin tissues. A computer code called VARSKIN has been developed at PNL for the evaluation of beta skin dose using the specific energy deposition tables of Berger. VARSKIN has been used to evaluate depth dose profiles in the first two millimeters of skin for several different contamination geometries (point source, finite disk source, infinite plane source), and for a number of different radionuclides of interest to the health physics community.

COMPUTER CODE VARSKIN

VARSKIN [1] was developed to evaluate skin doses resulting from beta-emitting contamination on the skin surface. VARSKIN assumes that the beta-emitting radionuclides reside on the skin in one of three configurations: a point, a thin disc, or an infinite plane. The code calculates the beta dose to a thin layer of cells at a user-input depth below the skin surface (typically 0.007 cm, matching standard recommendations for the critical depth). The dose is calculated at a number of points at the chosen depth, and these calculated dose values are smeared into an area-averaged dose. This smeared dose is calculated over a circular area centered below the center of the contamination region, and the area for the smeared dose is chosen by the code user (typically 1 cm^2).

The dose calculation in VARSKIN is based on the tables of specific absorbed energy published by Berger [2]. The specific absorbed fraction accounts for the fraction of emitted energy absorbed in a sphere of water centered about a point source of beta-emitters. The dose to any point is calculated in VARSKIN by performing a point-kernel integration over the source region. Doses are calculated to a number of points at the dose level. The choice of dose points is guided by first finding the maximum range of betas. This range allows the determination of an area under the center of a disc source that is essentially exposed to

(a) Work supported by the U.S. Nuclear Regulatory Commission under a Related Services Agreement with the U.S. Department of Energy under Contract DE-AC06-76RLO 1830, NRC FIN B2863.

an "infinite plane" of beta emitters. Only one dose point needs to be chosen in this region, since all dose values will be the same. The range also allows the determination of a region beyond the boundary of the source which is exposed to zero dose. The region of varying dose lies between the inner infinite-plane region and the outer zero-dose region. VARSKIN chooses 26 dose points in this boundary region. This scheme of choosing dose points enables an efficient smearing of the dose over the appropriate dose area.

VARSKIN reads specific absorbed energy fractions and other values from a library containing data for about 75 radionuclides. VARSKIN can include from one to five radionuclides in a calculation.

DEPTH DOSE CALCULATIONS

VARSKIN was used to calculate the depth dose profiles for a number of beta emitters residing on the skin in several different contamination configurations. For each radionuclide of interest, a series of VARSKIN calculations was run for a point source. Each of the calculations assumed a different depth, ranging from 0.007 to 0.15 cm. For each VARSKIN run, smeared area doses were ignored; only doses to points on the dose surface were used. The distribution of doses along the points at a given depth were recorded and plotted. A three-dimensional plot was then drawn up with a curve for each of the critical depths.

After the series of depth dose profiles was calculated for the point source, another series of calculations was performed for a disc source with radius 0.1 cm, and a third series of calculations was performed for a disc source with radius 0.5 cm.

The depth dose profiles for point- and disc-source skin contaminations composed of $^{90}Sr/^{90}Y$ are presented in Figures 1 and 2. The vertical axes of these graphs represent dose. There is no numbering on the vertical axis because the actual dose values depend on the source strength of the contamination, and the depth dose profiles will have the same relative shapes no matter what the absolute values. The y-axis corresponds to depth in the skin; a "picture" on the back wall of the graph indicates the approximate depths of the various regions of the skin. The x-axis of the graph corresponds to positions parallel to the surface of the skin.

CONCLUSIONS

An examination of the graphs presented here indicates that beta doses fall off rapidly with depth in the skin. This effect, of course, is a direct result of the limited range of betas in skin. The point source profiles are much more sharply peaked than the disc source profiles, and the dose values fall off much more sharply with depth. This effect reflects a near

$1/r^2$ dependency for a point source. The area-averaged dose to the skin would be far smaller than the maximum point dose value shown on the graph. The peak value falls off rapidly with increasing depth: at 0.050 cm, the maximum dose is only 1.3% of the maximum dose at 0.007 cm; at 0.1 cm, the maximum dose is 0.2% of the 0.007-cm maximum; and at 0.15 cm the maximum value is less than 0.1% of the 0.007-cm maximum.

The disc source profiles fall off much more gradually with depth. For the 0.5-cm disc source, the peak value at a 0.05 cm depth is 34% of the 0.007-cm maximum; at 0.1 cm, the maximum value is 19%; and at 0.15 cm the maximum value is 13% of the 0.007-cm maximum.

These depth dose profiles illustrate the dramatic difference between disc and point sources for producing doses at different skin depths. In evaluating an actual case of skin contamination, the health physicist must be careful to accurately assess the nature of contamination on the skin surface. The issue of area-averaging doses also plays a crucial role in arriving at a dose: the dose to a small point will be far higher than the dose averaged over a larger area.

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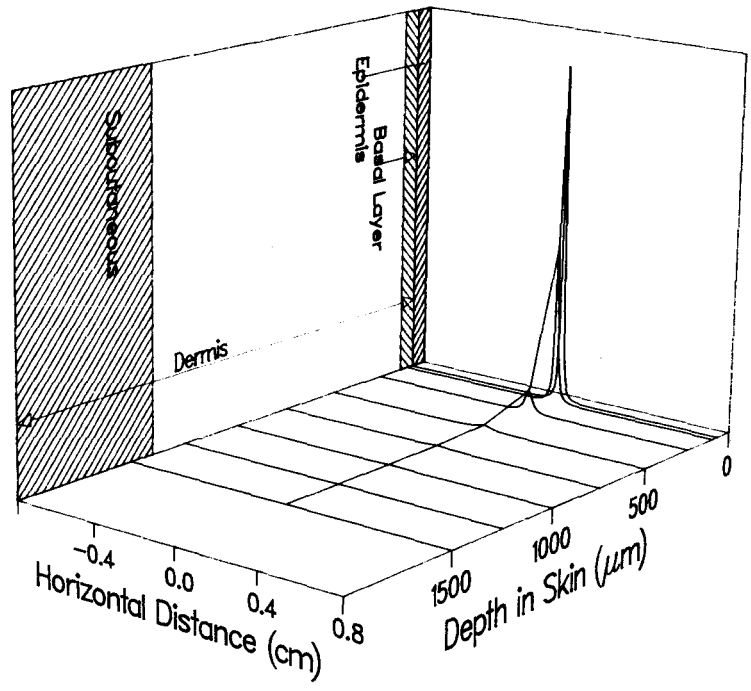

FIGURE 1. Dose Profiles in Skin $^{90}Sr/^{90}Y$ Point Source

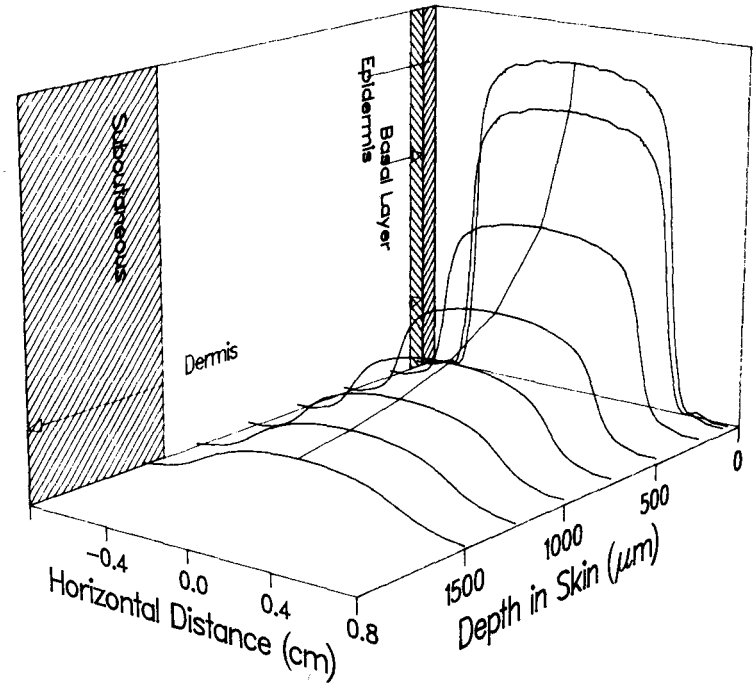

FIGURE 2. Dose Profiles in Skin $^{90}Sr/^{90}Y$ 0.5-cm Disc Source

RECOMBINATION CHAMBER AS A DEVICE FOR DIRECT DETERMINATION OF AMBIENT DOSE EQUIVALENT OF MIXED RADIATION

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ABSTRACT

The recombination chamber is a LET-dependent detector based on the columnar recombination phenomenon in high-pressure TE-gas. It is shown that response of the recombination chamber about two litres in volume and with walls 1g/cm² in effective thickness is proportional to the ambient dose equivalent of mixed radiation, practically of any composition and of any energy spectrum.

ACTIVITIES AT THE SWEDISH NATIONAL INSTITUTE OF RADIATION PROTECTION (NIRP) FOR THE CONTROL OF HAZARDS FROM NON IONIZING RADIATION

Enn Kivisäkk
The Swedish National Institute of Radiation Protection

INTRODUCTION

In 1976, the Swedish radiation protection legislation was amended to give NIRP the task to design and carry out a programme for the control of hazards from non ionizing radiation.

Initially optical radiation, laser and UV, was given priority in that programme because of the rapidly increasing use of optical sources for various purposes. A number of studies of emission characteristics, especially of UV sources, used for medical and cosmetical treatments, were carried out (1,2,3 and 4). In later years, the programme has become more and more focused on electromagnetic radiation and fields in the radiofrequency range, especially at very low frequencies.

OPTICAL RADIATION

Regulations concerning laser equipment calling for type-testing of all devices prior to marketing in Sweden have been issued by the NIRP (5). The testing, which is carried out by the National Testing Institute located in Borås, includes a classification procedure and an inspection of the technical protective devices. The technical substance in these regulations is in close correspondence with that of the IEC Publication 825.

NIRP has also issued regulations concerning work with lasers with the objective to protect the general public against harmful laser radiation (6). The regulations contain, among other things, rules for the set-up and operation of laser light shows.

One well-known result of the activities concerning optical radiation at NIRP is the set of sun-lamp regulations issued in 1982 (7). In these regulations limiting values are given for radiation at the treatment distance.

The source of optical radiation which causes by far the greatest number of detrimental effects is however the sun. Induction of various kinds of skin cancers by solar UV-radiation is one such effect which is causing health authorities in Sweden and also in the other Nordic countries growing concern. The rapid annual increase of cases of in particular malignant melanomas is now acknowledged as a serious threat to the health of the general public.

In 1960 there occured 3.2 cases of malignant melanomas per 100 000 in Sweden, in 1980 the corresponding number was 11.5. These incidence numbers are age adjusted to 1970. The annual incidence has thus more than tripled over these two decades and the annual increase is now 6% indicating a doubling time of approximately 12 years.

NIRP has during the latest years carried out several studies of solar UV intensities in Sweden and in popular vacation resorts such as the Canary Islands (8,9) and also funded such studies at external reseach laboratories (10). The material thus obtained is intended to be used in information campaigns with the objective to teach people sensible sun habits, which are now planned by public health authorities on central as well as local levels of the society.

LOW FREQUENCY ELECTROMAGNETICAL FIELDS

In the early eighties alarming reports about adverse pregnancy outcomes and birth defects among female VDU (Video Display Units) operators reached Sweden from the United States and Canada. These reports were published in the Swedish news media and caused a lot of concern at the Swedish office workingplaces, which at that time were in the beginning of the computerization process.

Because in many of the reports radiation was suspected to be the causal factor, an extensive study of emitted electromagnetic radiation and fields of various kinds from such equipment was carried out by L-E Paulsson et al at the NIRP laboratories for non ionizing radiation (11). The report from this study includes measurements of about 45 different makes of VDU's. The measurements continued however after the report was completed - now usually on commission from manufacturers. At the conference "Work with display units 86" held in Stockholm in the spring of 1986 Paulsson reported statistics from measurements of 147 different types of VDU's (12).

Figure 2 from (12) illustrates the distribution of measured magnetic induction (dB/dt) at the distance of 30 cm in front of the unit. The values in the figure are peak to peak values of the dominating component of the field, which usually is vertical. The lowest value recorded among these 147 types was 4 and the highest 345 mT/s i.e. a ratio of almost 100.

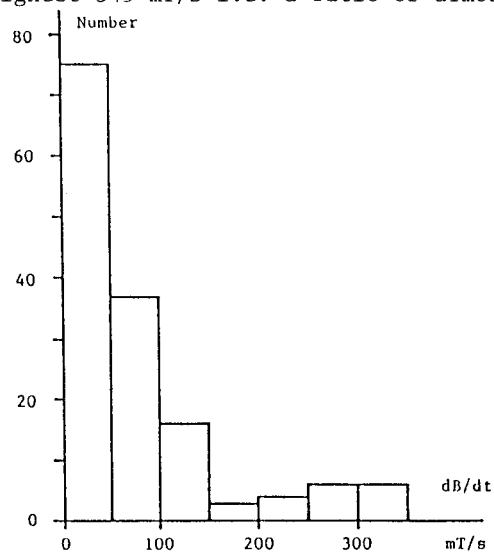

Figure 2. Statistical distribution of magnetic induction for 147 types of VDT's.

The measurements also showed that there was a trend towards higher magnetic field intensities caused by the desire for better visual ergonomics, which can be obtained for instance by using positive image polarity. Such demands lead to higher horizontal frequencies, usually resulting in high values of the magnetic fields. The measurements however also showed that this is not necessarily always the case. Units with high horizontal frequencies - in the interval 30 - 65 kHz - could also be found with low values of the magnetic fields.

The observations made during this series of measurements enabled NIRP to state low levels of emitted magnetical and electrical fields, which are technically achievable without impairing good performance from other aspects such as visual comfort and ergonomics (13).

Recently a voluntary testing system for video display terminals including keyboards has been established in Sweden on an initative from the Swedish goverment. The Swedish National Council for Metrology and Testing was given the task - in collaboration with several other national authorities and organisations, among them NIRP - to elaborate methods for testing of VDU's and keyboards and also accreditate laboratories for the voluntary testing of such equipment. The NIRP laboratories recieved accreditation in June 1987, and are so far the only accredited laboratories in the country.

The testing programme (14) is extensive and covers 45 different properties of the display unit and keyboard:

VDU		Keyboard	
Visual ergonomic properties	19	Ergonomic properties	4
Emission properties	7	Physical design	5
Physical design	6	Other properties	4

In a guide which will be published by the National Council recommended values etc. are given for the properties included in the test programme. For the emission properties these recommended values are based on the abovementioned report from the NIRP (13):

Static electricity: The equivalent surface potential should be within the the interval +/- 500 V.
Electrostatic lead off in the keyboard - resistance less than 100 Megaohms
Magnetic fields: dB/dt below 25 mT/s at a distance of 0.5 m
B " 100 " " "
No magnetic emission during longer intermissions
The magnetic field intensities are measured in the frequency range 1 kHz to 400 kHz.

IRP has initiated and finances several studies at external reseach laboratories, in order to clarify possible relationships between exposure to low frequency electromagnetic fields and biological effects. In two succesive studies the possible teratogenic effect of pulsed magnetic fields of similar kind as emitted by VDU's has been examined in pregnant mice (15,16). The studies have however yielded conflicting results.

In the first study, conducted by B. Tribukait at the Karolinska Institute in Stockholm, a slight overrepresentation of malformations was

found in the exposed foetuses. The study was however to small to allow for definite conclusions. In the second one, which was designed to extend the earlier experiment and thus included greater number of animals, no increase in the number of malformations was found - not even among dead foetuses. However the number of placental resorptions and of dead foetuses were higher in the exposed females. This had not been the case in the first studie.

An explanation for the different results in these experiments may be the fact that different strains of mice were used which may differ in their sensitivity for the applied electromagnetic fields. Therefore further experiments in order to clarify this are planned by NIRP.

NIRP has also initiated experiments in order to clarify possible ralation-ships between exposure to electrostatic fields from VDU's and skin disorders. At the time of writing (Nov 1987) no definite results are available, but so far no obvious, simple correlation has been found.

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SAFETY OF DIAGNOSTIC ULTRASOUND

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ABSTRACT

Various national and international organisations have adopted the patient exposure limits originally proposed by the AUIM, i.e. 100 mW/cm^2 and 50 J/cm^2. The review will cover

* these exposure limits in the light of present knowledge,
* techniques of exposure measurement,
* assessment of present patient exposure.

The conclusions will include

* present clinical recommendations for exposure limitation,
* present aims of ultrasound bioeffects research.

EFFECTS OF 2450 MHz MICROWAVE RADIATION ON MEIOSIS AND REPRODUCTION IN MALE MICE

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ABSTRACT

The paper presents a series of studies to examine effects of continuous wave 2450 MHz radiation on meiosis and on chromosomes of germ cells in male CBA/CAY or ICR mice, by means of the spermatocyte (SCT), heritable translocation (HTT) and dominant lethal (DLT) tests(1). Animals were exposed in an environmentally controlled waveguide system during two consecutive weeks, 30 min daily, six days a week. Specific absorption rates (SAR) were used in the range from 0.05 to 20 W/kg. With the SCT, it was demonstrated that chromosomal translocations can be induced by exposure during the first meiotic prophase, particularly during initial and early pachytene stages. The HTT results demonstrated that balanced translocations may be recovered among offspring of exposed males. The DLT provided confirmatory data on effects during prophase and indicated that chromosomal damage may be also induced by exposure of spermatids, during the maturation stage, and of spermatozoa. No changes were observed in spermatogonia. Thus, the effects of exposure were limited to one spermatogenic cycle. Genetically significant effects were induced at an SAR of 2 W/kg in the testes. For comparison, an SAR of 0.4 W/kg is used commonly as a basis for occupational exposure limits.

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RADIOFREQUENCY RADIATION
SAFE WORKING PRACTICES IN THE ROYAL AUSTRALIAN AIR FORCE

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The Royal Australian Air Force (RAAF) has long recognised the value of its workforce and the need to preserve their health and wellbeing to achieve operational objectives.

The Directorate of Air Force Safety (DAFS) is required by the Chief of the Air Staff to take all measures possible to prevent accidents and incidents in the RAAF, under the provisions of the Defence Instruction, 'Air Force Safety and Occupational Health Policy'. Consequently, the RAAF has exercised a pragmatic approach to radiofrequency radiation (RFR) and has always adopted and implemented strict exposure standards. DAFS receives technical advice on RFR from the Directorate of Telecommunications Engineering (DTELENG) and on occupational health from the Directorate General of Air Force Health Services (DGAFHS).

BEFORE 1981

Microwave radiation safety procedures were well established in the RAAF by the early 1960s. The RAAF accepted as a safe standard, that humans should not be exposed to microwave radiation at an average power density of greater than $10mW/cm^2$. An average power density of between $1mW/cm^2$ and $10mW/cm^2$ was considered safe for incidental or occasional exposures, but a power density of less than $1mW/cm^2$ was required for indefinitely prolonged exposures. Personnel were warned by Air Force Orders of the possible dangers to human tissue from exposure to microwave radiation and to avoid prolonged exposure to the primary beam of radars at less than 200 feet (about 60 metres).

Where aircraft radars were tested on the ground, operators were required to ensure that antennas were directed to open space and hazard areas delineated by warning signs, flags, and rope barriers. Some maintenance workshops illuminated flashing red lights when radars were tested, which restricted personnel from entering proscribed RFR hazard areas.

A broad publicity and training programme, which included publishing specific warnings in Base Routine Orders, ensured all personnel, not just technicians, were well aware of the possible hazards of microwave radiation exposures.

The effects of RFR energy on munitions and fuels were also stressed and under specific Defence Instructions, RAAF technicians

maintaining radars were also monitored for coincidental exposure to ionizing radiation from high voltage tubes, a practice which continues today.

In the mid-1960s, the Air Force purchased a set of broadband isotropic microwave radiation monitors for use by their Specialist Reserve consultants to establish compliance with the standard of the day. The need for each RAAF Base to have a set of monitoring equipment to check exposure situations that arose mainly in maintenance workshops and deployable communications units was recognised in the mid-1970s and 14 sets of monitors were distributed. Unfortunately, these meters were not robust and within a few years, over half of the sets were unserviceable.

POST 1981

In 1981, DGAFHS commenced a comprehensive survey of all RFR emitting equipment on RAAF Bases in response to a heightened general awareness of RFR hazards. These surveys were performed by one of us (K.H.J.) and all recommendations resultant from it were consistent with the proposed limits of the new Australian RFR exposure Standard being developed at the time. This development of a new Australian standard was being performed by a committee of the Standards Association of Australia, of which one of us (K.H.J.) was a member.

This new exposure standard, which became Australian Standard AS2772-1985 (Ref.1), was fully subscribed to and adopted by the RAAF as its operative exposure standard on the recommendation of the authors.

The occupational exposure limits contained in AS2772-1985 are illustrated in Figure 1. Note, however, that these limits refer to values averaged over any one minute period. For periods of exposure of less than 30 minutes, the Standard permits the limits specified to be increased by a factor of 30/t, up to a maximum factor of 5, where 't' is time in minutes. Notwithstanding this relaxation, in an area where severe RF shocks and burns exist, the limited period exposure level is not to exceed $10mW/cm^2$. In an area where severe RF shocks and burns have been eliminated, the limited period exposure must not exceed $100mW/cm^2$.

In addition, the peak pulse exposure level is limited to $1000mW/cm^2$, which is a very stringent requirement for users of radar systems, particularly the military. A detailed discussion of the implications and implementation of AS2772-1985 can be found in Ref. 2.

Australian Standard, AS2772-1985, is more restrictive and more complicated to administer than the Air Force standard it replaced. Consequently, a meeting of interested RAAF parties was convened in early 1986 to determine an implementation policy. The outcome of this meeting was as follows:

Initially, radiation hazard distances for both ground and

airborne equipment would be determined by calculation and published in Special Technical Instructions (STIs) for application by units.

Where these radiation hazard distances created operational problems, immediate on-site surveys would be carried out in order to clarify requirements.

The data gathered from the comprehensive Base survey programme would be used to verify the calculated radiation hazard distances.

All radiation hazard distances would eventually be promulgated in technical equipment and ground maintenance instructions contained in Defence Instructions (Air Force) Australian Air Publications (DIs(AF)AAPs).

A five year on-going Base survey programme would be established and administered by DGAFHS.

A Ground Safety Instruction (GSI) would be issued by DAFS, detailing Base survey requirements, local monitoring requirements, and instructions for deploying aircraft and transportable units.

Guidelines for continuation training and awareness programmes by Base Ground Safety Officers, which were to include detailed instruction of RFR hazards, and a requirement to regularly promulgate radiation hazard information in Base Routine Orders, were to be included in the GSI.

BASE SURVEYS AND COMPLIANCE

Of the fifteen major Air Force Bases, seven have been comprehensively surveyed, as well as two HF transmitting stations and the bulk of RAAF airborne equipment. On the basis of these surveys, a number of engineering controls have been instituted to achieve compliance with AS2772-1985.

These controls have included the fitment of interlocks to ground based radars to stop transmission in the event of a loss of drive to the antenna or when remotely changing the bearing of a precision approach radar reflector from one runway to another. Screened viewing windows have been fitted in the cabinets of high powered radar transmitters to obviate the need to open the equipment if arcing is suspected. RFR absorptive screens have also been provided for some testing procedures.

All Bases are now to be re-equipped with new broadband isotropic RF monitors. These monitors will be placed under the control of Base Environmental Health personnel, who will perform required RFR surveys, after being trained in their use.

Safe working practices have been instituted to achieve compliance with the Standard including the requirement for all RFR generators tested in the laboratory to be connected to a dummy load, or if this is not possible, the use of RFR absorbers. Similarly, antennas are required to be orientated to a safe bearing before transmission. The testing of aircraft communications inside hangars has been prohibited, as has maintenance on open, energised, wave guides. RFR hazard warning signs are required to be placed at appropriate distances from equipment under test and Base medical officers may, at their discretion, perform health surveillance on exposed individuals.

CONCLUSION

The RAAF has adopted the new Australian RFR exposure standard and instituted a comprehensive and co-ordinated set of procedures to implement that Standard, in an inter-disciplinary approach, involving scientific, health, safety and engineering professionals. On the basis of the comprehensive surveys performed, and the implementation of the strategies detailed in this paper, it can be concluded that RAAF personnel are not routinely exposed to RFR in excess of the allowable limits of AS2772-1985.

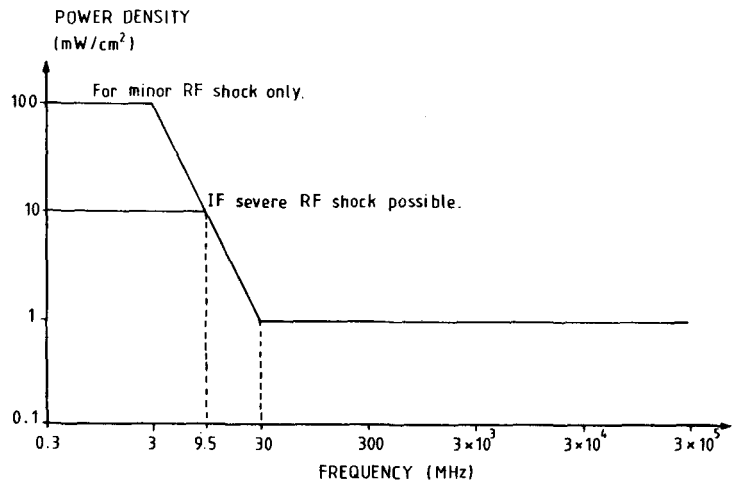

FIG. 1 EXPOSURE LIMITS - OCCUPATIONAL.

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EFFECTS OF RF LOW LEVELS ELECTROMAGNETIC FIELDS ON PARAMECIUM PRIMAURELIA*

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INTRODUCTION

In the last years many studies have been performed to examine biological effects of prolonged exposure at electric field low levels (1-2).
This great interest is linked to a specific interaction possibility, also related to the exposure length, between electromagnetic fields and biological systems without remarkable enhancement of organism's temperature.
Hence the need to investigate in vitro the possible cellular regulation mechanisms involved in these interactions, varying physical exposure parameters.

For these esperiments we choose ciliata protozoa Paramecium primaurelia as biological system.This is an interesting experimental model because:
a) share certain membrane properties with nerve and muscle cells such as electrical excitability, presence of a Ca++ transport system, exocytotic ability and mobility.
b) its division rate can be easily directly determined.
c) there are several mutant types involving alterations in ion permeability and Ca++ transport.

In a first moment the influence of some physical parameters at constant frequency of 27.12 MHz (at various power density)has been studied on growth rate;the growth rate is able to give a comprehensive description of the cell system situation.
At present we are trying to correlate the found effects with cytoskeleton structural modifications using immunofluorescence techniques.

MATERIALS AND METHODS

Cells of the wild type strains of Paramecium primaurelia were grown at 25°C according to the usual procedure (3) in Cerophyl inoculated the day before with Klebsiella aerogenes.
For the experiments 24 depression plates with a single cell in each depression have been used. Plates had been exposed at 0 (sham-exposed), 1, 10, 100 W/m2 CW and square wave modulated at 72 Hz in a TEM cell. Modulation at 72 Hz has been choosen by many Authors (4)

* This work was supported by Regione Piemonte - Grant n° 144 for Health Research and by the National Institute of Nuclear Physics - Section of Torino.

since it is similar to that employed in clinical applications to enhance bone healing.

The TEM cell was connected to a signal generator and an amplifier through the 24 hours. The instrumental chain composed by a signal generator, an amplitude modulator , an amplifier, a bidirectional coupler, power sensors and a power meter allows the generation inside TEM cell of an electromagnetic field at a frequency of 27.12 MHz with an accuracy of $\pm$ 0.7 dB (5).
The whole system is connected to a computer that assures constant conditions of electric field strength.

By daily isolation before the exposure we had obtained isogenic lines used for every couple of plates: one for control group and one for the exposed one.

The TEM cell had been maintained at 25°C $\pm$ 0.2°C by a hot air flow set by a feed-back system and placed in a climatized room.
Temperatures of exposed cultures and of control groups were measured by an optical fiber probe connected to a fluoroptic thermometer interfaced with a computer.
The 0.4°C range has not influence on growth division rate (4).
The cell cycle of Paramecium primaurelia takes nearly 8 hours.

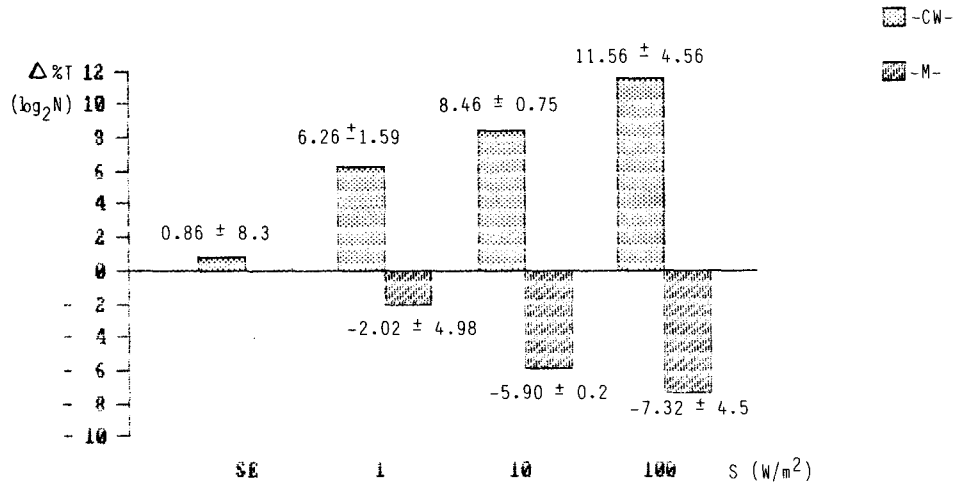
Cell number was determined by a microscopic examination. The daily division rates was calculated by taking the $\log_2$ of the number of cells produced in each depression by a single cell after 24 hours. Those depressions where we had the death of the cell, or we have no cell division (may be cell in autogamy) were not taken into account.
We choose the 24 hours period because after this time the average number of paramecia in every depression is optimal for counting and statystical analysis.
The experimental unit is the growth rate averaged on the set of 24 depressions in a plate.
Statistical analysis of the results was by Mann-Whitney's U test to compare means of growth rates between plates exposed to the same power density to the correlated isogenic control plates.

RESULTS

The advancement of percent variation of growth rate versus power density for CW and modulated exposures is shown in Fig.1.
Tab. 1 reports the numerical consistency of experimental groups and the statistical significance of the different tests.

We can observe that the percent variation of growth rate raises with increase power density as regards CW fields and decreases as regards modulated fields.

Fig. 1 - Growth rate percent variation (Δ % T) of Paramecium primaurelia versus power density in air, under 27.12 MHz continuous wave (CW) and square wave modulated (M) conditions. As regards square wave modulated fields, power density value regards to the "on" phase.



Tab. 1

	SE	1 CW	1 M	10 CW	10 M	100 CW	100 M	W/m²
N. plates	6	10	10	6	6	10	10	
N. isolated Paramecia	144	240	240	144	144	240	240	
N. counted Paramecia	1152	1972	1872	1195	1036	2280	1680	
Mann Whitney	N.S.	N.S.-5%	N.S.-5%	1-5%	5%	1-5%	1-5%	(N.S.= Not Significant)

DISCUSSION AND CONCLUSIONS

The first data emerging from analysis of above results, is the opposite pattern of growth rate variation depending on CW or modulated exposure.

The pattern suggests that the wave shape of the electromagnetic field is more important with regard to induction of effects on the cell than the amount of the energy carried by the wave. Concerning the electromagnetic field strength to which Paramecia are exposed , it must be underlined that the true value of power density in culture medium is about 6400 times lower than in air.

Previous studies show the influence of electromagnetic fields on Ca++ ions influx inside various type of cells (6), and in particular the Paramecium (7).

There's evidence for the involvement of Ca++ ion in regulation of a lot of different cellular mechanisms. In particular it influences polymerization of tubulin, polymer playing an important part in cytoskeletal organization of the cell (8).

A first hypothesis of work concerning biophysical interpretation of obtained results, is that variations of growth rates is connected

to an influence of electromagnetic fields on mitotic spindle microtubules organization, mediated by Ca++ ion concentration variation.

To verify this hypothesis we are setting up a series of immunofluorescence experiments with the use of monoclonal antibodies anti-α and anti-ß tubulin conjugated with FITC, in order to detect the presence of some morphological alterations of Paramecia at mitotic stage depending on incident electromagnetic field features (9). In this way we think to have more useful data for the understanding of the opposite effects of growth rate of CW fields and of modulated fields.

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EXPOSURE TO RADIOFREQUENCY (RF) IN INDUSTRY. EXAMPLES OF MEASUREMENT AND IMPROVEMENT.

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Introduction

During the last 20 years in Italy there has been a progressive increase in industrial use of radiofrequency equipment for heating materials, thus creating serious problems in terms of health-risk assessment for the workers involved and in terms of setting up procedures and standards for prevention (2,7,8,9,12). It has recently been estimated (1985) that in Italy there are about 8000 apparatuses with approximately 12000 exposed workers mainly in the plastic, wood and metal-working industries (9). In the Occupational Health Institute of Milan a specific group has been dealing with RF apparatuses in industry since 1980 with a view towards occupational exposure, equipment improvement and medical surveillance. This paper reports the measurement results so far, some of the most meaningful equipment improvements, and considerations about the way and materials to be used when designing RF equipment improvement.

Methods

Measurement of electric (E) and magnetic (H) field strengths was performed with an Aeritalia instrument model TE307 equipped with isotropic balanced sensors and a model TE308 optical fibre remote repeater(1). The survey on every piece of equipment was devided into three phases:

1) study of the technical features of the machine and of the characteristics of the working environment;
2) analysis of the tasks and exposure times;
3) measurement of radiation levels at the work station and comparison of results with international standards and the proposed Italian standards (1982) for maximum values of electromagnetic fields at the work place (4); when needed, the measurements were repeated after the improvement.

Results

The survey results of the Occupational Health Institute of Milan are shown in Tab.1. Electromagnetic field average values, standard deviations and ranges are reported with reference to the work place for both metal-engeneering and plastic/wood industries. Table 2 illustrates the results of an improvement on a metal-working apparatus. Table 3 reports electric and magnetic field strengths for an unshielded machine and for a shielded one with the same main characteristics in the plastic industry.

Discussion

From Table 1 it can be seen that workers' exposure varies greatly depending on type of application, device power, electrode shape, source-worker distance and, mostly, on whether or not some proper shieldings have been implemented; these results are compatible with those of other authors (5,6,11). Table 2 and 3 show that, even where electromagnetic fields are very high, a successful improvement can be carried out.

Result analysis of the improvement measures confirms, in agreement with other authors (3, 10) that the factors influencing the shielding efficacy are:

1) the metal shielding material used: at low frequencies, $f < 10$ MHz, stainless steel or copper is recommended;
2) the shielding surface: the greater the surface, the greater the efficacy of the screening device; if the shield is made out of wire netting, it should be kept in mind that the larger the mesh the lower the efficacy of the screening device;
3) the thickness of the shield: efficacy increases with the thickness of the shield;
4) distance between the shield and the source: the shield must be installed as far as possible from the source;
5) grounding of the shields and metallic surfaces that can transmit the electromagnetic field at a distance must be separate and of suitable size according to the frequency of use.

Finally it has to be pointed out that:

a) generally, the material needed for the improvements is cheap and easy to use even on equipment that is already installed;
b) when suitably designed and implemented, the improvements permit the electromagnetic field to be kept within the most stringent limits required by international regulations.

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TAB. 1 - Technical characteristics and exposure values at the workplace. 137 RF apparatuses.

TECHNICAL CHARACTERISTICS		METAL-WORKING ENGINEERING	PLASTIC and WOOD INDUSTRIES
Number of apparatuses		80	57
Frequency range (MHz)		0,3 - 2	13,6 - 29
Power range (kW)		1,5 - 600	0,5 - 25
Electric field (V/m)	average value	38	103
	standard deviation	157	192
	range	2 - 2150	2 - 850
Magnetic field (A/m)	average value	1,96	0,18
	standard deviation	3,47	0,64
	range	0,02 - 33	0,02 - 1

TAB. 2 - Metal engineering industry: horizontal metal tempering. Electric and magnetic field values before and after improvement (f = 300 kHz, P = 18 kW)

MEASUREMENT POSITION	INITIAL VALUES		VALUES AFTER IMPROVEMENT	
	E (V/m)	H (A/m)	E (V/m)	H (A/m)
Eyes	2150	33	36	1,5
Trunk	860	14,5	31	0,7
Gonads	430	5,9	46	0,5
Transit area	130	0,3	9	0,05
Adjoining work stations	25	0,05	1	0,05

TAB. 3 - Plastic industry: welding of plastic materials. Values of electromagnetic field for non-shielded and shielded machineries (f = 27 MHz; P = 12-18 kW)

MEASUREMENT POSITION	NON-SHIELDED MACHINERY		SHIELDED MACHINERY	
	E (V/m)	H (A/m)	E (V/m)	H (A/m)
Eyes	800	0,30	42	0,18
Trunk-gonads	600-700	0,60	38	0,32
120 cm from electrodes	220	0,30	20-22	0,02
200 cm from electrodes	75	0,20	8	< 0,02

METALLIC IMPLANTS AND EXPOSURE TO RADIOFREQUENCY RADIATION

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There is increasing use of radiofrequency radiation (RFR) in industry for communications, welding, security, radio, medicine, navigation etc. It has been recognised for some years that RFR may interact with cardiac pacemakers and steps have been taken to prevent this interference. It is less well recognised that other metallic implants may also act as antennas in an RFR field and possibly cause adverse health effects by heating local tissues. There are a large and increasing number of implants having metal components which may be found in RFR workers. These implants include artificial joints, rods and plates used in orthopaedics, rings in heart valves, wires in sutures, bionic ears, subcutaneous infusion systems and (external) transdermal drug delivery patches[1]. The physician concerned with job placement of such persons requires information on the likelihood of an implant interacting with RFR so as to impair health. The following outlines the approach developed in Telecom Australia, beginning with the general principles and then presenting a discussion of a specific example.

PRINCIPLES

The following approach is used to assess the possible health hazard to staff with metallic implants who are exposed to RFR.

1. Information is obtained on the type of implant, its anatomical location, geometry and type of metal used. The implant is then classified as either a rod, ring, plate, or solid, or combination thereof.

2. Information is obtained on the frequencies, field strengths and likely duration of exposure.

3. The interaction between the incident fields and the implant is then modelled using two approaches:
(i) Where the maximum dimension of the implant is equal to one half of a wavelength of the RFR in tissue. Here the implant is behaving as a resonant object which is considered to be the worst case and a detailed analysis is necessary.
(ii) Where the maximum dimension of the implant is less than one-quarter of a wavelength of the RFR in tissue. Here the implant is considered to be electrically short and simplifying quasi-static approximations can be used.

Rates of energy deposition in surrounding tissues are then calculated.

4. The effect of any excess heat in tissues surrounding the implant is then considered with due consideration to cooling efficiency and sensitivity of that tissue. The interaction must not cause more than a $1^{o}C$ temperature rise.

5. Medical advice is then given to the individual and to management on what precautions, if any, are to be taken.

CASE EXAMPLE

The above points are illustrated in the case of a 32 year old male radio technician who has undergone cardiac surgery and still has the wire sutures securing the sternum.

1. The chest X-ray and clinical notes revealed 1mm stainless steel wires forming loops and near-vertical twists of wire some 2mm in diameter (Fig. 1). These structures were classed as both rods (the wire twists) and rings.

2. The subject is predominantly exposed to frequencies between 6 and 22 MHz at very low levels ($<2mW/cm^2$) but the possibility exists of exposure to the allowable limit of $25mW/cm^2$ for up to 8 hours with provision for a short term (<6 minutes) exposure limit (STEL) of up to $100mW/cm^2$ (AS2772-1985[2], Fig. 2). In future he may be required to work at frequencies above 30MHz where the exposure limit is $1mW/cm^2$ for up to 8 hours with provision for a STEL of $5mW/cm^2$.

3. The interaction between the incident fields and the implants was assessed by the following two methods and subsequent tissue temperature calculated.
(i) Although the implants shown in Fig. 1 are a combination of wires and loops the approach was to first model the wire twist and then investigate the effect of adding the loop. This approach is justified because vertical electric (E)-fields produce the strongest absorption in the body and the wire twists are so oriented as to maximize pick up; conversely the loops are so oriented as to minimize pick up. In treating the case of the wire twists acting as half wave resonant objects in a complex medium, use was made of a mini-numerical electromagnetics code (MININEC) which is a method of moments analysis of thin wire antennas in free space[3]. The actual computer code used was a very recent and improved version written in PASCAL by Dr A.W. Davis*. The code was altered to accept a complex medium and currents calculated in the wire twists at resonance assuming an incident plane wave. These results compared favourably with

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others[4]. The effect of the loop on the induced currents in the wire twists was also investigated. The maximum E-field enhancement at a distance of half a wire radius from the tip of the implant was then calculated. The final two steps in arriving at a temperature rise included, (a) calculation of the fraction of the incident E-field that reaches the implant assuming a planar tissue geometry, and (b) calculation of the heat flow assuming a small cylindrical tissue volume centred on the tip of the implant. The temperature rises are listed in Table 1 for the resonant range of the implants of 1650 to 3000MHz corresponding to wire twist lengths of 12 and 7mm respectively. At the STEL the tissue temperature rise exceeds $1^{o}C$.

(ii) The quasi-static approximation consisted of modelling the wire twists as prolate spheroids (the body obtained by rotating an ellipse about its major axis) and calculating the enhancement in the E-field at the tip of the implant[5]. That fraction of the incident E-field reaching the implant was found from calculations of the internal E-field distribution in whole body block models of man[6]. Note that the maximum induced E-field occurs at 80MHz where the whole body is acting as a resonant object. The resultant temperature rises at frequencies of 3, 9.5 and 80MHz are also listed in Table 1. The frequencies of 3 and 9.5MHz were chosen for calculation because the exposure limits reach maximum values of 100 and $10mW/cm^2$ respectively (Fig. 2). At the STEL the tissue temperature rise approaches $1^{o}C$.

Table 1 Tissue temperature rises at the exposure limits.

Frequency (MHz)	Exposure Limits (mW/cm2)	Temperature Rise (deg. C)
1650-3000	1	0.27
1650-3000	5 (STEL)	1.4
80	1	0.15
80	5 (STEL)	0.77
9.5	10	0.011
3	100	0.0045

4. The implants are embedded in cartilaginous bone which is in contact with skin tissue capable of readily dissipating heat. Although some of the wire loops protrude inside the rib cage the most critical parts were found to be the tips of the wire twists which are external to the rib cage and not in contact with overtly sensitive tissue.

5. The advice given to the subject and management was that work can proceed at exposure levels up to the occupational limits for frequencies below 30MHz. However, no STEL work is permitted particularly at frequencies between 50-150MHz and in the gigahertz range.

Future work will include the evaluation of plates, particularly those in the skull, and solids such as hip replacements. In addition experimental verification of the calculations is a priority.

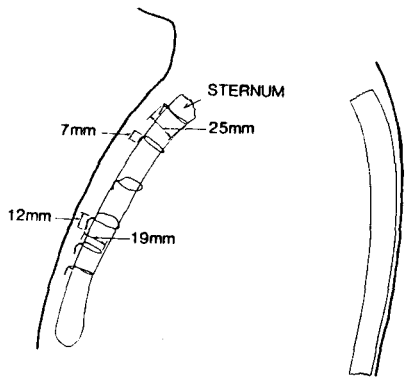

Figure 1: Schematic of sternum from the side showing wire implants.

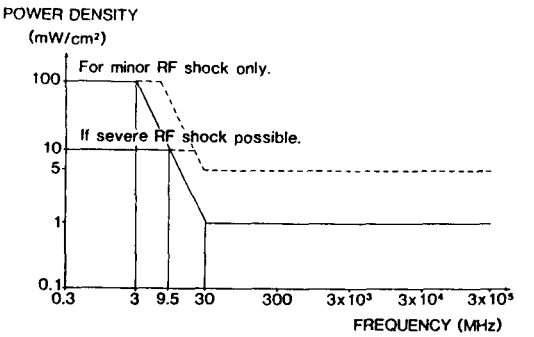

Figure 2: Australian exposure limits - occupational. The dashed lines represent the maximum short term exposure limits.

ACKNOWLEDGEMENTS

The authors are greatly indebted to Dr A.W. Davis for supplying a copy of his improved MININEC computer code. The manuscript was typed by Ms Christine Anderton and the artwork produced by Mr Ian Rielly. The permission of the Chief General Manager, Telecom Australia to publish this paper is hereby acknowledged.

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RF FIELDS AT BROADCAST TRANSMITTING SITES

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1. INTRODUCTION

The purpose of broadcasting is to radiate information over large areas to randomly located receivers which may be mobile and use inefficient antennas. Quite high powers must therefore be radiated to provide reliable services, giving rise to RF fields which can exceed the Maximum Exposure Limits (MEL) given in AS 2772-1985 when close to the transmitting antennas.

Measurements commenced in the early 60's have been expanded over the past 2 years and moved towards measuring all national broadcasting sites (over 400) and services (over 700). Field levels measured are summarised below.

2. TRANSMITTERS AND ANCILLIARIES

2.1 Modern transmitters are generally well sealed with very few instances of levels exceeding even the non-occupational (24 hour) limits. Levels of 2 or 3 times the 24 hour limits may occur near adjusting knobs and meters, but can usually be reduced by simple means.

2.2 Older MF and HF transmitters use discrete components rather than sealed assemblies and often include viewing facilities for the high power stages. They are also larger with more cabinets to be bonded together and interconnected, giving rise to attendant problems of RF leakage. Some are open at the top, and have open wire transmission lines passing over areas readily accessible to staff.

Levels as high as 2.5 $(A/m)^2$ magnetic field and 5×10^6 $(V/m)^2$ electric field have been recorded. However significant effects due to pick up in instrument leads below 1 MHz (found later) suggest that the E fields were in fact considerably lower - probably by an order of magnitude. Fitting open steel mesh (say 5 cm grid) dramatically reduced these fields to below 0.005 $(A/m)^2$ and 2×10^3 $(V/m)^2$ in the highest field locations - near the coils and valves of high power stages (to 50 kW) and below transmission lines.

3. VHF/UHF

All transmitters in the National Service in Australia (TV started in 1956) are well sealed, though fields up to 10^4 $(V/m)^2$ can be found near the ends of open circuit pyrotenax used in older RF loads.

3. ANTENNA SYSTEMS

3.1 At MF E fields picked up by instrument leads can cause measurement errors, which are worst close to antennas due to the high and fairly homogeneous fields and to the existence of a radial field - whereby leads cannot be in a lower field than the probe head nor can they be readily oriented for minimum pick-up.

Proximity effects are also significant, with the presence of the measurers body altering the field. Recent series of tests using a variety of methods, combined with detail calculations by Dr K. Joyner of Telecom Research Laboratories, show lead effects as high as 30 times (on a power density or $(V/m)^2$ basis) and body proximity effects up to 5 times. Measurements indicate that fields in the vicinity of a mast having a 240m earth mat of radial copper wires over poor earth and carrying dual 50 kW services (77^o and 96^o electrical height) are:-

25 m from mast	E- 250 V/m	H- 0.05 $(A/m)^2$
150 m from mast	20 V/m	below 0.01 $(A/m)^2$
240 m from mast	9 V/m	

Transmitting antennas are also efficient receiving antennas. With 10 kW fed to short top-loaded standby antennas some 270 m away high fields exist near the base of the main mast. When a series of short circuits are applied (to the inputs of the antenna coupling units (ACU's) and across the mast base) the E field is reduced to 450 $(V/m)^2$ and the H field rises to 0.1 $(A/m)^2$ at 250 mm from the base insulator.

Much can however depend on the antenna current distribution. Short circuit locations may need to be chosen to alter the positions of current and voltage nodes to bring both E and H fields below AS 2772 limits at the base. As MF transmission lines are often electrically short the variation achievable can be limited, and detuning of ACU's may be necessary.

Other large structures, such as mains power supply towers and their earth wires, can act as effective receivers and raise fields in their vicinity. A doubling of fields has been recorded.

There can also be very high fields in the vicinity of open ACU's. Fields measured have exceeded $10^7 (V/m)^2$ and 100 $(A/m)^2$. Most coupling units are however enclosed/ shielded. At stations of 100W or lower transmitter power all measured fields are below MEL except within 200 - 500 mm from open ACU's and the base of short antennas.

3.2 At HF many of the older curtain antennas use short circuit and open circuit sections of balanced open wire transmission line as tuning stubs. With 50 kW and 100 kW transmitters fields of the order of 10^5 $(V/m)^2$ and 3 $(A/m)^2$ are measured 0.5m from such tuning stubs with fields being very dependent on position due to the standing waves.

Matched sections of transmission line are naturally associated with more constant fields, typically:-

POWER	DISTANCE FROM LINE	E FIELD	H FIELD
10 kW	1 m	10^4	0.1
50/100 kW	1 m	2×10^4	0.2
	2 m	10^3	0.01
250 kW	1 m	5×10^5	3
	2 m	10^5	0.4
	4 m	10^4	0.02

Matching of newer high powered antennas is by use of appropriate lengths of transmission lines of different impedances. On these tuning sections fields of some 5 times the above have been measured.

HF antennas rely on ground reflection of fields to achieve their designed angle of radiation, giving a region of higher fields on the ground. For the type of curtain antennas typically used for HF broadcasting the highest ground level fields occur some 15 m to 25 m in front of the antennas :-

10 kW	200 $(V/m)^2$	0.004 $(A/m)^2$
100 kW	3×10^3 $(V/m)^2$	0.04 $(A/m)^2$
250 kW	10^4 $(V/m)^2$	0.1 $(A/m)^2$

Where adjacent antennas are active the highest fields occur in front of the common support and may approach twice these levels.

5m behind the screen of an active antenna (and away from transmission lines) fields are :-

250 kW	5×10^3 $(V/m)^2$	0.01 $(A/m)^2$

3.3 VHF/UHF

These frequency bands are considered together as their high gain antennas are commonly mounted on the same support structures. Powers are typically 100 kW effective radiated power (e.r.p.) for VHF TV to 300 kW e.r.p. for UHF TV.

Some antennas/support structures have no levels above the MEL to within about 1 m of active antenna panels. There can however be great variations in level which are not readily predictable, but appear to be largely due to resonances occurring in structural members at these frequencies. It is not unusual to encounter localised fields, either E or H, of some 2 or 3 times MEL at unexpected locations in the antenna column, even in locations remote from the antennas. For example ladder guides at times appear to act as waveguides at UHF, while ladder sections may transmit vertically polarised VHF considerable distances along a tower.

Localised Fields as high as 10^6 $(V/m)^2$ and 2 $(A/m)^2$ have been measured, though E field maxima are more normally 1 or 2 orders of magnitude lower. Most of these "hot spots" are very small - eg some 10 or 20 cm across. Even with larger areas (eg a 2 m gap between a platform and a ladder guard near a vertically polarised VHF antenna) the fields drop very rapidly with distance to below MEL 20 or 30 cm from the high field region. Fortunately small reductions in transmitter power (3 or 6 dB) generally lead to large reductions in these spurious fields. Where levels above MEL cover extensive areas (due to antenna back radiation) reductions in fields are quite closely related to transmitter power reductions.

On low powered stations of 100 W erp or below, a field has marginally exceeded the 24 hour (non-occupational) exposure limit in only two situations.

4. SUMMARY

Despite the high powers needed for broadcast services the transmitters and ancilliaries only give rise to low electromagnetic fields.

Antennas are fundamentally associated with high fields. Further, dimensions of some support structures tend to give rise to resonant "hot spots" of high fields (though high E and high H fields rarely occur at the same locations). This necessitates careful measurement of even quite low powered services to define conditions whereby fields can be reduced below MEL's when work in antenna areas is necessary.

ACKNOWLEDGEMENT

The permission of the Director, Broadcasting, for presentation of this material is gratefully acknowledged.

LEVELS OF 50Hz MAGNETIC FIELDS IN THE HOME AND OFFICE ENVIRONMENT IN AUSTRALIA

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The subject of exposure to 50/60Hz magnetic fields and the possible adverse health effects amongst exposed groups has received an enormous amount of publicity recently. Although a number of studies have been published in the U.S.A. concerning magnetic field exposures in the home and environment[1,2], no such data has been published for Australian conditions.

The aim of this study was to quantify the 50Hz magnetic field levels in the home and office from domestic appliances, office equipment and electrical wiring.

SURVEY EQUIPMENT AND RESULTS

The three survey instruments used were commercial devices marketed under the tradename "Gauss Maus" and manufactured by the Arlunya Division of the Dindima Group, P.O. Box 106, Vermont, Victoria, 3133. These instruments are handheld portable devices comprising a measuring unit linked to a polarization sensitive sensor unit by a spring coil cable. The usual operation is to hold the measuring unit in one hand and the sensor unit in the other which can be moved around to measure magnitude and distribution of the magnetic flux density. They are designed to measure peak levels of extremely low frequency magnetic fields up to a maximum of 99mG (9.9μT). They operate in either a wideband mode (12Hz to 1.3kHz, 3dB points) or in a bandpass mode (50Hz $\pm$ 6Hz, 3dB points). However, only the results taken in the bandpass mode are reported here. The calibration of one of the three instruments was checked by measuring the magnetic flux density at a known distance from a wire carrying a known current. The survey results are summarized in Tables 1, 2 and 3.

OBSERVATIONS AND DISCUSSION

The mean ambient levels of magnetic fields found in various rooms of the average suburban house varied between 0.5 and 1.3mG.

However, the maximum values quoted in Table 1 for the front door, lounge, master bedroom and bedrooms 1 and 2 were measured in the same house. This house was located close to a 22kV distribution line with bedrooms 1 and 2 approximately 5 metres from the power line. It was determfined that the external wiring was the source of the fields in those bedrooms.

The magnetic fields associated with distribution lines depend on the current loading on the system and were found to vary by a factor of 2 to 3 over a 24 hour period. In addition, the magnetic fields within a house were found to vary by up to a factor of 2 depending on the usage of domestic appliances. The majority of levels presented in Table 1 were measured in the early evening when appliance usage and loading on the distribution system were close to maximum.

Another source of magnetic fields was found to arise from the unbalanced current flow between a house and the external distribution system resulting from multiple earth neutral (MEN) systems which are mandatory in Australia. With such a system not all the current that flows into the house returns back to the local transformer via the neutral line; an amount of current flows to ground because the neutral is also earthed at the house via an earth stake or through the plumbing as well as at the local transformer. The amount of current flowing in the earth was calculated from a measurement of the associated magnetic field and found to be 0.025 times the household current. It must be noted that this figure would be very dependent on soil conductivity.

Table 1. Levels of 50Hz magnetic fields (mG, rms) measured in houses.

Room	Minimum	Maximum	Mean(A)
Front door	0.2	3.7	1.2
Family room	0.4	2.3	1.0 (8)
Lounge room	0.3	3.2	1.0
Dining room	0.3	1.3	0.5 (11)
Hallway	0.4	1.1/5.3(B)	0.6 (6)
Master bedroom	0.1	4.5	1.0
Bedroom 1	0.1	7.0	1.2
Bedroom 2	0.1	7.1	1.3 (9)
Bedroom 3	0.4	0.5	0.5 (2)
Kitchen	0.4	3.5	0.9
Laundry	0.4	0.4	0.4 (6)

Notes: (A) The mean levels were based on measurements in 14 houses, except where the number of measurements appears in parenthesis.

(B) The level of 5.3mG was measured when the hot water storage unit located in the roof was heating. This measurement was not included in the calculation of the mean.

For a loop of unbalanced current as shown in Figure 1 the maximum magnetic flux density, B, at the distance indicated and in a direction perpendicular to the plane of the loop was calculated according to Weber[3] and found to be 0.55mG for every Ampere of current flow in the MEN system.

Therefore, an ambient level of 0.55mG could be produced if a total household current of 40A were drawn and 1A (0.025x40)

flowed through the MEN system. Certainly this situation is not going to exist throughout the entire 24 hour period and it would thus appear that external wiring as well as appliance usage were the main sources of ambient levels in a house, particularly when that house is located close to power lines.

Table 2. Levels of 50Hz magnetic fields (mG, rms) measured 30cm from domestic appliances.

Appliance	Individual Measurements		
Electric blanket	0.8	1.3	1.8
Hair dryer	2.1	3.5	7.8
Toaster	4.2	7.1	8.5
Electric kettle	1.4	1.8	2.6
Frypan	2.3	2.5	2.8
Electric Fan	0.4	0.7	0.7
Colour TV (screen)	3.5	6.0	7.1
Fluorescent fixtures	1.5	1.9	3.4
Electric ranges (front)	7.8	8.5	11
Ovens (front)	1.4	1.8	2.3
Refrigerators (front)	0.6	0.7	0.8
Irons	0.8	1.4	1.6
Vacuum cleaners	18	23	33
Washing Machines (front)	2.1	2.5	2.8
Drills and saws	7.1	11	13

Table 3. Levels of 50Hz magnetic fields (mG, rms) measured in offices and at various distances from office equipment.

Position/Equipment	Minimum	Maximum	Mean
Centre of office	0.1	0.5	0.3 (12)
Computer terminals (30cm)	0.8	3.6	2.0 (10)
Photocopiers (10cm)	0.7	20	7.9 (4)
Telex machine (30cm)			0.8 (1)
Letter punch (30cm)			8.5 (1)
Electric typewriter (30cm)	0.3	4.7	1.9 (4)

Note: The values in parentheses are the number of measurements.

Use of vacuum cleaners, electric ranges, drills and saws do expose persons to signficantly higher levels of magnetic fields for periods of time approaching several hours per day. However, the fields close to electric blankets (<15cm) exceeded 99mG (peak) and of the appliances tested these provide the highest and most prolonged exposure for persons who use them.

It was not possible due to the limited dynamic range of the measuring instrument to determine with any confidence the fall off with distance of the magnetic fields from the appliances measured in this survey. However, from a detailed inspection of the work of Gauger[2] it can be concluded that, in general, for distances greater than 30cm, the fall off in magnetic field follows an inverse cube law with distance.

The levels of magnetic fields measured in offices appear to be lower than those found in the home. However, the levels from office equipment are comparable with those from domestic appliances.

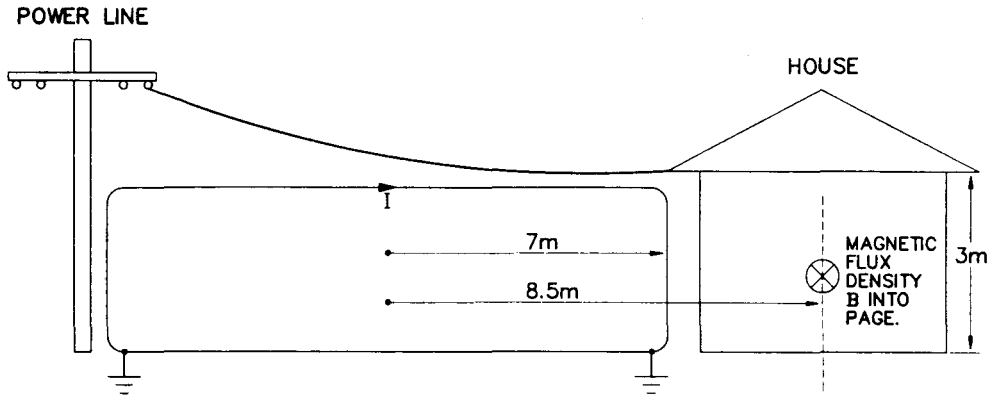

FIGURE 1. SCHEMATIC REPRESENTATION OF THE CURRENT LOOP FORMED FROM THE UNBALANCED CURRENT FLOWING IN THE MULTIPLE EARTH NEUTRAL SYSTEM.

ACKNOWLEDGEMENTS

The authors wish to thank Mr Peter Wallace, Mr Ron Owen and Mr Tanh Dovan of the Victorian State Electricity Commission and Mr Ian Macfarlane of the Telecom Australia Research Laboratories for helpful discussions.

The permission of the Chief Health Officer, Victorian Health Department and the Director Research, Telecom Australia to publish this work is hereby acknowledged.

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EFFECTS OF 2450 MHz EXPOSURE ON HUMAN LYMPHOCYTE TRANSFORMATION IN VITRO

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ABSTRACT

There are contradictory reports in the literature on the effects of microwave exposure on transformation of lymphocytes. To resolve this question, human lymphocytes have been exposed in vitro to 2450 MHz radiation under carefully controlled conditions at various power levels and for periods up to 120 hours. Lymphocyte tranformation in exposed samples was compared to control samples, sham-exposed or incubated in a conventional tissue culture incubator. Heparinized peripheral blood samples were obtained by venipuncture from healthy volunteers, lymphocytes were separated by gradient centrifugation (Lymphocyte Separation Medium, Litton Bionetics) and incubated in chromosome medium 1A without phytohemagglutinin (Gibco) at a standard concentration of 10^6 cells/ml medium. Using the automated dosimetry microwave exposure system described in a companion paper (Joyner et al., this Congress) the temperature profile over time was recorded for each sample, and average and peak specific absorption rates (SAR), specific absorption (SA), and thermal dosage (TD) were calculated. At harvest, cells were counted and their viability was tested using neutral red, Janus green, and trypan blue staining. Standardized cytological preparation were made using a "Cytospin II" (Shandon) cytocentrifuge, air-dried, fixed in methanol and stained with a combination of Wright and Giemsa stains. The percentages of untransformed (small) lymphocytes, intermediate and lymphoblastoid cells were determined, based on morphological features, quantitated using an "Optomax" image analyzer. The correlation of lymphocyte transformation with SAR, SA, and TD was examined. The implications of the results for a "thermal" vs "nonthermal" mechanism of interaction will be discussed.

CONCEPT AND VALIDATION STUDIES OF THE REAL-TIME REACTOR-ACCIDENT CONSEQUENCES ASSESSMENT MODEL "ECOSYS"

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ABSTRACT

The Chernobyl accident has demonstrated the urgent need for computer programs for real-time assessment of potential radiological consequences of major reactor accidents and for timely recommendations of useful and cost-efficent counter measures. During the past decade the dynamic radioecological program "ECOSYS" has been developed by us for nuclear accident consequence assessment with high resolution in space, time and exposure pathways. The Chernobyl reactor accident leading to relatively high contamination of Southern Germany provided excellent conditions for realistic validation studies of concept, sub-models and parameters of ECOSYS. To this purpose more than 7000 low level and in-situ gamma spectroscopy measurements were performed to study experimentally the behaviour of radionuclides in foodchains and in the urban environment and to compare the results to theoretical predictions of ECOSYS. The results show good agreement in the contamination levels of important food stuffs and in external exposure dose rates from a given surface contamination. Improvements were necessary in the assumptions regarding the food consumption habits which changed considerably - and in the functions describing the weathering off from urban and plant surfaces.

The results of this validation study and the concept of the improved computerized model, which has subsequently been converted into a real-time code, will be discussed in detail.

L'ACCIDENT DE TCHERNOBYL; DIFFICULTES DE LA COMPARAISON DES MODELES ET DES MESURES

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L'accident de Tchernobyl a conduit les différents Etats à évaluer les doses que les populations ont été susceptibles de recevoir. Dans une première phase ces évaluations nécessitent la mise en oeuvre de modèles, dont le but est de prévoir les conséquences de l'accident, essentiellement à court terme. Dans une seconde étape, les mesures confirment les prévisions des modèles, et servent à estimer les conséquences à plus long terme.
Il a semblé pertinent d'entreprendre une comparaison des différentes techniques de calcul et de mesure qui ont été utilisées dans les différents pays de la Communauté, afin de mettre en évidence les difficultés de comparaison des estimations ou des évaluations qui ont été faites.

I - LES MODELES

Les modèles utilisés peuvent être classés en trois grands groupes, qui s'enchaînent en séquences:

- Les modèles "météorologiques" ont pour but de définir les trajectoires suivies par le polluant. Ils nécéssitent la prise en compte de données locales (intensité de la source, hauteur effective du rejet, vitesse et direction du vent, gradient thermique vertical...), et de données synoptiques (champs de pression ou champs de vent)
- Les modèles "de concentration" évaluent la concentration le long des trajectoires précedemment définies, et calculent le champ de concentrations au sol ainsi engendré, en superposant un modèle de diffusion
- Les modèles "de dose" évaluent les équivalents de dose reçus par les populations. Ils nécessitent la prise en compte de vitesses de dépôt, de facteurs de transfert et de facteurs dosimétriques.

I-1 comparaison des trajectoires

Lorsque l'on superpose les trajectoires estimées par différents modèles, on constate que les divergences peuvent être très grandes: Si les trajectoires suivies par les particules émises le 27 Avril à 0 heures, et estimées par différentes équipes sont tout à fait similaires (figure 1), il n'en n'est pas de même de celles émises le 26 Avril à 12 heures (figure 2). Les données météorologiques de bases étant en principe identiques (radiosondage de Kiev, Organisation Météorologique Mondiale), on peut logiquement attribuer ces différences aux modèles eux-mêmes: discrétisation de la source, pas de temps du calcul, prise en compte du vent vertical, extrapolations dans le temps et dans l'espace des données... L'observation de ces divergences nous a conduit a rechercher si, sur l'ensemble de la période du rejet (une dizaine de jours), il existe un type de modèle dont les résultats sont significativement meilleurs que les autres. En toute rigueur,ceci relève d'un exercice d'intercomparaison qui sortirait du cadre de cette étude, et nous avons dû nous contenter d'une comparaison beaucoup plus sommaire: Nous avons sélectionné 13 sites répartis sur l'ensemble de l'Europe, et pour lesquels nous disposons de mesures systématiques, et nous avons regardé si les résultats de ces mesures étaient compatibles avec les trajectoires données par les modèles. La conclusion de ce travail est que tous les modèles ont un pourcentage de bons résultats compris entre 75 et 85% ce qui, compte tenu de l'empirisme de la méthode suivie, ne permet pas de

conclure à la supériorité systématique d'un modèle sur les autres.

I-2 La diffusion

La comparaison des différents modèles de diffusion à partir d'une même trajectoire a fait l'objet de nombreux travaux antérieurs /2/,/3/. Ces travaux ont montré que à grande distance du point de rejet, les résultats ne diffèrent pas de plus d'un facteur 2, ce qui est tout à fait acceptable compte tenu des autres sources d'incertitude.

I-3 L'évaluation des dépôts

Dans tous les modèles, le dépôt sec est évalué classiquement à l'aide de vitesses de dépôt sec, tandis que le dépôt par la pluie nécessite l'introduction d'un coefficient de lavage (Royaume-Uni, Italie) ou d'une vitesse apparente de dépôt (France). Les variations de ces paramètres d'un modèle à l'autre conduisent à des évaluations du dépôt qui peuvent varier d'un facteur 3.

I-4 Le calcul des doses

Les facteurs de dose permettant de relier les quantités ingérées ou inhalées aux équivalents de dose reçus sont définis dans la Publication 30 de la CIPR pour ce qui concerne les adultes. Pour ce qui est des enfants, des divergences pouvant atteindre un facteur 10 (ingestion de césium par les nourrissons) ont été observées /4/. Il faut toutefois noter que ces incohérences ont été rapidement corrigées.

II - LES MESURES

La comparaison des mesures doit être menée avec beaucoup de prudence, car il faut s'assurer de la cohérence des conditions expérimentales. Or, de ce point de vue, il existe de nombreux obstacles à la comparaison de mesures réalisées par des laboratoires différents:

les techniques de prélèvement

L'influence de la technique de prélèvement peut être mise en évidence à l'aide de trois exemples:

Le cas de l'herbe: Frissel /5/ a montré que l'activité massique d'un échantillon peut varier d'un facteur 5 selon que l'herbe est coupée à ras du sol, ou à 5cm de la surface. En général, l'activité massique de la partie supérieure de l'herbe est supérieure à celle de la partie inférieure pour les prélèvements effectués avant le 20 Mai; cette tendance s'inverse pour les prélèvements effectués ultérieurement.

Le cas de l'iode dans l'air: Lorsque l'iode organique n'a pas fait l'objet d'un prélèvement spécifique, l'activité totale de l'iode a été faite en multipliant l'activité de l'iode associé aux aérosols par un facteur qui varie selon les auteurs entre 3 et 5.

Le cas des sols: L'épaisseur de la couche prélevée est un paramètre essentiel, notamment lorsqu'il s'agit de court terme après l'accident. Certains auteurs prélèvent des couches de 1 cm, tandis que d'autres intègrent la totalité de la contamination en prélevant une couche de 5 cm.

la représentativité de l'échantillon

La quantité prélevée varie de façon considérable d'un laboratoire à l'autre (dans le cas de l'air, de quelques dizaines à quelques milliers de m^3). Ce facteur est directement lié à la limite de détection. Dans le cas des produits alimentaires, il faut s'assurer que l'échantillon est représentatif de ce qui sera effectivement consommé: On a pu constater que des prélèvements de lait effectués dans des fermes voisines donnaient des résultats très différents, parce que les animaux n'étaient pas soumis aux mêmes régimes de stabulation, mais aussi parce que les dépôts peuvent varier très sensiblement en des points voisins.

les conditions de stockage et de transport (cas des produits volatils)

Ce paramètre n'est pas sans incidence sur le résultat de la mesure: par exemple, le transport en container réfrigéré des échantillons permet une meilleure conservation des produits volatils.

la préparation des échantillons

La finalité de ces mesures étant l'évaluation des quantités ingérées, un certain nombre d'auteurs préfèrent effectuer les mesures sur des produits lavés, tandis que d'autre s'en tiennent à des mesures sur le produit brut. Il s'en suit une apparente discordance, qui peut atteindre un facteur 10.

les techniques de mesure

La difficulté de l'interprétation des spectres obtenus est illustrée par le problème du neptunium: ^{239}Np a en effet été identifié dans les prélèvements atmosphériques par plusieurs laboratoires (en France, Suède, Finlande), tandis que d'autres estiment qu'il s'agit là d'une confusion avec ^{132}Te. Cette ambiguïté dont, à notre avis, l'origine devrait être éclaircie, met en évidence la nécessité d'une banque de données de spectrométrie commune, applicable aux mesures effectuées dans l'environnement.

De tout ceci, il ressort que la comparaison des résultats des mesures effectuées par différents laboratoires n'est pas immédiate, et nécessite une connaissance approfondie des conditions expérimentales.

III - CONCLUSION

L'accident de Tchernobyl a vu la mise en oeuvre simultanée d'un grand nombre de modèles, et a permis la comparaison des hypothèses et des valeurs des paramètres sous-jacents. A l'échelle qui nous intéresse (quelques milliers de kilomètres), le seul véritable problème est celui des trajectoires suivies par le polluant. A cet égard, les incertitudes liées au terme source ont une importance prépondérante. IL faut toutefois noter que les divergences que l'on peut observer, et qui peuvent être considérables pendant les périodes de marais barométrique, tendent à s'atténuer sous l'influence de la durée du rejet (une dizaine de jours dans notre cas).

Du point de vue des mesures, les exemples présentés dans ce document montrent qu'il est nécessaire de rechercher une harmonisation au niveau des méthodes de prélèvement, des techniques de mesure et de la présentation des résultats. Les Organisations Internationales concernées ont un rôle essentiel à jouer pour susciter la réflection commune nécessaire pour parvenir à cette meilleure harmonisation.

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Figure 1 Trajectoires calculées à l'aide de différents modèles pour une émission le 27 Avril à 0 heure

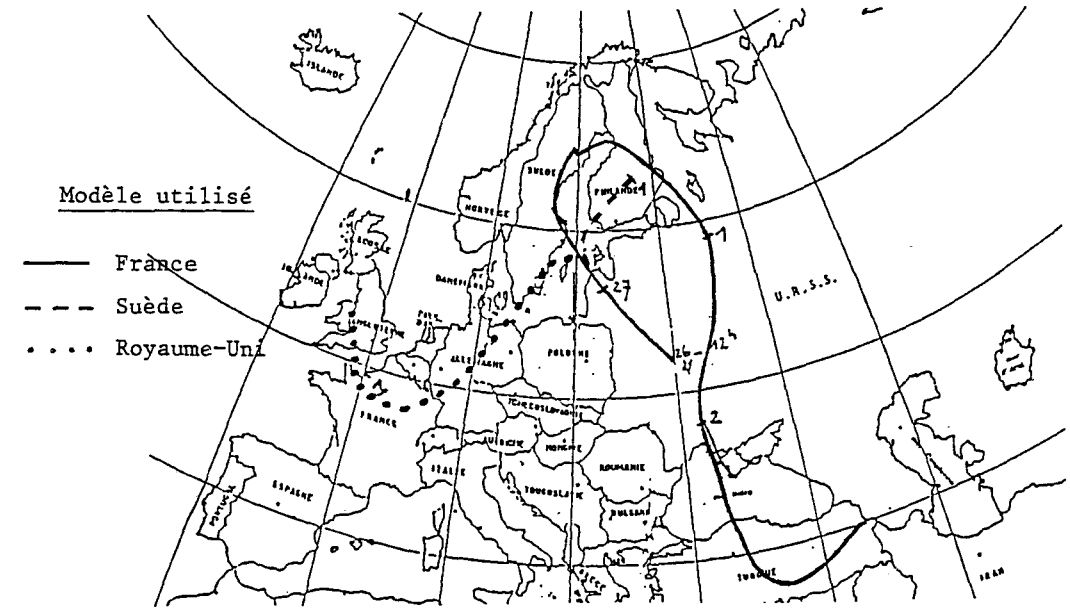

Figure 2 Trajectoires calculées à l'aide de différents modèles pour une émission le 26 Avril à 12 heures

INFLUENCE OF PRECIPITATION, BUILDING AND PLUME RISE ON THE DOSES RECEIVED BY THE POPULATION IN CASE OF ACCIDENTS IN A PHWR-POWER PLANT

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ENACE S.A.

The influence of precipitation, buildings and plume rise on the doses received by the population for some hypothetical accidents proposed for the CNA-II was analyzed. The CNA-II is a 745 MWe heavy water pressure vessel reactor, fuelled with natural uranium and installed by Paraná River (Argentina). CNA-II is under construction.

Two hypothetical accidents were analyzed: loss of coolant accident (LOCA) with containment isolation failure (occurrence probability: approx. $2.0 \times 10^{-7} y^{-1}$) and LOCA with normal operation of safety systems (probability: approx. $1.0 \times 10^{-3} y^{-1}$).

The atmospheric dilution factor was estimated according to Pasquill model because it is suitable for flat terrain like Paraná's River zone. Six weather conditions (A to F) were considered for each accident. The most probable condition is D. For ground level releases a height of 10 m was assumed. Stack height is 40 m.

The following irradiation paths were analyzed: external exposure due to cloud passage and to deposition on ground and inhalation. The ingestion path was not taken into account for the calculation of the immediate accident consequences because of the delation due to transportation and/or processing time. Countermeasures were not considered.

PRECIPITATION EFFECT

Rain produces iodine and aerosol washout with the consequent decrease of their concentration in the radioactive plume and increase of deposit on ground. The model proposed in IAEA safety series N° 57 was used. The doses were calculated under the plume-center line for weather conditions C and D because rain is considered possible only under these conditions.

The results obtained for the critical group (x = 1 km) in case of LOCA with containment isolation failure show that dose considering precipitation is 22% larger than dose without considering rain, for weather condition C and 6% for condition D. In case of LOCA with normal operation of safety systems the difference is negligible: 1%.

It must be said that aerosol and iodine releases are overestimated because their behaviour inside the containment was not considered (all the activity initially present in the containment was supposed to be released). So the expected influence of precipitation should be lower than that obtained with the above mentioned assumption. It should also be taken

into account that accident and rain probability must be multiplied so, the total probability is lower than that without considering precipitation.

BUILDING EFFECT

Buildings near the emission source produce eddies which increase the dispersion in the travel direction. The model proposed in IAEA safety series N° 57 was used. This effect has to be considered when the building height is larger than two times the release height. For accidents with release through stack this condition is not fulfilled. It must be said that there are no cities near CNA-II nuclear power plant. So the turbine building was selected because is one of the biggest.

The results show that the dose obtained considering building effect is lower than that without taking it into account. The difference increases with the stability of weather condition and begins to be noticeable for weather condition D: 13%. For weather condition E dose considering building is approx. 25% lower than dose without considering building and for the worst case (condition F) this difference is 40%. It can be concluded that calculations made without taking building effect into account are conservative, with differences only noticeable for stable weather conditions (E, F).

PLUME RISE EFFECT

Efflux velocity and buoyancy produce a rise of the real release height, thereby reducing the ground-level concentration. The model of Briggs was used to estimate plume rise value. Once this value is estimated it is added to the real release height to obtain the effective release height.

The results obtained for the critical group (x = 1 km) in case of LOCA with normal operation of the safety systems (release through stack) are the following: dose considering plume rise for weather condition D is approx. 23% from the value without taking plume rise effect into account. For weather condition E the value is approx. 7%. The largest influence of plume rise is observed for weather condition F: dose considering plume rise is 0.06% from the value without considering this effect.

The conclusion is: plume rise effect should be considered specially for stable weather conditions (E, F) when realistic values of dose to the critical group are needed, otherwise the obtained dose values would be too much overestimated.

DISTANCE INFLUENCE

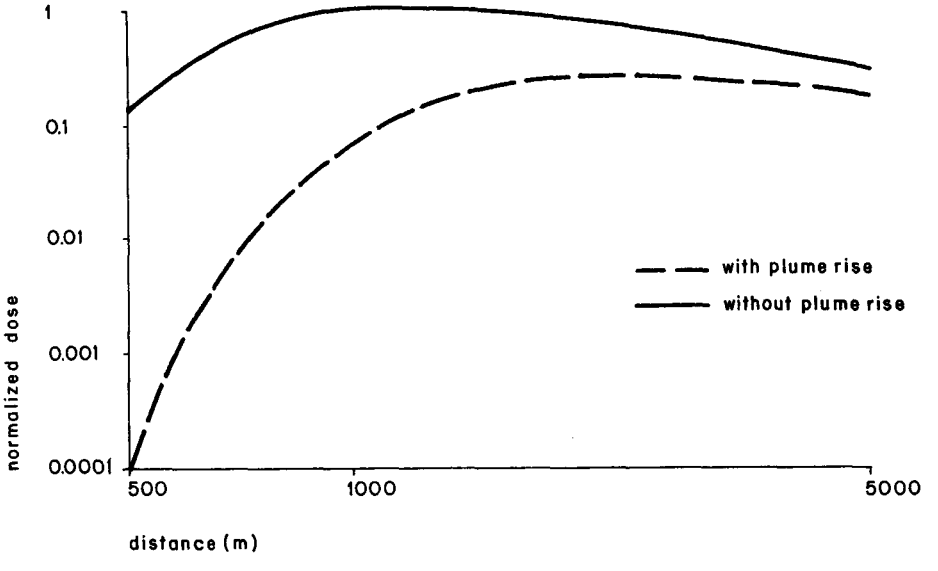
Some graphs corresponding to the analyzed accidents are included as example.

Plume rise effect is observed in Graph 1 for a LOCA with normal operation of safety systems and weather condition E. It is observed that the influence of plume rise decreases with increasing distance and is very important near the source.

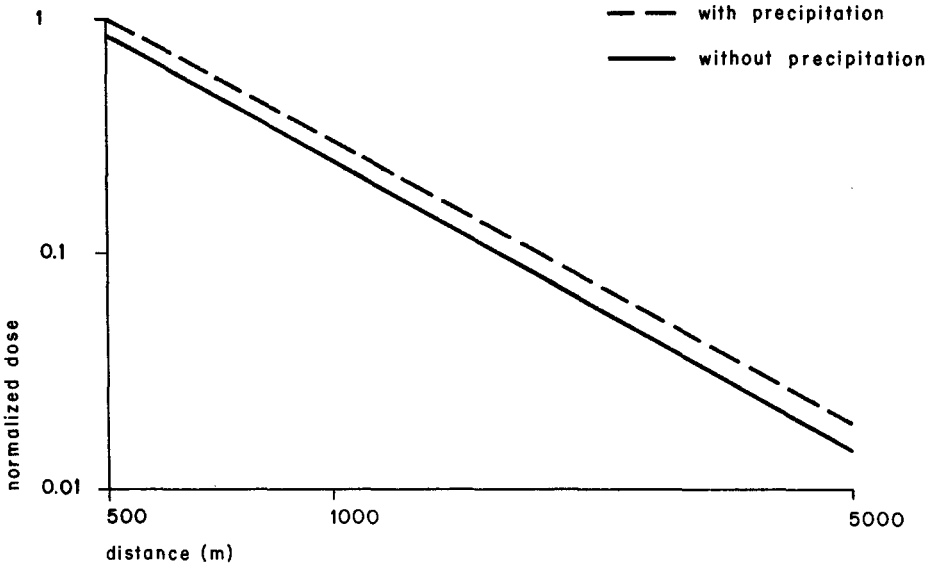
Graph 2 represents the influence of precipitation for weather condition C in case of LOCA with containment isolation failure. It is observed that the influence of precipitation is approximately constant with increasing distance.

Graph 3 represents the influence of building for the above mentioned accident and weather condition F. The influence of building decreases with increasing distance.

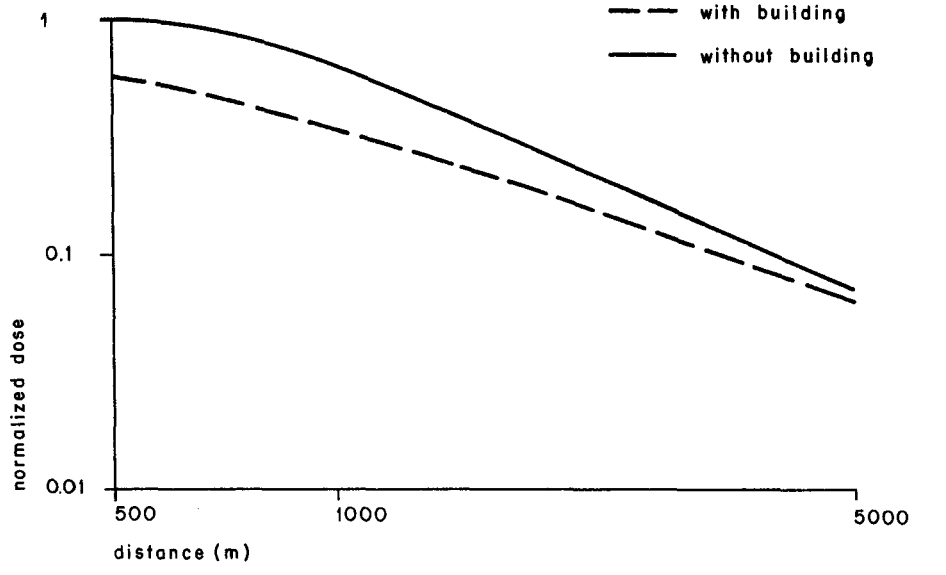
Reference: Influence of precipitations, buildings and plume rise on the doses received by the population in case of accidents in the CNA-II Power Plant. María C. Conte. TN3-002-1986. ENACE.



Graph 1 - LOCA with normal operation of safety systems. Plume rise effect. Weather condition E.



Graph 2 - Precipitation effect. Weather condition C.



Graph 3 - Building effect. Weather condition F.

DOSE ESTIMATION IN CASE OF LOCA FOLLOWED BY CORE MELT-DOWN IN THE ARGOS PHWR-380 MWe

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Argos PHWR-380 is a medium size nuclear power station (380 MWe) with a pressurized heavy water reactor of the pressure vessel type, fuelled with natural uranium or advanced fuels. It was designed by ENACE S.A. (Argentina's nuclear engineering company) and is thought to be particularly suitable for developing countries. The plant uses the technology employed in Atucha I unit (a 367 MWe NPP, which has been operated since 1974) and Atucha II plant (a 745 MWe NPP, which is under construction) but with enhanced safety features and significant upgrading.

It is ensured that the Argos complies with the demanding safety requirements of the Argentine national regulatory authority and with relevant international safety standards, guides and recommendations.

In this study it was analyzed the influence of release height and retention of radionuclides inside the containment on the consequences of a hypothetical loss of coolant accident (LOCA) followed by core melt-down.

One of the lessons learned in the field of nuclear safety is the need of ensure the confinement of radioactive material during these hypothetical accidents. For that purpose the Argos is equipped with a venting system (fig. 1) which is designed to prevent the disruption of the containment and the consequent uncontrolled release of radioactive materials into the environment that could occur in such an extreme case. If the pressure increases unexpectedly within the containment, the venting system is designed to stabilize the pressure at a safe value by regulating the release of excess gases and steam into the atmosphere through the filter system. Argos has a high free volume/power ratio. The design criterion is that the result of this hypothetical and extremely unlikely situation will be such that even the critical group of the population would not be exposed to projected doses higher than 0.1 Sv (this level would not usually justify radiological intervention or countermeasures).

To demonstrate the effectiveness of this system three different cases of release after loss of coolant accident (LOCA) followed by core melt-down were analyzed: immediate release after accident, containment overpressure failure (approx. 16 to 19 days after accident) and controlled release (the activity is released through stack approximately 12 days after accident with a flow rate of 0.5 m^3/s to keep the pressure at a safe value. It was supposed that filter system is not operative). The selected accident is a medium LOCA (0.1 A rupture). Large LOCA is excluded for the Argos reactor because the occurrence probability is several orders of magnitude lower than for an event considered improbable.

CONSEQUENCES ANALYSIS

The dispersion of the activity released into the atmosphere was estimated using Pasquill's model (flat terrain siting was supposed). The lowest wind speed possible for each weather condition was adopted. Ground level releases were supposed to occur at 10 m height. Stack height is 41 m. The critical group is located 1 km away from the NPP.

The activity inventory was estimated with ORIGEN-2 computer code, for the corresponding average fuel burnup. Release coefficients as function of temperature and behaviour of aerosol and iodine inside the containment were also considered. The latter effect was estimated using NAUA Mod. 4 computer code. Tritium inventory was supposed to be completely released. The exposure paths considered are: inhalation, ingestion and external irradiation due to cloud passage and to deposit on ground. Countermeasures were not taken into account.

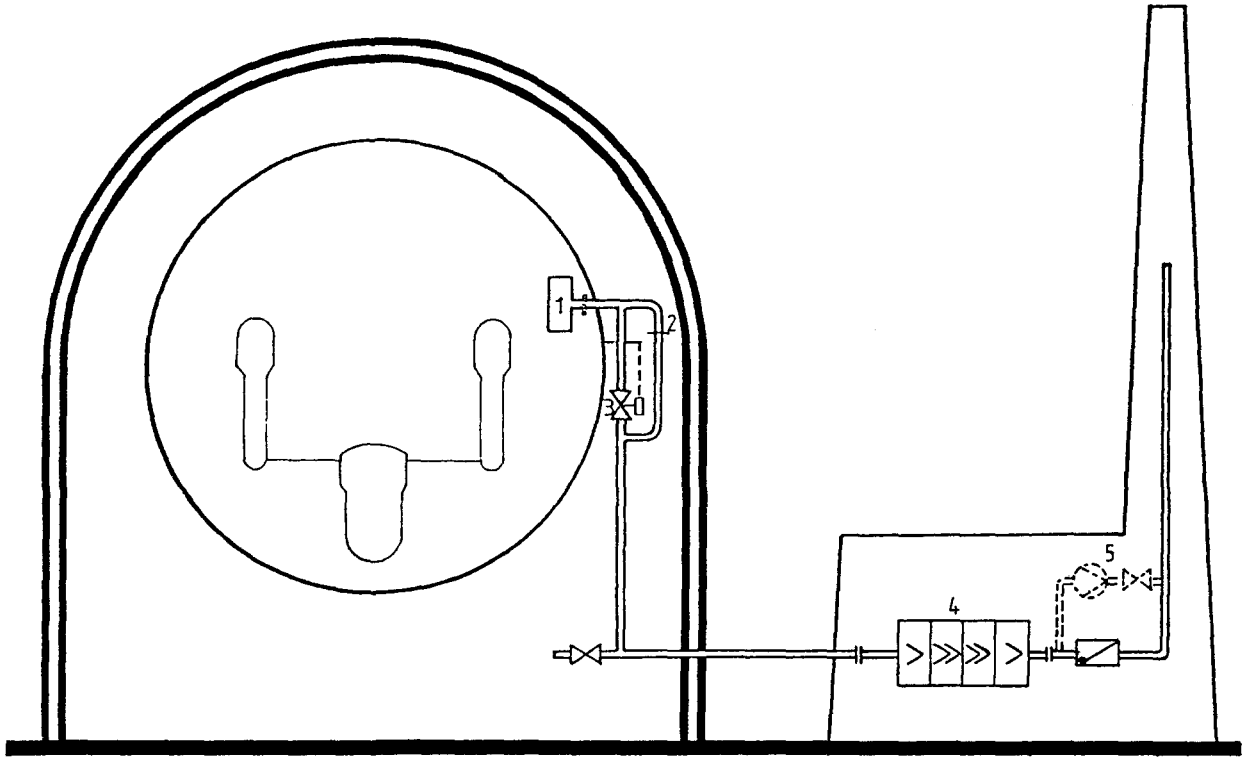
The estimated doses values for the critical group (x = 1 km) show that the influence of controlling the release increases with the stability of weather condition. The ratio between dose in case of controlled release and dose due to immediate release after accident is 0.3 for weather condition D (the most probable) and 0.002 for condition F (the most unfavorable). The ratio between dose due to controlled release and dose in case of containment overpressure failure is 0.56 for condition D and 0.04 for condition F.

It was also observed that the influence of release height increases with the stability of weather condition. For condition D the dose to the critical group in case of release through stack is approximately 40% from the value in case of ground level release. For weather condition F the value is 0.3%.

The release height, retention of iodine and aerosol inside the containment during a larger time period and the favorable free volume/power ratio are the main factors which lead to lower doses in case of controlled release: 25 times lower than in case of containment overpressure failure and 500 times lower than the value in case of immediate release.

Reference: Argos PHWR-380 "Argentina offers a 380 MWe PHWR with enhanced safety features". A.J. González et al. Nuclear Engineering International. May 1987.

Fig. 1 - Vented Containment System



1 - Drop Separator
2 - Rupture Disk
3 - Valve Actuated by Internal Pressure
4 - Filters
5 - Bypass

A COMPUTER-AID SYSTEM FOR OFF-SITE NUCLEAR EMERGENCY SITUATIONS MANAGEMENT

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INTRODUCTION

In order to better deal with nuclear accidents and the following emergency situations at its five nuclear centres, the ENEA (the italian nuclear research board) has developed a computer-aid emergency system the main feature of which is the link amoung five computers each sited at one of the nuclear centres, and a sixth devoted to control and monitor all the communications amoung them. The whole system is designed in such a way as to assure the maximum degree of reliability and a sufficient flexibility to fit the specific needs of each centre and to face with possible extensions and changes in procedures and basic data.

The system is aimed at calculating and predicting the atmospheric transport, dispersion and radiologic impact of radioactivity accidentally released by ENEA nuclear plants starting from meteorological and inventory data which are acquired in real time by the system itself. The exchange of communications, informations and data between the Centres and the ENEA H.Q. is also assured. Since the beginning the system has been mainly conceived as a tool higly useful to decision-makers during emergency situations.

It should be stressed that the system has a sufficient flexibility to allow also the analysis of the routine releases from the Centres and to provide useful data in case of emergency situations coming from both national and extra-boundaries accidents. In addition, in the near future, it will run all the radiation protection related codes under development at the ENEA, including a personnel dose record-keeping program concerning about 2000 radiation Workers.

GENERAL SYSTEM ARCHITECTURE

The whole system is mainly composed of a network system, the on-site subsystems and a coordination central system.

The first one links the various computers, each of which is located in the on-site emergency room, to the central emergency room of the ENEA H.Q. and to the computer of the control board (nuclear safety and radiation protection directorate), as shown in fig. 1.

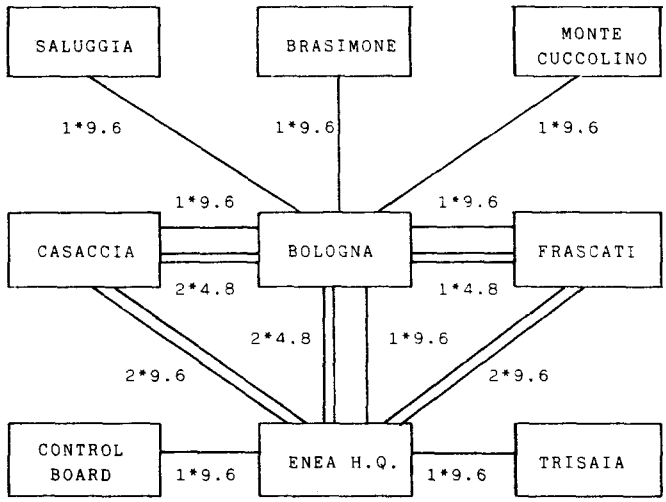

Fig. 1 - The ENEA DEC network system (kbaud).

Each on-site subsystem carries out the following tasks:

- real-time data acquisition from the meteorological tower, the stack monitors, the remote field radiation monitors, the in-plant area radiation monitors;
- data reduction, if any, validation and update;
- data formatting and preparation for their transfer;
- evaluation of the radiological situation by the simple diffusion and transport code SPADE (bidimensional, no orography);
- receiving the results of more complex codes run at either the main central computer or the control board computer.

The emergency coordination centre system has the following tasks:

- to receive the on-site radiological and meteorological data;
- to update the main data-base;
- to evaluate the radiological impact on the environment and on the population by running complex diffusion and transport codes;
- to replace the local computer, as for the accident consequences analysis, in case of the latter's failure.

HARDWARE FEATURES

The application programs and the simulation codes which run or will run on the various computers of the emergency management system need a fair large memory as they shall be managed at various

priority levels by a multi-user and a multi-tasking operating system. So 16-bit computers of the uVAX family have been chosen.

SOFTWARE FEATURES

The software is composed of indipendent modules, so that it attains a high flexibility, has a hierarchical tree based menu driven structure and works in an interactive way. Its gross structure is sketched in fig. 2 and comprises five main packages:

- on-line data acquisition;
- data filing;
- environmental data analysis;
- communication and its monitoring;
- accidental analysis.

One of its main features is that the various functions and inputs are fully automated; however, of course, request functions and manual inputs are allowed at many stages and levels.

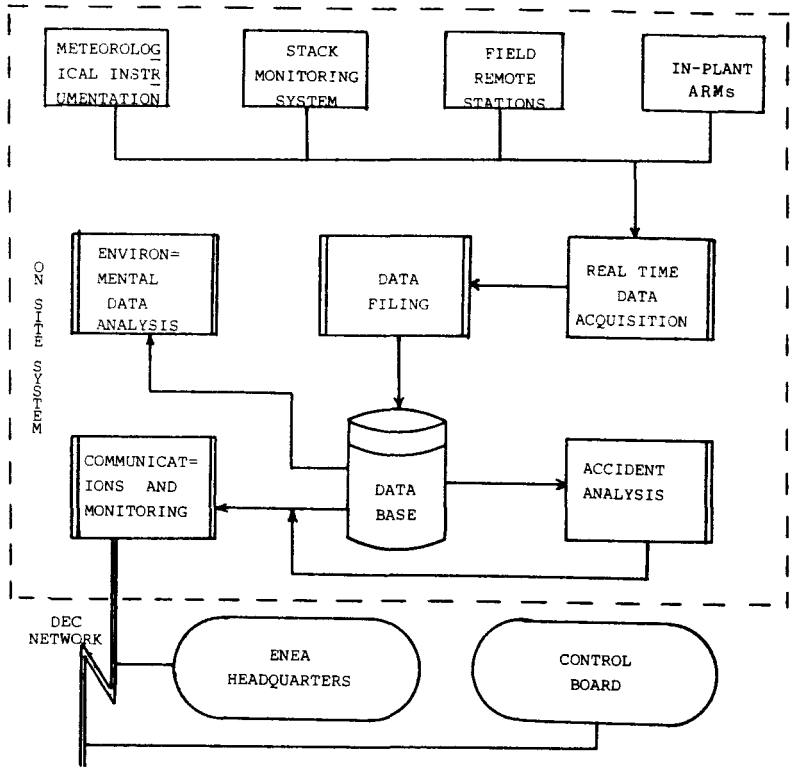

Fig. 3 - Main System Structure

The communication and monitoring module must coordinate, manage and monitor all the communication-related actions between, among and within the centres (local and coordination). Its main function is the monitored data transfer (automatic function), which comprises the environmental data reading, the data validation, the data packing and transfer and their back-up on an overlapping circular 48-hour memory. In addition request functions, such like start-up request in case of drills, test and pre-alarm conditions and requests to receive the results of codes run on the other computers, can be performed.

In addition to the above mentioned tasks, the described module performs the task of monitoring and filing all the communications, informations and functions and continuously checking the status of the lines pointing out the malfunctions, if any. These tasks are usefull not only during the emergency phase itself but also on the occasion of drills, when an "a posteriori" check is needed in order to ascertain the whole emergency machinery.

The accident analysis package is mainly based upon the SPADE code (Sequential Puff Atmospheric Diffusion Evaluation), a simple gaussian code which uses a variable trajectory advection atmospheric dispersion model and calculates dispersion factors inside a 10 km radius. This figure is based on Design Based Accident evaluations, whilst major severe accident consequences will be treated in the future.

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A PERSONAL COMPUTER PROGRAM FOR RADIOLOGICAL IMPACT EVALUATION AFTER ACCIDENTS

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INTRODUCTION:

A large accidental radionuclide release creates an emergency situation where likely off-site doses and contamination levels are required to be rapidly estimated for emergency actions.Such fast estimates can best be obtained using computers. Recent development of high capability personal computers,has opened a new avenue where with dedicated programs in BASIC language can be loaded from floppy disks. In this report details of such a program are given for an APPLE compatible system. It is not a difficult task to write the program on an IBM compatible machine.

2.0 DISPERSION MODEL:

The Pasquill-Gifford double Gaussian dispersion model is used. For ground level release, Giffords modification to include the building wake parameters has been adopted. The expressions used to compute ground level time integrated concentrations X(x,0,0) at a distance x for a release of Q Ci or Bq are:

ELEVATED RELEASE

$$X(x,0,0)=(Q/\sigma_y\sigma_z\pi u)[\exp(-h^2/2\sigma^2_z)]\ (Ci\text{-}s/m^2) \quad (1)$$

where

Q = Source strength (Curies)

u = wind speed at plume level (m/s)

h = release height (m)

σ_y, σ_z= plume spread parameters (m)

GROUND LEVEL RELEASE:

$$X(x,0,0)=Q/(\pi\sigma_y\sigma_z+CA)U \qquad (Ci\text{-}s/m^2) \qquad (2)$$

where

A = Crossectional area of the building normal to the wind and
C=A shape factor=0.5

DEPOSITION MODEL:

Contamination level $W_d(x,y)$ due to dry deposition and deposition correction factor F_d are obtained using the source depletion model viz.

$$W_d(x,y)=V_g Q_x \exp[-(y^2/2\sigma^2_y + h^2/2\sigma^2_z)] \quad Ci/m^2 \qquad (3)$$

$$Q_x = Q.F_d \qquad (4)$$

$$F_d=[\exp(\int_0^x \exp(-h^2/2\sigma^2_z)/\sigma_z dx)]^{(-\sqrt{(2/\pi)}.} \qquad (5)$$

where

Q_x= Depleted source strength (due to deposition)

Vg = Deposition velocity (m/s)

To maintain accuracy and speed of computation, the integral in Eq (5) is evaluated with respect to lnx. The integral to be evaluated is then

$$I_d = \int_1^{\ln x} (\exp(-H^2/2\sigma^2_z/x\sigma_z)d(\ln x)$$

3.0 PLUME SPREAD PARAMETERS:

Plume spread parameters for Pasquill model are normally obtained from graphs. For computational purposes however it is convenient to use the scheme given by Eimutis and Konicek (1972) where following type of relations are used

$$\sigma_y = A_y(s)\, x^{0.9031}$$

$$\sigma_z = A_z(s)\, x^{B(s)} + C_z(s)$$

where s indicates the stability (A to F)

4.0 Wind variation with height:

Wind speed used for plume rise calculations is at the stack level while wind speed for concentration calculations is at the plume level,

For ground level release however it is customary to use 10m level wind. Wind speeds at these heights are obtained from the wind speed at measurement height by using the power law wind profile expression.

$$u/u_1 = (Z/Z_1)^{n/(2-n)}$$

where u is the wind speed at height Z and the wind profile parameter n=0.2,0.25 or 0.5 according to stability being unstable,neutral and stable respectively.

5.0 PROGRAM STRUCTURE:

The program structure is divided into following steps:

(1) Read and store calculational constants and strings. Constants include

value of π, its square root etc

values of A_y, A_z, B_z, C_z,

values of profile parameter n

plume gamma doses at 29 distances from 0.1km to 100km for ground

level release and elevated release for a set of stack heights.

String variables include, wind direction N,NNE etc, stability class names A,B,C etc

(2) Interactive dialogue for input and checking of source data,metereological data, printing format, distances etc.

(3) Computation of Time integrated GLC and ground contamination due to deposition.

(4) Computation of doses: These include (i) Plume gamma doses for noble gases.(ii) Inhalation dose from Iodines(iii) Ingestion dose from deposited iodines under certain assumed food chain parameter set. Since under emergency

conditions iodine and noble gases are important the dose computation routine is confined to these only. For other isotopes, the inhalation and ingestion doses can be computed using dose factors.

(5) Printing of output: In APPLE computer, printing the rersults in the desired format is not possible directly since "PRINT USING" statement is not available. To overcome this problem two subroutines have been developed to convert a given number to a string variable and transform it into another string which can be printed as E-Format or the I-format as needed.

Subroutines are used whenever possible. Subroutines are called for computation of σ_y σ_z, concentration of ground level release, concentration for elevated release and deposition correction. Input of each parameter is through a separate subroutine to facilitate individual corrected entry. Interactive dialogue with the user is also through subroutines. Similarly, iodine dose computation and computation of gamma doses are also with subroutines.

6. APPLICATIONS

The program can be used to obtain exposures and contamination levels very fast. The time taken for obtainning values for 10 distances is about 5 minutes after the program disk is loaded. The program has instructions to ring bell whenever input information is asked. Also, PRINT channel for the printer is activated only during the INPUT/OUTPUT printing thus avoiding unneccessary printing of the interactive dialogue. Thus the program is expected to be useful not only in a radiation emergency but also for routine dispersion calculations

7. REFERENCES

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NEW DEVELOPMENTS IN THE RAPID ANALYSIS OF OFF-SITE CONSEQUENCES OF AIRBORNE RADIOACTIVE RELEASES

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INTRODUCTION

A number of techniques exist for the assessment of the offsite consequences of accidental airborne releases of radioactive material. These range from hand calculation methods using graphs and tables (reference 1) to very complex wind field computer models (reference 2). The larger computer models require specialist personnel to operate them and, being centrally based, may not be available to operators due to communications breakdowns. Smaller computer models of varying complexity exist which can be used by the site personnel to determine the need for countermeasures in the early stages of an accidental release.

The UKAEA UMPIRE system (reference 3) was developed to provide predictions of doses to individuals from airborne releases. Advances in computer technology made it necessary to update the hardware and carry out modifications to the programs. A method of incorporating monitoring data to modify the dose predictions has been under development and the increase in computing power has enabled this to be included in the new system.

PHILOSOPHY

The immediate needs of the health physicist in advising on the need for countermeasures ranging from sheltering and stable iodine tablet issue to evacuation are to predict the current and future doses to individuals (within a few km of the site). A simple method for doing this based on monitoring results is used in the Harwell Survey Report Scheme (reference 4). However, in the case of fluctuating releases, changing weather conditions and a number of different sampling locations it is difficult to interpret the results or to predict the effects at other locations.

The UMPIRE 2 system uses measured and forecast weather conditions to predict the spread of the plume. It can use a previously calculated source term to give individual dose predictions. Once monitoring data has been obtained the interactive mode allows the source term, wind direction and effective release height to be modified. The modified data is then used to predict the consequences at locations where monitoring has not yet been carried out.

DATA REQUIREMENTS

Meteorology: To determine the consequences of an airborne release data on meteorological conditions is needed. In the simplest form this consists of time, wind direction, wind speed, cloud cover and rainfall rate. The atmospheric stability category which governs the dispersion is obtained from these values and an estimate of the mixing depth is also made. For consistency the simple method in reference 5 has been used.

Alternatively, a scheme exists to obtain meteorological data from the Meteorological Office in terms needed for this sort of calculation. The PACRAM scheme gives stability category, mixing height, wind speed and wind direction predictions at various times and distances from the site.

Release: To run the system in the predictive mode requires data on the source term consisting of the nuclides releases, the rate of release and duration and the effective height of the release. The interactive mode assumes a release of unit activity of limiting nuclides in terms of countermeasures ie I131, Pu239 and 1 MeV/Bq noble gas at a uniform rate for 4 hours at ground level unless a different source term is defined.

MODELLING

The dispersion model used is a simple two dimensional Gaussian model as described in reference 5. Plume depletion by dry and wet deposition and consequent ground deposited activity are calculated.

The airborne and ground deposited activity is divided into alpha and beta/gamma emitters. Doses from inhalation, groundshine and cloudshine are calculated for individuals. The data for inhalation dose to three age groups (infants, 10 year old children and adults) are taken from a computer data base developed by NRPB (reference 6). Groundshine doses are interpolated from data taken from reference 1 and from the WASH-1400 study (reference 7). Gamma cloudshine dose is calculated using the simple technique described in reference 3 switching from line source to a look up table to a semi infinite cloud model depending on the shape of the plume.

RESULTS

The dose isopleths can be displayed overlaid on a simplified map of the area surrounding the site selected. Doses to different organs and for the three age groups are available from inhalation only or inhalation plus external radiation. The development of the release can be followed by changing the interval between the start of the release and the display of results.

The modifying effects of introducing countermeasures can be shown.

A summary printout gives the input data and doses at selected locations in the plume at various times after the start of the release. This enables the health physics controller to concentrate resources on immediately affected areas while making preparations for countermeasures in other areas.

MONITORING RESULTS

For the purpose of comparison with monitoring data the following values are also calculated:

1. Gamma dose rate at 1m above ground.
2. Airborne beta/gamma activity kBq m^{-3}
3. Deposited beta/gamma activity kBq m^{-2}
4. Airborne alpha activity Bq m^{-3}
5. Deposited alpha activity Bq m^{-2}

By comparing the predicted and measured results modifications can be made to the source term with respect to quantity, proportion of nuclides and release height. Airborne activities can be tracked back to the current release rate and nuclide proportions; ground deposited activity gives an integrated history of the plume (although the situation is complicated if local rain showers occur) and gamma dose rate, when corrected for ground deposited activity and airborne particulate, indicates the noble gas activity (in terms of Bq-MeV).

A method has been developed (reference 8) to allow corrections to be made for differences between the measured wind direction and the effective plume direction and to scale the magnitude of the release (either overall or in sections) to fit the monitoring results. Dividing the release into alpha and beta/gamma emitters allows the source term to be calculated in terms of I131 and Pu239 which give a pessimistic estimate of the doses incurred.

If more detailed nuclide information is available from a stack monitor a detailed inventory or gamma or alpha spectrometry results this can be substituted for the basic source data. The program then calculates the offsite consequences of the release modified by the survey results.

USER INTERFACE

A major aim of the UMPIRE project has been to ensure that the system can be operated easily in a high stress situation. Experience with the early system has shown that it must be possible to alter any part of the input data rapidly. UMPIRE2 is controlled from a single page divided into topic areas. Most

items can be altered individually although some such as site and location within the site are interdependent, in these cases the user is prompted to enter the appropriate data.

FURTHER DEVELOPMENT

Despite the recent accident at Chernobyl it is in fact very unlikely that the system will ever be used in an actual accident. It is therefore proposed that a simulator is developed which produces aritifical data, possibly with random spurious results, which can be fed to emergency assessment personnel in emergency exercises. A second computer will probably be needed for this.

CONCLUSIONS

The UKAEA UMPIRE system has been used in emergency exercises to provide information to the health physics personnel assessing offsite consequences of an airborne radioactive release. UMPIRE2 has been developed to enable monitoring data to be used to reconstruct the release and extrapolate the resulting doses to other locations.

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ADDCOR: AN ATMOSPHERIC DISPERSION AND DOSIMETRY CODE FOR OPERATORS AND REGULATORS

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ABSTRACT

A general purpose atmospheric dispersion and radiation dosimestry code is described. This code provides a ready means for estimating the degree of compliance of radiation doses received by members of the public and others with the ICRP dose limits. The code is especially applicable to the normal atmospheric releases from a nuclear establishment with a diverse and varying range of activities. It can also be applied to accidental releases.

Major features of the code are that it
- provides for a number of sources and receptor locations which can be readily varied;
- provides hourly nuclide release rates from summary input data with hourly meteorological data;
- contains an extensive library of the dose conversion factors;
- allows a variety of exposure routes and individual habits to be included.

The code can also be used for estimating collective doses. Finally, the code has been written in FORTRAN 77 for portability and has been highly structured to aid approval reviewing by authorizing bodies.

An illustration of the code is included.

A COMPUTER MODEL ON CONSEQUENCES OF SUDDEN RELEASE OF RADIOACTIVITY INTO THE ENVIRONMENT

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ABSTRACT

Events like Chernobyl and Three Mile Island have demonstrated the vulnerability of reactors to accidents and human errors in spite of elaborate safety measures. Radioactive plume from such accidents can propagate over large distances. This calls for a quick assessment of the situation for emergency planning. Keeping this in mind a software has been developed on a personnel computer (an IBM PC compatible) with 256 K memory. This predicts the exposure level due to accidental release of radioactivity in the atmosphere. Gaussian dispersion has been assumed for both horizontal and vertical propagation. This is a modified version of WASH-1400 (Ref: USNRC(1975) Reactor Safety Study - an assessment of accident risks in US commercial nuclear power plants NUREG-75/014). Input release characteristics include magnitude, duration, isotopic concentration, probability and height of release. Meteorological factors like wind speed and atmospheric stability have been incorporated in a time variant mode. This software has the added advantage of the presentation in color graphics enabling quick visual judgement.

This was employed to evaluate the dose levels around Chernobyl. The meteorological data were deduced from the published accounts. A half an hour release of 8 MCi, excluding noble gases, on the first day at a height of 1200 m was assumed with I-131 as the dominant component. For noble gases a 100% release was assumed (Ref: IAEA Safety Series No. 75-INSAG-1, 1986). The external exposures are estimated to be 140 mR/hr each for the population around 4 km distance. This compares reasonably well with the data published in IAEA report.

COMPUTER-CONTROLLED CONTINUOUS MEASUREMENT OF RADIOACTIVE AEROSOL NEAR NUCLEAR POWER PLANTS

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ABSTRACT

The essential components of the continuous measurement system of radioactive aerosol consist of the air sampler and the counting equipment. We have been continuously measuring radioactive aerosol concentration since April 1986. The radioactive aerosol in the air consists mainly of radon(^{222}Rn), thoron(^{220}Rn) and their decay products. The concentration of these nuclides fluctuate about two order at outdoor(Shimo and Ikebe,1979) and are much higher than the ordinary released concentration of nuclides from nuclear power plants. Accordingly, it is very difficult to separate them based on real-time measurements. We found a solution to measuring alpha and beta activities simultaneously during the collection of radioactive aerosol, so we are now able to discriminate between manmade radioactive nuclides from the Chernobyl accident and the decay products of radon.

INTRODUCTION

It has been very important to continuously measure the variation of environmental concentrations of radioactive aerosol for the detection of the nuclides from nuclear power plants, and for the study of internal irradiation caused by inhalation of radon, thoron and their decay products. The method of detecting manmade radioactive nuclides in aerosol is generally to measuring activities of aerosol sample after decrease of the short-lived radon daughters by radioactive-decay. However, this method has never been accomplishment to measuring concentrations of short-lived radon daughters .

In this work, we adopted the continuous measurement system of radioactive aerosol in the air, which is accomplished through filter collection and by simultaneously measuring both alpha and beta activities during the collection. The pulse height from the detector depends on an incidence radioactive ray, so they can be separate by pulse height. If aerosol consists of only short-lived radon daughters, so the ratio of beta concentration with alpha concentration is almost constant. Therefore, we are able to discriminate between beta-emission nuclides and the short-lived radon daughters.

THE STRUCTURE OF MEASUREMENT EQUIPMENT

Figure 1 shows the computer-controlled continuous measurement equipment. It is located in a monitoring station. The equipment is composed of an air collecting system and a counting system. We use

a roll glass-cellulose filter paper(TOYO HE-40T, 90m) to collect aerosol with a pump. The size of an intake is 45mm in diameter. The detector is placed opposite to an intake. Accordingly, during the collection, it is possible to measure both alpha and beta activities for aerosol on the filter. The flow rate is about 200l/min.

It should be noted that a glass-cellulose filter paper can not collect gaseous iodine, so we adopted a charcoal-cartridge treated with 10 percent-weight tri-ethylene-di-amine(TOYO CHC-50, TEDA 10%). For the purpose of effective collection for gaseous iodine in a month, the air through a charcoal-cartridge is branched from the air intake and a heater is equipped at previous roots of collection to gaseous iodine(Yoshida and Naritomi,1974).

The counting system consists of a plastic scintillation detector coated with zinc sulfide silver-activated on it's surface, a single channel analyzer, a counter, a coincidence circuit, a timer, a printer and a sequence controller. A controller which has a microcomputer for the management of this continuous measurement system gives two signals. One of the signals starts the filter step transfer to the collecting system, the other starts the counter. The pulse height from the detector depends on an incidence radio-active rays. The pulse height originated in alpha is much higher than the one originating in beta. Therefore, it is able to separate alpha-ray from beta-ray signals by a single channel analyzer. The pulse from the single channel is counted by a counter.

The intermixture rate to alpha from beta is almost 0% and the intermixture rate to beta from alpha is about 10%. Efficiency for alpha from the U_3O_8 plate standard source is about 10% and the beta is about 30% from the same source with a 27mm aluminum shield to alpha. The minimum detectable activity is 70mBq/m^3 for alpha activity and 180mBq/m^3 for beta activity when all of the radioactive aerosol is composed of short-lived radon daughters.

We have five systems. Four systems among them are placed in a monitoring station near nuclear power plants. The distance from

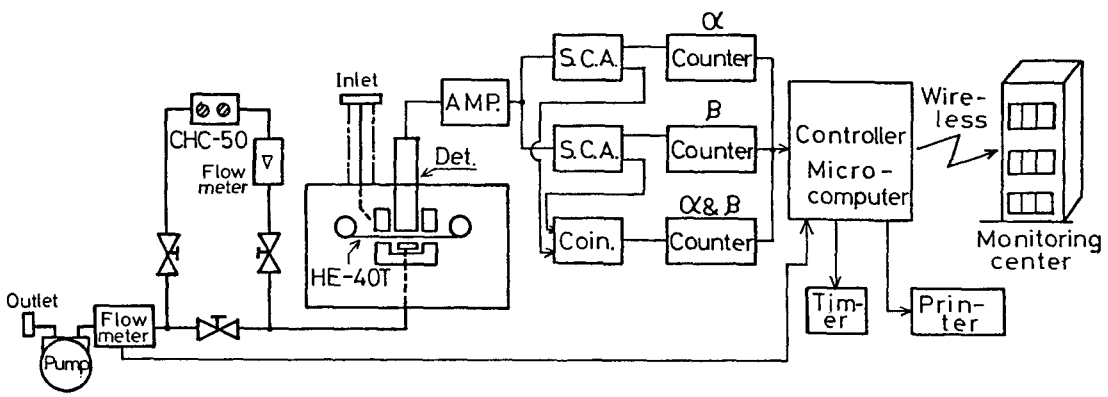

Fig.1 The structure of measurement equipment.

the monitoring station to the plants is between 1km to 2km. The distance of each monitoring station is about 10-15km, and the distance from the monitoring station to our monitoring center varies from 10km to 50km (Hayakawa,Iyo and Ohnishi,1982). Measurement data is automatically sent to the monitoring center at ten minute intervals by a wireless transmission controlled by a computer at the monitoring center. The concentration of radioactive aerosol is calculated by using a 32bits computer. The calculation for concentration of short-lived radon daughters is done by assuming the radioactive equilibrium with the short-lived radon daughters and neglecting the thoron series.

RESULT

Figure 2 shows the monthly arithmetic mean of alpha and beta in the surface layer. Seasonal variations of short lived radon daughters at ground level show a maximum in June-September and a minimum in January-March. The concentration is about two times higher at the maximum season than minimum season. The results of the measurements show that the concentration level of short lived radon daughters is close, except at OGURUI. The seasonal variation is also very similar.

The Chernobyl accident occurred in late April 1986. Figure 3 shows the concentration of short lived radon daughters calculated by alpha or beta counting, and the ratio of beta to alpha concentration. The diurnal variations show a minimum at noon and a maximum at midnight. The average concentration of short-lived radon daughters during April 1986 was 2.07Bq/m^3 for alpha counting (with a range from 0.59 to 9.91 Bq/m^3), and 1.78 Bq/m^3 for beta counting (with a range from 0.55 to 8.44 Bq/m^3). These concentration levels were the same during May 1986. So, it is hard to discriminate between radioactive nuclide aerosols released from nuclear power plants and short-lived radon daughters merely by observing the concentrations from alpha or beta measurements.

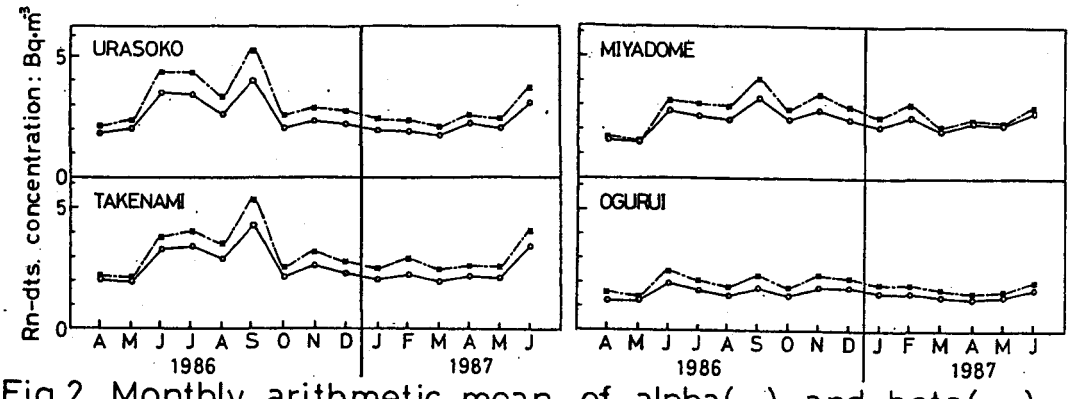

Fig.2 Monthly arithmetic mean of alpha(---) and beta(——) in the surface layer in Fukui during Apr. 1986 - Jun. 1987.

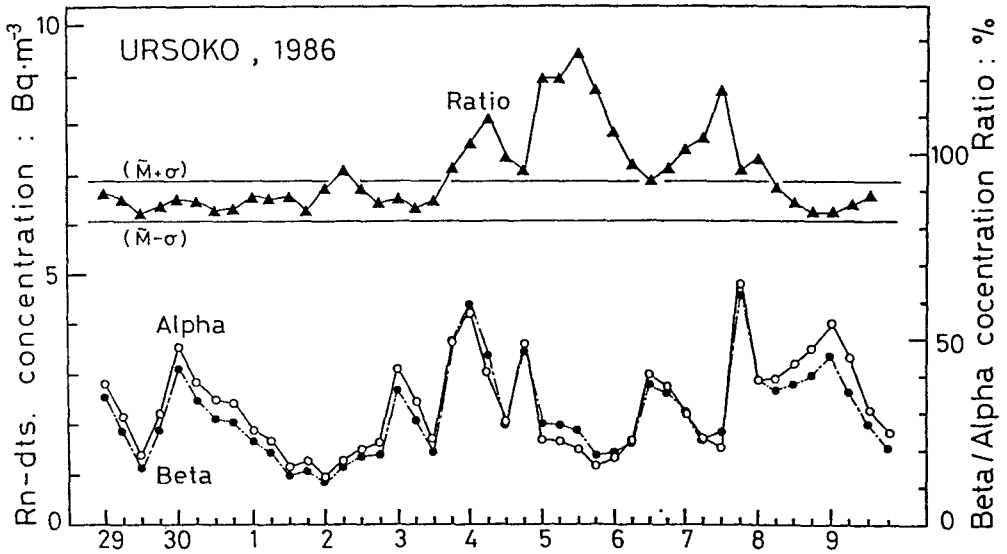

Fig.3 6-hours averaged Rn-dts. concentration by alpha and beta, and beta/alpha ratio in the surface layer during 29.Apr-9.May.

The average of ratio, which is beta concentrations to alpha concentrations, is 87% and the standard deviation is 5%, during April 1986. The ratio is almost changeless. But, the remarkable change of ratio is shown in Figure 3. They correspond to the additional nuclides of a short-lived radon daughter's activity as measured at the Chernobyl accident independently on real-time aerosol monitoring.

CONCLUSION

From April 1986 to June 1987, seasonal variations of short-lived radon daughters showed a maximum in June-September and a minimum in January-March. In spite of real-time measurement for radioactive aerosol, we are able to discriminate between the radioactive aerosols from the Chernobyl accident and the short-lived radon daughters.

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REMOTE MEASURING SYSTEM FOR MONITORING THE IODINE IMMISSION IN THE VICINITY OF NUCLEAR POWER PLANTS *)

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Radioactive substances may be released into the atmosphere during major accidents at nuclear facilities. Since the emission rate and composition of the mixture of emitted nuclides is generally not known, appropriate measurements must be made in the vicinity of the plant. The radiation exposure in the possibly critical exposure pathways, i.e. the whole-body dose due to Y-submersion and the thyroid dose due to incorporation of radioiodine, should be determined immediately at various locations in the environment.

In general, existing systems only provide telemetric monitoring of the Y-dose rate during an accident. A suitable iodine measuring instrument and a data transmission and processing system have therefore been developed at the Nuclear Research Centre Jülich for use during accidents (1-3). This system consists of six iodine measuring instruments in the vicinity of the Nuclear Research Centre Jülich (KFA) and a central measuring unit which are connected to each other via a dedicated line. During an accident, the iodine measuring instruments determine the iodine thyroid dose which is proportional to the I-131 activity collected in the iodine detector.

The iodine measuring instrument contains a small NaI scintillator (10 mm in diameter x 10 mm in height) in a lead shielding 15 cm in thickness (Fig. 1). The scintillator is surrounded by a ring-shaped iodine collecting cartridge through which a controlled air flow of 100 l/h can be sucked via an aerosol filter. This air supply is only switched on from the central measuring unit in the case of an accident. A glass-fibre filter is used as the prefilter. The iodine collecting cartridge is filled with silver zeolite granules. All hose connections and the filter casing are made of Teflon. The lead shielding contains a heating element by means of which the collecting cartridge is heated to 10°C above the ambient temperature. This prevents a reduced iodine deposition at high atmospheric humidity. A Ba-133 source is mounted above the NaI scintillator. It serves to control the amplification of the detector by means of a two-channel discriminator adjusted to the Ba-133 Y-line. The pulses recorded in the NaI scintillator are measured by a single-channel discriminator adjusted to the 356 keV Y-line of the Ba control source.

The central measuring unit interrogates the individual stations at intervals of 1 minute, calculates 10-minute mean values and converts the count rates into dose values. The values are displayed on a screen, plotted on a multiple recorder and printed out together with date and time. Two threshold values

*) Supported by funds from the Federal Ministry of the Interior, funding number St.Sch. 936

can be preset. The failure of or a defect in a measuring station is identified by the central measuring unit and indicated in the measuring records.

The iodine measuring system can be extended to form a comprehensive accident measuring system which is also capable of measuring the γ-dose rate and assessing the aerosol concentration. In order to demonstrate this, one of the six iodine measuring instruments was extended by a γ-probe with two Geiger-Müller counters and by an aerosol monitor.

The measuring range for the iodine thyroid dose extends from 1.5 mSv to 20 Sv. The value of the lower detection limit is given by the statistical fluctuations of the count rates of the Ba-133 control source. The measuring instrument is equally suitable for measuring elementary and organically bound I-131. Even in the case of high atmospheric humidity, the collection of organically bound iodine is not impaired. The amplification of the iodine measuring instrument is controlled so well that the count rate varies by less than 2.5 % in the temperature range from 1°C to 40°C. The amplification control is insensitive to interfering radiation. If, for example, 9.3 times as much radiation from Cs-137 as from I-131 is recorded in the window of the I-131 line (Fig. 2), the amplification is nevertheless kept constant for at least a further 24 h. External γ-radiation is attenuated by a factor of more than 10,000 due to the lead shielding. Radioactive noble gases only increase the detector reading very slightly. The increase in this reading corresponds to a thyroid dose which is small in comparison to the dose rate in the cloud of radioactive noble gases. The iodine measuring instrument requires practically no servicing since the air flow is only switched on when required.

The iodine measuring system has been in operation since April 1985 and no major malfunctions have occurred to date. The iodine thyroid dose caused in Jülich by the reactor accident at Chernobyl was smaller than the lower detection limit during the normally applicable measuring time of 10 minutes. By increasing the measuring time to 1 hour it proved possible to measure the iodine thyroid dose caused by the disaster at Chernobyl (Fig. 3).

In order to be able to measure also the iodine thyroid dose in the case of minor incidents or accidents at great distances, the lower detection limit is in future to be reduced by a factor of at least 10. After preliminary measurements it seems possible to achieve a lower detection limit of at least 0.1 mSv using a selected CdTe detector 10 mm in diameter and 2 mm in thickness in conjunction with the iodine accident measuring system. Since the temperature dependence of the amplification of the CdTe detector is considerably lower than that of the NaI scintillator, no amplification control would be required for the CdTe detector and the entire accident measuring system would thus be further simplified.

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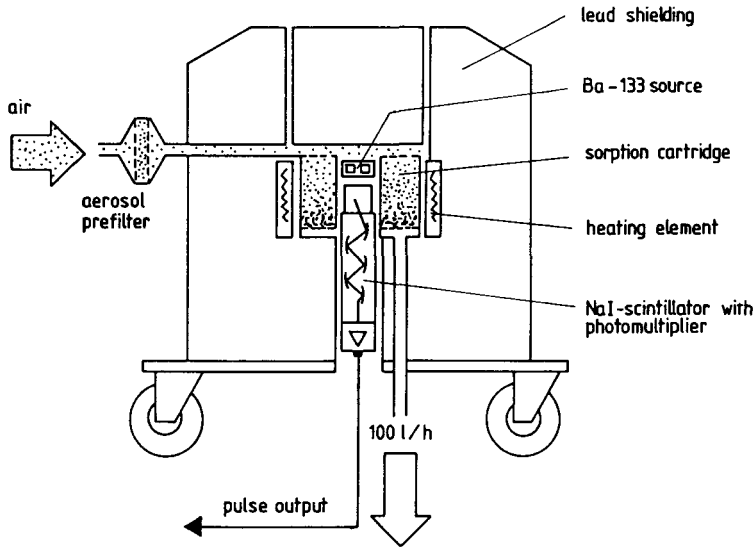

Fig. 1: Diagramatic representation of the iodine measuring instrument

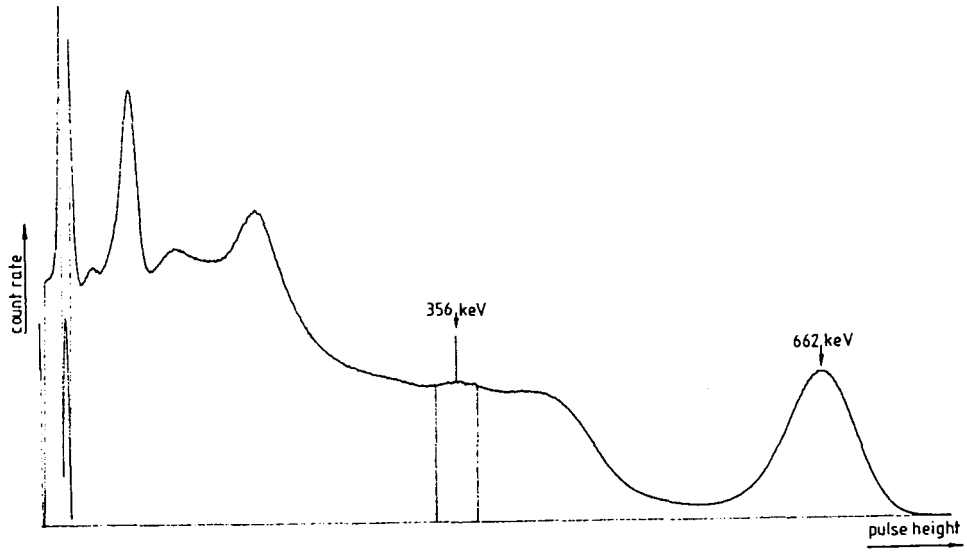

Fig. 2: Y-spectrum of Cs-137 and Ba-133 for a count rate ratio of Cs-137: Ba-133 = 9.3 : 1 in the window of the 356 keV Ba-133 Y-line. The position of the window is shown

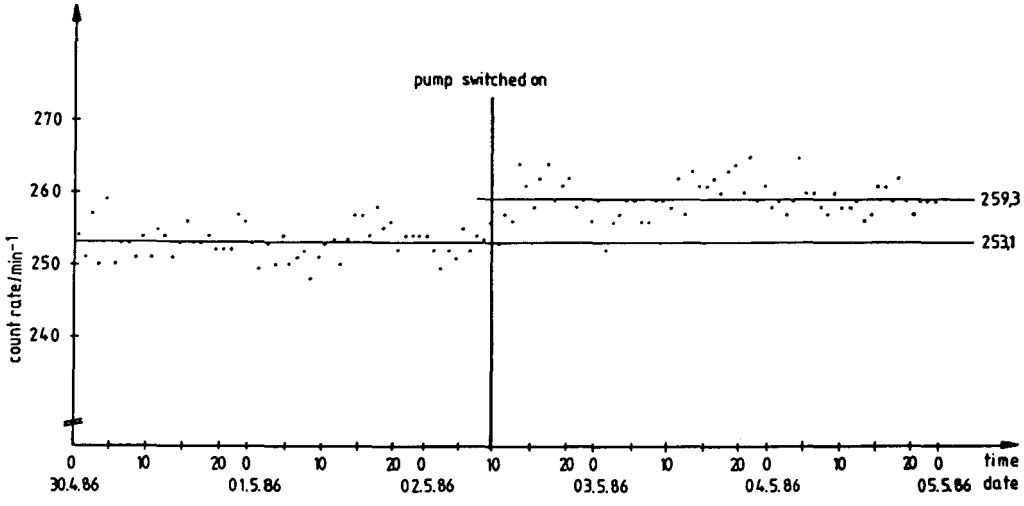

Fig. 3: Reading of an iodine measuring instrument after the Chernobyl accident

IN-SITU MEASUREMENT OF RADIOACTIVE GASES USING Ge(Int) SPECTROMETORY FOR ESTIMATING THE GASES FLOW RATE

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ABSTRACT

Recently, according to the improvement of fuel performances, short-lives nuclides such as ^{19}O and ^{16}N have become mainly of the gaseous radioactivity in the turbine system at BWR plants. Therefore, the variation of the flow rate in main steam system greatly infuluences to the area radiation level, and it is very important to grasp the flow rate and the relations to the dose rate. On the other hand, the conventional flow rate measuring devices are accompanied with their inherent instabilities of indication on measuring the steam flow.

In this study, we confirmed the existence of these short-lives nuclides and their contribution to area dose rate by in-situ measurment using a portable Ge(Int) spectrometer, and developed a method for estimating flow rate by radioactive gases analysis.

IN-SITU MEASUREMENT AND STUDY

a. Measuring Device

The measuring device comprises Ge(Int) detector and pulse-height analyzer. The relative efficiency and resolution of Ge(Int) detector are 20 % and 2 keV, respectively. It has a liquid-nitrogen container of 1.2 ℓ capacity and is capable of performing one-day measurement. The pulse-height analyzer has a memory capacity of 4098 channels and high-voltage power supply unit, amplifier and a cassette tape for recording measured data are all built in. The measurement result is analyzed by a separate computor. The shielding around the detector is made of 10 cm thick lead and has a collimating function and therefore is capable to perform measurement even at the field where the dose rate is at high level.

b. Measurement around Main Steam Line

The measurement is performed around main steam line at both reactor outlet side and low-pressure turbine inlet side (See Fig. 1). The measurement corresponding to electrical output was made at low-pressure turbine inlet side, and the measurement at a time of full power only was made at reactor outlet side.

The result of gamma-ray spectrum and surface dose rate shows in Table. 1 and Fig. 2.

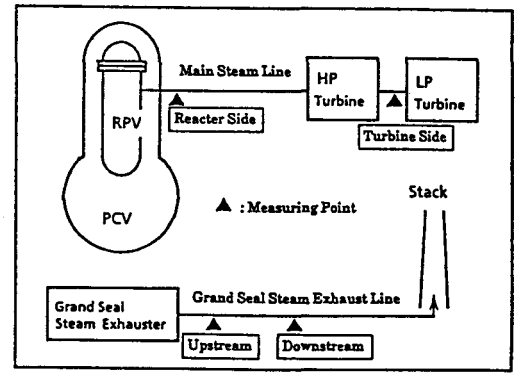


Fig.1 Measuring Point

From the measurement by Ge(Int)detector, only two nuclides of ^{16}N (half-life 7.13 sec.) and ^{15}C (half-life 2.4 sec.) were analyzed. The following differences are seen by looking from the peak count ratio of gamma-ray of 6.13 MeV and 5.3 MeV.

$$Rr = \left\{ \frac{^{15}C\ 5.3\ MeV}{^{16}N\ 6.13MeV} \right\}_{\text{Reactor side}} = 0.752$$

$$Rt = \left\{ \frac{^{15}C\ 5.3\ MeV}{^{16}N\ 6.13MeV} \right\}_{\text{Turbine side}} = 0.321$$

Since these differences are considered to have resulted from the difference of radioactive decay effect during steam flow from reactor side to turbine side, the steam travelling time (T) is obtained from the following equation:

$$Rr = \exp (\lambda n - \lambda c) T = Rt$$

$$T = \frac{\ell n \left(\frac{Rt}{Rr}\right)}{\lambda n - \lambda c} \qquad (1)$$

λn : Decay constant of ^{16}N

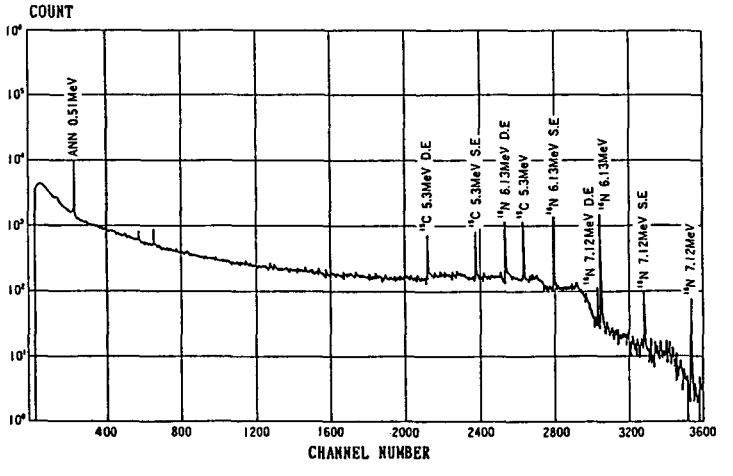
λc : Decay constant of ^{15}C

The steam travelling time from reactor side at full power to turbine side calculated by equation (1) will be approx. 4.4 seconds.

ELECTRICAL ONTPUT	TURBINE SIDE				REACTOR SIDE			
	DOSE RATE	PEAK COUNT RATE			DOSE RATE	PEAK COUNT RATE		
		^{16}N(6.13MeV)	^{15}C(5.3MeV)	^{16}N+^{15}C		^{16}N(6.13MeV)	^{15}C(5.3MeV)	^{16}N+^{15}C
MWe	mR/hr	cps	cps	cps	mR/hr	cps	cps	cps
163	23	24.0	2.1	26.1	400	--	--	--
183	31	31.3	4.0	35.3	450	--	--	--
254	59	53.2	9.4	62.6	620	--	--	--
309	105	71.2	19.6	90.8	720	--	--	--
357	140	83.6	26.8	110.4	830	44.4	33.4	77.8

Table 1 Result of gamma-ray spectroscopy and surface dose rate of main steam line measurment

Fig. 2 Gamma-ray spectrum at main steam line



The relationship between electrical output and dose rate will be almost 1 to 1 around main steam line, while rising ratio of dose rate of turbine side against electrical output is larger compared that of reactor side (See Fig. 3).

This can be explained by the fact that the nuclides contributing to dose rate are of only two kinds of ^{16}N and ^{15}C and that flow rate is proportional to electrical output. That is: the dose rate at reactor side which corresponds to the output at a ratio of 1 to 1 becomes larger at turbine side because of decay effect that if flow rate becomes larger the flow travelling time becomes faster and decay effect becomes smaller.

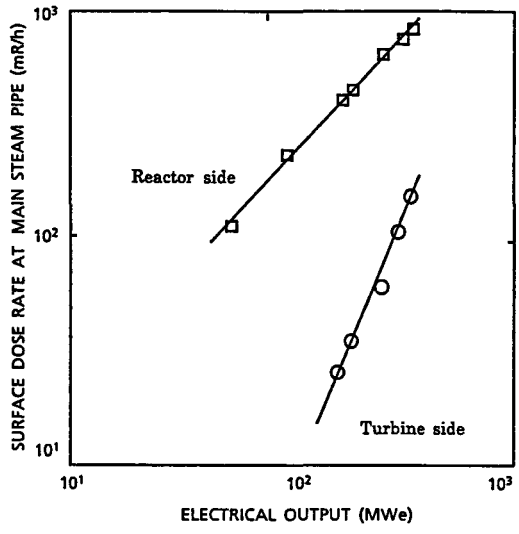


Fig. 3 Relationship between dose rate and electrical output

This is verified by the fact that the result of reverse operation of the peak count of ^{16}N and ^{15}C at each power level at turbine side to the peak count at reactor side from decay effect and steam flow rate, calculated by the following equation, often agrees with inclination curve of the dose rate at reactor side (See Fig. 4).

$$Nr = Nt * \exp(\lambda n) * R * T$$

T : Steam travelling time at full power (4.44 sec.)
R : Full power/applicable power ratio
Nt: Peak count at turbine side (actual measurement)
Nr: Peak count at reactor side

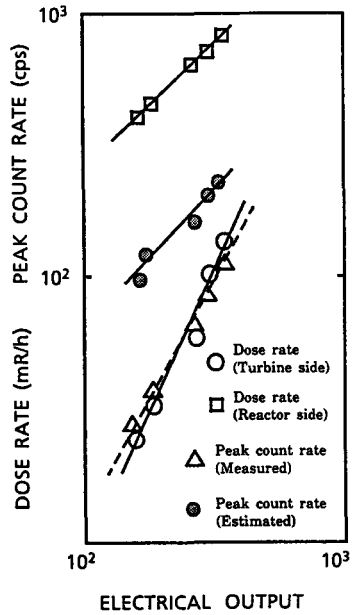
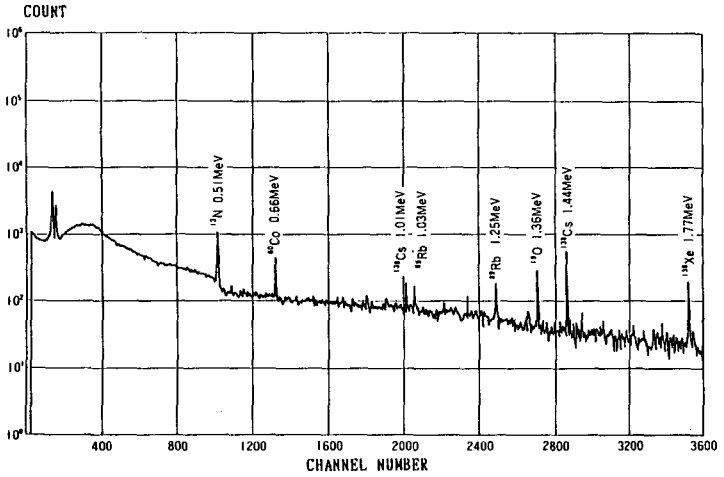


Fig.4 Dose rate and peak count rate at each electrical output

c. Measurement around Turbine Gland Seal Steam Exhaust Line

For measurement on turbine gland seal steam exhaust line, the measurement was performed on both upstream and downstream side like the case of main steam line, using Ge(Int) detector (See Fig. 1). Examples of measured spectrum are shown in Fig. 5. From the concerned exhaust line, gamma-ray of ^{19}O (1357 keV, half-life 9.96 min.), ^{13}N (511keV, half-life 26.9 sec.),etc. were analyzed.

Fig. 5 Gamma-ray spectrum at turbine grand seal steam exhaust line



Like the case of main steam line, the steam travelling time obtained from the difference in the peak count of ^{19}O and ^{13}N was 41.8 sec. Further, the gas flow rate obtained from the distance between measured points (21.8 m) and pipe cross sectional area (0.235 m^2) was 442 m^3/hr, and this value agrees quite well with the indication of conventional process flowmeter.

CONCLUSION

By in-situ measurement using Ge(Int) detector, identification of short-lives nuclides, which is difficult to analyze with the normal sampling measurement, becomes possible and, in addition, the nuclides contributing to the dose rate is clarified and also gaseous travelling time or flow rate are obtainable.

In the past, at nuclear power plant it was considered that contribution of fission products to the dose rate such as noble gas contained in the gas was large and that indication of gas monitor was affected by this contribution.

However, at present where fuel performance has been improved, indication of the monitor is affected by ^{13}N, ^{16}N, ^{15}C, ^{19}O, etc.

Accordingly, it is necessary to pay attention to the fact that flow rate gives influence to variation of gas monitor indication since these nuclides are of short-lives nuclides.

A NEW TYPE OF STACK GAS MONITOR DIRECTLY INDICATING EXPOSURES IN ENVIRONMENT

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INTRODUCTION

In case of reactor accidents, it is desirable that a stack gas monitor for measuring of mixed radioactive rare gases of high-concentration indicates a value directly proportional to the exposure rates in the environment. A new type of stack gas monitor was developed to predict the exposure rate in the environment through measurement of exposure rate and an average γ-ray energy of released mixed radioactive rare gases. This paper describes the principles of the new measurement method.

PRINCIPLE OF A NEW STACK GAS MONITOR[1)]

The exposure rates at ground surface in the downwind axis due to radioactive rare gases released from an exhaust stack have been calculated by the following equation[2)].

$$D(x_0,y_0,0) = K\cdot\sum_{i,j} Q_i\cdot P_{i,j}\cdot E_{i,j}\cdot\mu_a(E_{i,j})\cdot\int_0^{\infty}\int_{-\infty}^{\infty}\int_0^{\infty}\frac{e^{-\mu(E_{i,j})\cdot r}}{4\pi r^2}\cdot B(E_{i,j},r)\cdot\chi(x,y,z)\,dx\,dy\,dz \quad \cdots \quad (1)$$

where, $D(x_0,y_0,0)$: exposure rate at a point $(x_0,y_0,0)$
K : conversion factor to exposure rate (5.08×10^{-2} $\mu R\cdot m^3/MeV\cdot B_q\cdot h$)
r : distance between a volume element in the radioactive plume and the calculation point
Q_i : radioactivity release rate of the i th nuclide
$P_{i,j}$: absolute emission rate of the j th γ-ray of the i th nuclide
$E_{i,j}$: γ-ray energy
$\mu_a(E_{i,j})$: energy absorption coefficient of air
$\mu(E_{i,j})$: attenuation coefficient of air
$B(E_{i,j},r)$: dose build-up factor of air
$\chi(x,y,z)$: radioactive concentration in a radioactive plume for a unit release rate from exhaust stack

In the calculation, however, complicated nuclide analysis of released radioactive rare gases must be made so that the radioactivity release rate of respective nuclide could be obtained.

A new stack gas monitor, as shown in Fig. 1, has two detectors in a limited space, whose γ-energy dependence is different each other. When mixed radioactive rare gases are allowed to flow continuously through the limited space (cylindrical gas tank), exposure rates at the center of the cylinder, D, can be measured with the detectors and average γ energy of the gases, $\bar{E}$, can be also determined using the sensitivity ratio between the response of the detectors. If the quotient of radioactive concentration by γ ray flux density is defined "γ-ray collection function", ε, the equation (1) can be written as the following equation (2).

$$D(x,y,0) = \frac{D \cdot V}{\varepsilon} \int_0^{\infty}\int_{-\infty}^{\infty}\int_0^{\infty} \frac{e^{-\mu(\bar{E})\cdot r}}{4\pi r^2} B(\bar{E},r)\cdot\chi(x,y,z)\,dx\,dy\,dz \quad \cdots\cdots \quad (2)$$

where V: exhaust air flow rate from stack.

Therefore, the exposure rate in downwind axis at ground surface can be calculated directly without the measurement of radioactivity release rate by laborious nuclide analysis. In the equation (2), D, V and $\bar{E}$ are measurable phisical quantities and the value of ε can be determined experimentally using the standard radioactive rare gases such as ^{41}Ar and ^{133}Xe. DV/ε has a special unit (R/h·m²·1/h) and so the authors call it "4π exposure rate release per unit time".

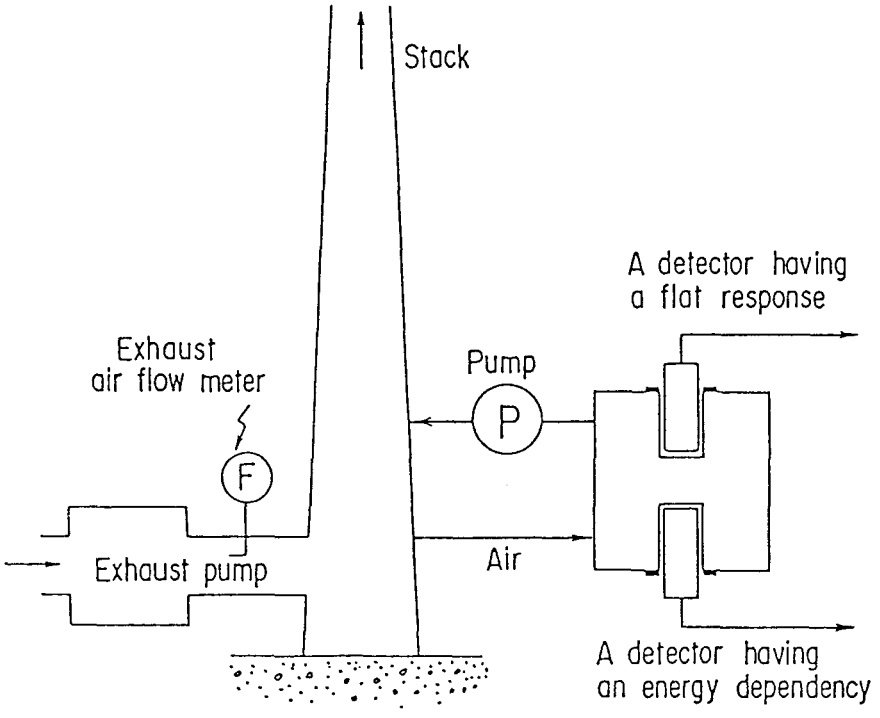

Fig. 1 Schematic diagram of an emergency-use stack gas monitor

RELATION BETWEEN THE INDICATED VALUE OF THE MONITOR AND THE EXPOSURE RATE IN THE ENVIRONMENT

In the equation (2), the term of integration means the product of radioactive concentration and the volume element, that is, the integration of radioactivity divided by area. It is thus the γ-ray flux density at the exposure point $(x_0, y_0, 0)$. Accordingly, the equation (2) is written as

$$D(x_0, y_0, 0) = \frac{D \cdot V}{\varepsilon} N(x_0, y_0, 0) \quad \cdots\cdots\cdots\cdots\cdots\cdots \quad (3)$$

where $N(x_0, y_0, 0)$: γ-ray flux density at the point

In the equation (3), $V \cdot N(x_0, y_0, o)$ has the dimension of length (m), so it can be the γ-ray collection function at the point in the environment $\varepsilon_{en}(x_0, y_0, o)$. Therefore the equation (3) is written as

$$D(x_0, y_0, 0) = \frac{\varepsilon_{en}(x_0, y_0, 0)}{\varepsilon} \cdot D \quad \cdots\cdots\cdots\cdots\cdots\cdots \quad (4)$$

where $\varepsilon_{en}(x, y, 0)$: γ-ray collection function at an exposure point in the environment

In the equqtion (4), when the ratio $\varepsilon_{en}(x_0, y_0, 0)/\varepsilon$ defined as $K_1(x_0, y_0, 0)$, is regarded as a constant the exposure rate at a exposure point in environment is found to be proportional to the indicated exposure rate in the stack gas monitor.

$\varepsilon_{en}(x_0, y_0, 0)$ in eq. (4) is calculated with GAMPUL code[3] for the degrees of atmospheric stability, for a specific height of the stack, with the normalization of γ-ray energy 1 (MeV), radioactivity release rate 1 (B_q/h) and wind velocity 1 (m/s). So the values of $K_1(x_0, y_0, 0)$ are obtained with atomospheric stability as parameter.

There is the possibility that the value of $K_1(x_0, y_0, 0)$ varies with γ-ray energy of released radioactive rare gases. Therefore, the correction factor for γ-ray energies other than 1 (MeV) was calculated by the equation (5). However, variation of the γ-ray collection function ε with the energy can be negligible because the cylindrical space of the stack gas monitor is sufficiently small compared with the environment and the energy albedo due to gas tank walls is less than 10 %.

$$K_2(\bar{E}) = \frac{\int_0^\infty \int_{-\infty}^{\infty} \int_0^\infty (4\pi r^2)^{-1} \cdot e^{-\mu(\bar{E}) \cdot r} \cdot B(\bar{E}, r) \cdot \chi(x, y, z) \cdot dx\ dy\ dz}{\int_0^\infty \int_{-\infty}^{\infty} \int_0^\infty (4\pi r^2)^{-1} \cdot e^{-\mu(1) \cdot r} \cdot B(1, r) \cdot \chi(x, y, z) \cdot dx\ dy\ dz} \quad \cdots \quad (5)$$

Where,

$K_2(\bar{E})$: γ-ray energy correction function for $K_1(x_0,y_0,0)$

The calculation with eq. (5) can be done with GAMPUL code[3] for respective degrees of atmospheric stability. With $K_2(E)$ and $K_1(x_0,y_0,0)$, the eq. (4) is given as eq. (6). In the equation, the fact that the exposure rate is inversely proportional to the wind velocity at an exposure point, is taken into account.

$$D(x_0,y_0,0) = \frac{K_1(x_0,y_0,0)\cdot K_2(\bar{E})}{u} \cdot D \quad \cdots\cdots\cdots\cdots\cdots \quad (6)$$

Where,

u : average wind velocity (m/s).

As the variation of $K_2(E)$ with γ-ray average energy of released radioactive rare gases is about ±50 % in a wide energy range of 50 keV to 2 MeV, so that the value is essentially taken as 1. Therefore, the exposure rate at an exposure point is obtained easily from the measured exposure rate in the monitor, the wind velocity and the proportional constant $K_1(x_0,y_0,0)$ determined by atomospheric stability, irrespective of the γ-ray energy.

CONCLUSION

Major features of this gas monitor are as follows. Exposures in the environment due to release of mixed radioactive gases from a reactor can be estimated accurately and rapidly without laborious means such as nuclide analysis. And calibration and adjustment of the apparatus can be simplified. So this type of stack gas monitor is suitable especially in case of reactor accident.

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APPLICATION OF ACTIVATED CARBON FIBER TO A FILTER USED FOR AIRBORNE RADIOIODINE SAMPLING

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INTRODUCTION

An airborne radioiodine sampling filter is required to have low pressure drop, mechanical strength enough to a practical use and high collection efficiency under high relative humidity(RH). To develop a filter to meet the requirements, the influences of impregnation amount of triethylenediamin(TEDA) on the collection efficiencies for methyl iodide and the reaction rates were investigated for several kinds of activated carbon fiber varied in specific surface area, pore diameter, etc. Silver silica gel(Sut Chemi, AC6120)[1], silver zeolite(CTI Nuc., AgX Type III)[2], silver alumina(Hitachi Co.)[3] and granular activated charcoal were also examined for comparison.

A new type filter made of activated carbon fiber(ACF filter) was developed based on the above experimental results. The ACF filter was examined for the pressure drop by the filter and collection efficiency for methyl iodide being compared with other types of filters such as an activated charcoal cartridge(ACC)[4] and an activated charcoal filter paper(ACP)[5].

EXPERIMENTAL

Specification of activated carbon fibers(Toyobo Co. Ltd.) tested are shown in Table 1. The surface of the fibers is smooth and there is no macropore on the surface.

Table 1 Specification of examined activated carbon fibers

Fiber	Diameter of fiber (μm)	Density (g/cm^3)	Specific surface area(m^2/g)	Mean pore radius (nm)	Bulk density (g/cm^3)
Fiber-A	19	1.30	1100	2.0	0.073
Fiber-B	25	1.05	1500	2.0	0.050
Fiber-C	15	0.90	1650	2.0	0.055
Fiber-D	25	1.10	1300	2.4	0.057
Fiber-E	9.0	0.70	2000	2.0	0.050
Fiber-F	9.0	1.10	1400	2.0	0.19

An apparatus used for testing on collection efficiency was made up of an air compressor, a dryer, a boiler, a $CH_3{}^{131}I$ generator, a test holder, etc. The main parts were installed in a constant temperature box. For the examination of fibers, the test holder was composed of a cartridge, in which an adsorbent such as

the fibers, Ag-zeolite etc. was packed, and a back-up trap. On the other hand, for the examination of filters, the holder was composed of a test filter, such as ACF filter and ACP, and a back-up trap.

An air mixed with $CH_3{}^{131}I$($2x10^{-9}$ g/cm^3) was drawn through the test holder. Tests were carried out under RH of from 20 % to 90 %, temperature of 30 °C, a flow rate of 50 liter/min and sampling time of 60 min. After collecting the iodine, the radioactivities in the cartridge, the test filter and the back-up trap were measured to calculate collection efficiencies.

RESULTS AND DISCUSSION

1. CHARACTERISTICS OF ACTIVATED CARBON FIBER

IMPREGNATION AMOUNT OF TEDA TEDA is usually impregnated to collect organic iodide. Figure 1 shows the relationships between the impregnation amount of TEDA and the collection efficiencies at 90% RH. Collection efficiencies for all fibers varied depending on the amount of TEDA. It was observed that the collection efficiency had a maximum at a certain impregnation amount of TEDA. The impregnation amount was adopted for the following examinations.

REACTION RATE OF ACTIVATED CARBON FIBER WITH METHYL IODIDE A collection efficiency for methyl iodide depends on thickness of adsorbent, flow rate, sampling time, RH, etc. Therefore, the performance of filter is generally expressed by a rate constant defined by the following equation [6,7]: $C/C_0 = \exp(-K \cdot T)$,where C: outlet concentration, C_0: inlet concentration, K: rate constant, T: residence time in the adsorbent. Table 2 shows the rate constants of fibers measured at 90 % RH.

Table 2 Rate constants for reaction with methyl iodide

Adsorbent	Shape	Impregnant	Rate constant (s^{-1})
Fiber-A	fiber	0.7 wt% TEDA	16
Fiber-B	fiber	3.0 wt% TEDA	23
Fiber-C	fiber	6.5 wt% TEDA	53
Fiber-D	fiber	4.3 wt% TEDA	49
Fiber-E	fiber	7.3 wt% TEDA	68
Fiber-F	fiber	6.0 wt% TEDA	121
Activated charcoal	granule(12x30)	5.0 wt% TEDA	56
Ag-zeolite	bead(10x16)	38 wt% Ag+	18
Ag-silica gel	bead(10x16)	12 wt% $AgNO_3$	23
Ag-alumina	bead(10x16)	9 wt% $AgNO_3$	38

The rate constant of Fiber-A which has small specific surface area was smallest. On the other hand, the Fiber-F which has large specific surface area had the largest rate constant in all fibers. These results and Table 1 indicate that the fibers with

large surface area per unit volume have a tendency to have large rate constants. Table 2 also shows the rate constants of silver silica gel, silver zeolite, silver alumina and granular activated charcoal for comparison. The rate constant of Fiber-F was about 2, 7, 5, 3 times as large as that of activated charcoal, Ag-zeolite, Ag-silica gel and Ag-alumina, respectively.

2. CHARACTERISTICS OF ACF FILTER

SPECIFICATION OF ACF FILTER The ACF filter was made of knit of Fiber-F impregnated with 6 wt% TEDA based on the above results. It was 60 mm in diameter, 2.5 g in weight and 4 mm in thickness. The influence of the distribution of activity in the filter on a geometric counting efficiency could be neglected because of the thin thickness. The filter has mechanical strength enough to a practical use and the fragments of the fiber do not come off.

PRESSURE DROP BY ACF FILTER The relationships between the pressure drop and the flow rate were investigated for the ACF filter, ACC(Toyo Sangyo Kagaku, CHC-50), ACP (Toyo Sangyo Kagaku, CP-20) and a glass filter paper for aerosol sampling(GFP). The pressure drop of ACF filter was 14 mmAq at 50 liter/min(face velocity: 42.5 cm/s). It was 1/14, 1/10 and 1/7 of ACP's, GFP's and ACC's, respectively.

EFFECT OF RELATIVE HUMIDITY ON COLLECTION EFFICIENCY Effect of RH on the collection efficiency of the ACF filter for methyl iodide was measured. The relationships between collection efficiencies and RH are shown in Figure 2 comparing with the results for ACP. The efficiencies for both the ACF filter and the ACP decreased with increase of RH. The collection efficiency of the ACF filter was 67 % at 90 % RH, which was about 7 times as high as that of ACP. The collection efficiencies for 8 h sampling time were 50, 65 and 88 at 90, 75 and 60 % RH, respectively. With increasing sampling time from 1 h to 8 h, the collection efficiency decreased by only a few percent at 60 % RH and by 15 % at 90 % RH.

PRACTICAL USE The ACF filter was practically applied to collect airborne radioiodine in a ventilation air of a research facility. Sampling conditions were as follows: sampling time of 24 or 48 h, RH of 34 to 78 % and sampling flow rate of 50 liter/min. The collection efficiencies of the ACF filter were 92 to 93 % and 85 to 91 % for the sampling time of 24 h and 48 h, respectively. On the other hand, the collection efficiencies of the ACP were about one fourth of those of the ACF filter. It is proved that the ACF filter has excellent performance for practical use.

CONCLUSIONS

To find a suitable activated carbon fiber as an adsorbent for radioiodine, the collection efficiency for methyl iodide was examined for six kinds of fibers varied in specific surface area, pore diameter, etc. Based on the results, ACF filter with 60 mm in diameter and 4 mm in thickness was made of knit of Fiber-F impregnated with 6 wt% TEDA. The ACF filter has mechanical strength enough to a practical use and fragments of the fiber do

not come off. The pressure drop by the ACF filter was 14 mmAq at a flow rate of 50 liter/min, which was 1/14 of that by the ACP. The ACF filter was practically applied to collect airborne radioiodine in a ventilation air of a research facility. The collection efficiencies of the ACF filter under flow rate of 50 liter/min were 92 to 93 % and 85 to 91 % for the sampling time of 24 h and 48 h, respectively.

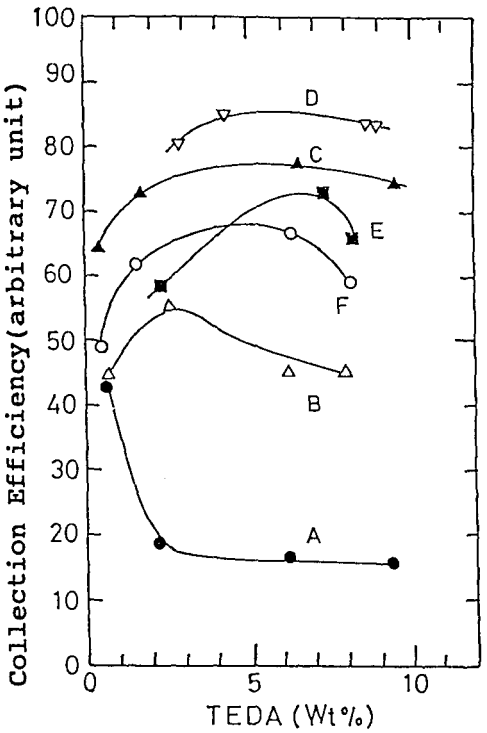

Fig.1 Effect of impregnation amount of TEDA on collection efficiencies of ACFs for methyl iodide at flow rate of 50 liter/min, RH 90 % and 60 min sampling time.

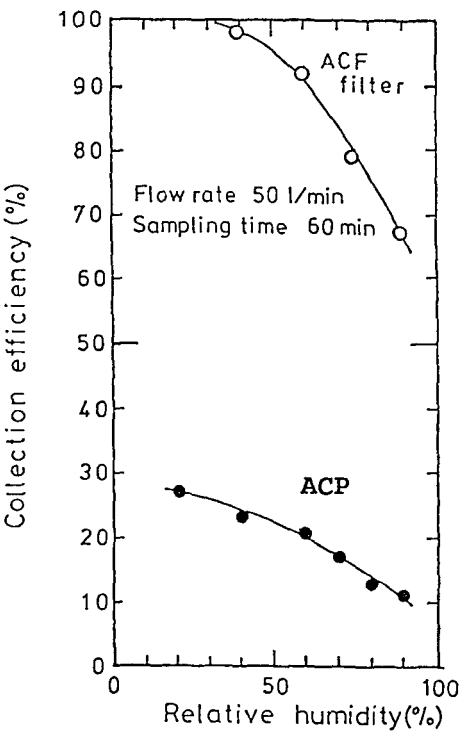

FIG.2 Effect of relative humidity on collection efficiencies of ACF filter and activated carbon filter paper(ACP).

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FIRST RESULTS OF TESTING A STEPWISE ROTATED AEROSOL FILTER SYSTEM FOR ENVIRONMENTAL MONITORING

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INTRODUCTION

The monitoring of the gross beta aerosol activity released from nuclear facilities is of great importance for environmental surveillance. In order to measure this activity by one monitor, both during normal operation (down to 10^{-2} Bq/m^3) and in the case of an accident (up to 10^6 Bq/m^3) this device must cover a range of more than eight orders of magnitude. Moreover this equiqment should not be too expensive in construction and operation, it should be very simple in regard to the sampling technique and it should comply with all requirements made by national regulations and modern data handling. Until now no such monitor has been available. In the past few years the Nuclear Research Center Jülich has developed a stepwise rotated aerosol filter system for environmental monitoring. The system and the first results of its tests will be reported in this paper.

BASIC CONSTRUCTION AND SPECIFICATIONS

The basic construction of the stepwise rotated filter system is shown in Fig. 1. The air enters from the right. The filter is contaminated in the contamination chamber, from where the air goes to the pump. There are 43 filter spots, 41 to measure the air concentration, one for the calibration of the detectors, and one for the background measurements. Three end-window detectors measure the contaminated filter spots: D1 during contamination, D2 immediately after contamination and D3 after the decay of

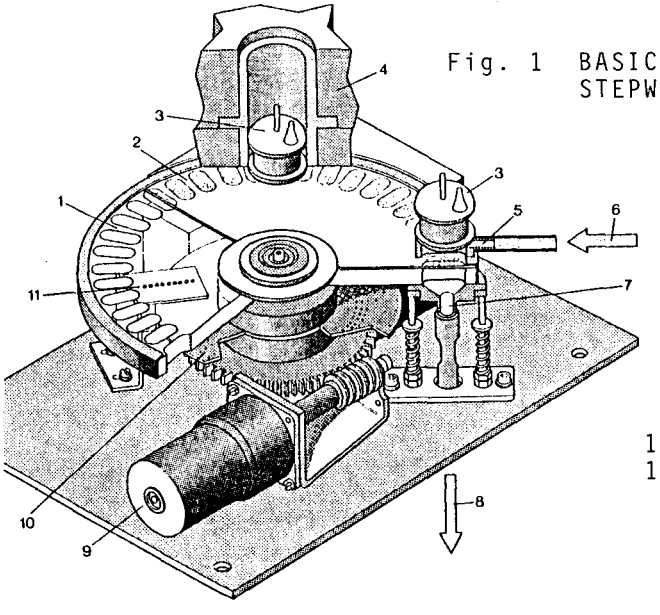

Fig. 1 BASIC CONSTRUCTION OF THE STEPWISE ROTATED FILTER SYSTEM

1 rotated disc with holes
2 filter
3 end-window detector
4 shielding
5 contamination chamber
6 air input
7 suck-in device
8 to the pump
9 motor to rotate the disc
10 coded position disc
11 position control

the short-lived natural activity. To guarantee an operation without interruption two such devices form one unit. For a good decay analysis of the detector D2 a contamination time of each of the filter spots should last one day at least. Various criteria cause software to initiate a rotation of the filter under the chosen detector. The air passes at 1 m^3/h through the filter spot of 6 x 20 mm^2. The rotating disc has a diameter of 200 mm. The measured specification of the three detectors, their efficiencies, their backgrounds and their maximum possible rates are given in Table 1.

detector VALVO 1441	efficiency	background	maximal possible rate
		pulses per second	
D1	0.13	0.1	10^4
D2	0.20	0.1	10^4
D3	0.20	0.1	10^4

Table 1 MEASURED SPECIFICATIONS OF THE DETECTORS

THE FUNCTION OF DETECTOR D1

During normal operation detector D1 measures, as a rule, only the natural radioactivity, which is not constant. In Fig. 2 the measured values of a stepwise moved filter tape (curve S1) and of the stepwise rotated filter system (curve D1) are compared. The measurements are carried out during the same contamination period. The differences are caused by the different types of filter paper (S1: charcoal contaminated filter paper; D1: glass-fiber filter) and the amount of air pumped through the filter spots. The air flow of the stepwise moved filter tape is higher by a factor of 10, which is also the reason for the reduced air flow after about 10 hours contamination time. If the curves are normalized they are identical. From the shape of a "mean curve" and its errors a signal is to be derived, if artificial radioactivity is in the air. But this is still under development.

In the case of an accident the detector D1 must be able to measure every possible high concentration. From the pump rate, the detector efficiency, the maximal possible pulse rate and a contamination period of 3 min the maximum measurable concentration of 1.5 10^6 Bq/m^3 is calculated. If we also operate the tube in the non-linear part of the tube characteristic the sampling time or the measurable concentration can be greater by a factor of about 6. Taking into account all possible filter spots (2 x 41), this device is able to operate for about one day without changing the filters. Moreover, the development of the air contamination is documented by the sequence of the contaminated filter spots, which may be later measured by gamma spektrometry in the laboratory. In the case of an accident the rotation of the filter is controlled by the software and a given maximum pulse number.

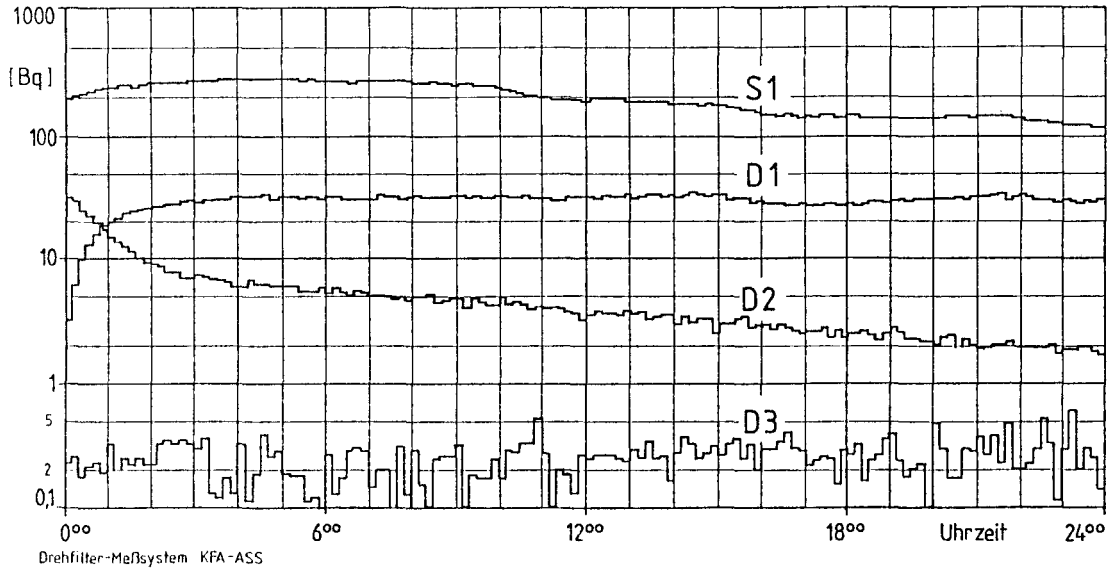

Fig. 2 MEASURED CURVES OF THE STEPWISE MOVED FILTER TAPE (S1), THE STEPWISE ROTATED FILTER SYSTEM (D1), THE DECAY CURVE (D2), AND THE LONG-LIVED RADIOACTIVITY (D3)

THE ANALYSIS OF DETECTOR D2

In accordance with national requirements, in environmental surveillance a filter is normally contaminated over a 14-day period. After the decay of the short-lived natural nuclides the filter is measured in the laboratory. The results are available about 20 days after the beginning of contamination. In our new device, the detector D2 starts the analysis about 3 hours after the end of the contamination period (Fig. 2). The results are available at the end of the second day after the beginning of contamination. By this procedure a contamination of a long-lived nuclide with a concentration of about 0.05 Bq/m^3 can be analyzed. An example of such an analysis in shown in Fig. 3.

Analyse Aerosole - Abklingverhalten
(Analysis of aerosol radioactivity decay)

Gerechnet bis 1421 Minuten = 23.69 Stunden		calculated up to 1421
Ausgewertete Datensätze = 142		min = 142 measurements
Pb 214 (Ra B)=	3.36 Bq ± 7 %	
Pb 212 (Th B) =	1.12 Bq ± 23 %	
Langl.Anteil =	0.61 Bq ± 19 %	long-lived activity

Fig. 3 NUMERICAL RESULTS OF A DECAY ANALYSIS WITH DETECTOR D2

This analysis gives the first indication of artificial air contamination within 2 days. This result has to be confirmed by the measurements of detector D3.

MEASUREMENT OF THE LONG-LIVED RADIOACTIVITY BY DETECTOR D3

The long-lived radioactivity deposited on the filter is measured by detector D3 after a decay period of 5 days for natural short-lived radioactivity (Fig. 2). The detection limit of about 10^{-4} Bq/m^3 is determined by the detector efficiency, the well-known background, and the measuring time of 24 hours. Our tests have shown that it is necessary to pump off the air of the volume around the detectors to get a low and constant background which if possible is not influenced by the concentration of radon and its daughters. The detection limit achieved is more than one order of magnitude lower than the requirements of our national regulations.

CONCLUSION

The reported results have been obtained by one prototype of a stepwise rotated filter system, which has been in operation under realistic conditions for more than six months. For the interpretation of the data a simplified software was used. The tests have shown that calculated detection limits and a measuring range of eight orders of magnitude are achieved. This means that the stepwise rotated filter system is a good monitor for environmental surveillance, both for normal operation and in the case of an accident.

The analysis of the measurements of detector D2 indicates an artificial air contamination at the end of the second day. This is much faster than the results of the environmental surveillance procedure normally used, which are available about 20 days after the beginning of the contamination of the filter.

In 1988 seven such units are to be built to replace the old stations of the inner surveillance ring of the Nuclear Research Center Jülich, which is now equipped with stepwise moved filter tapes. Operation of these new stations with the central processor and a sophisticated software has still to be tested.

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IMPROVEMENT OF TRITIUM ENRICHMENT ELECTROLYSIS CELL FOR ENVIRONMENTAL MONITORING

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INTRODUCTION

The measurement of Tritium in environmental water sample has being carried out for the monitoring of the contaminated level by effluent from the atomic institution and the investigation in geochemistry, e.g. streams of water under ground. These measurements have need of the pre-concentration of HTO by any method. In recent years, some methods have been discussed for these purposes, e.g. electrolysis and thermal diffusion. In these methods, the electrolysis has been employed generally, because of compact apparatus, easy handling and possibility of certaion results.

Where, on the pre-concentration, more correct data of Tritium activities can be gotten by using large volume of water sample. But, it requires a long time and large electric energy. Up to this time, one pair of slender electrodes made of materials of nickel or iron has been employed. However, for such electrodes, the re-adjustment of electric current should be made to make itself agreeable to change of the electrode surface area in sample water, because of decrease of effective surface area of electrode, as reduces gradually by electrolysis.

In this study, the electrolysis glass cell of its volume 100 ml inserted multi plate electrode was designed and produced, and, its characteristics for concentration of HTO and conditions of procedure e.g. required electric current density and time for electrolysis, were clarified.

EXPERIMENTAL

1. Apparatus and procedure for pre-concentration

As illustrated in Fig. 1, as for the electrolysis cell, the bottom of cylindrical hard glass vessel which was dismeter of 40 mm and length of 200 mm was formed into rectangular shape (size ; 50 mmH x 40 mmW x 15 mmD). And the multi plate electrode consisting of five plates (size ; 50 mmH x 35 mmW, thickness; 1.0 mm, combination of nickel and iron) were set in the part. The effective total area of the electrode was about 60∿62 cm^2.

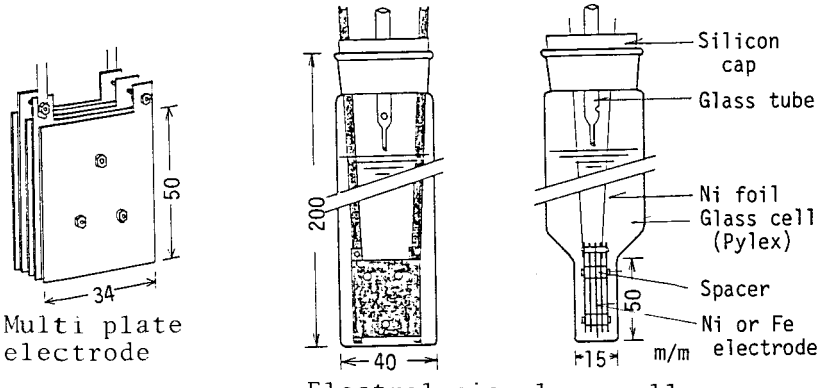

Fig. 1 Diagram of electrode and electrolysis cell

The experiments were carried out setting five cells connected in the electric circuit into the controlled low temperature water bath (2∿5°C). The individual cell held the sample water of 100 ml. The sample water in each cell decreased to about 15 ml by electrolysis of current density between 50∿280 mA/cm^2.

2. Measurement of Tritium and calculation of recovery rate

The experiments on pre-concentration efficiency were carried out using slightly activity Tritium solution (3 x 10^{-3}μCi/ml). Samples of 8.0 ml before and after electrolysis were pipetted out and infused into 20 ml vials with liquid scintillator Triton-100X (Packard Co., Ltd.)12 ml. Radioactivities of each vial were counted using low back ground liquid scintillation counter TRI-CARB 3255 type (back ground; 10 cpm, counting efficiency; 25%). Recovery percent (R) of HTO are calculated with next formula.

$$R\ (\%) = (Cf \times Vf) \times 100 / (Ci \times Vi) \qquad (1)$$

where, Ci and Cf ; radioactivities of Tritium before anf after electrolysis, Vi and Vf ; volume of sample water before and after electrolysis.

RESULT AND DISCUSSION

1. Decreasing rate of sample water by electrolysis

If electro-chemical change of electro-materials is disregarded, decreasing rate of sample water by electrolysis are calculated from total electric current used with Faraday's constant.

The solid lines in Fig. 2 show the decreasing rate in the range of 80∿280 mA/cm^2 in current density. When water volume decrease to 26 ml, the top of electrode appears on the surface of sample water. Therefore, the electric current must be re-adjusted in a few hours to responded value according to electrode surface area under water. Therefore, decreasing rates of sample water is slightly late.

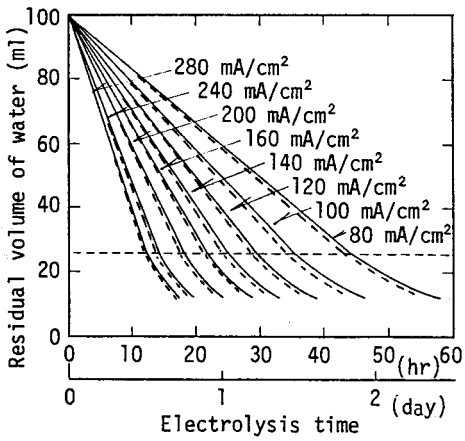

Fig. 2 Decreasing of sample water by electrolysis

The required time to decrease the sample water from 100 ml to 15 ml by electrolysis is 16∿57 hours with above current density range. Under the higher current density, the required time is shorter. But, as show in the next clause, the higher current density, being the more loss of HTO. Then, for the purpose of measurement with a high accuracy, it needs the definite knowledge to divide the time to pre-concentrate HTO and to count the Tritium activities.

While, broken lines shown in Fig. 2 are decreasing rates of sample water obtained experimentally. These rates are more rapid than that obtained by calculation, because of loss of sample water yield in electrolysis by volatilization.

2. Correlation of Tritium recovery and current density

It has been reported that the optimum current density on the electrolysis concentration was 100 mA/cm^2. The value was determined by experimental results using one pair electrode cell of cathode and anode.

Then, correlations of recovery rate of Tritium and current density used multi plate electrode cell are different with a kind of electrode materials as shown in Fig. 3. The recovery rate is higher in range 60 100 mA/cm^2 used Ni-Ni electrode. But, the range is wider of 60∿140 mA/cm^2 used Fe-Fe electrode. Still, in the range of current density over 100 mA/cm^2 , the rate using of Ni-Ni is lower. The other hand, the rate using Fe-Fe is not down.

Turning the attention only to recovery rate of HTO, employing Fe-Fe electrode leads to more effective than Ni-Ni. But, if employ iron plate as anode, its defect is more corrosionable. Using nickel plate as cathode and iron plate as anode, the level of HTO recovery is intermediate of those values of Ni-Ni and Fe-Fe.

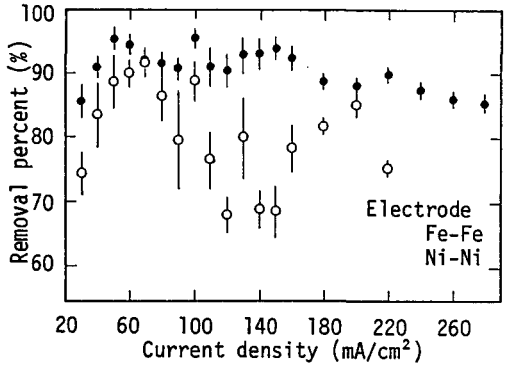

Fig. 3 Removal rate of HTO

While, as comparison of a kind of materials, the recovery percentages at the next conditions, current density ; 180 mA/cm^2, low temperature water bath ; 2 °C, shown in Table 1 .

Table 1 Tritium recovery rate by electrode of Fe-Fe, Fe-Ni, Ni-Ni

	Electrode		
	Fe-Fe	Fe-Ni	Ni-Ni
Tritium recovery (%)	90	85	75

3. Required time for Tritium measurement

On the measurement of Tritium activities, required total time is consistent for 1) Pre-treatment (e.g. oxidation of organic substance in sample and evaporation etc.), 2) Pre-concentration of HTO, 3) Counting of Tritium activity.

In above three terms, 1) is same in any sample and measurement condition. 2) and 3) are different with following conditions, that activity level of sample, counting conditions (e.g. counting efficiency and back ground level etc.), pre-concentration effect (recovery and concentration rates of HTO etc.), and expected level of precision of data. Therefore, at the conditions of measurement that expected precision levels as expressed in fomula (2) are ±10% or ±20%, Tritium concentration levels about 10, 20, 50 and 100 pCi/l in samples and using liquid scintillation counter mentioned at experimental term, the required total time to pre-concentrate and count of the sample were calculated as shown in Fig. 4 .

$$\{(Ns/t)+(Nb/t')\}^{1/2} \times 100 \Big/ [\{(Ns \pm \sqrt{Ns})/t\} - \{(Nb \pm \sqrt{Nb})/t'\}] = 10 \text{ or } 20 \qquad (2)$$

where, Ns and Nb ; counts of sample and back ground, t and t' ; counting time of those.

Solid lines shown in Fig. 4 are required time for pre-concentration of HTO.

As a result, for the sample of higher Tritium level and the higher permissible level of statiatial error, it needs longer time to pre-concentrate than that to count. As a concrete example, setting up the current density of 180 mA/cm^2 and expected precision level of ±10% for Tritium level in the low sample of 10 pCi/l , it needs for 20 and 10 hours to pre-concentrate and count.

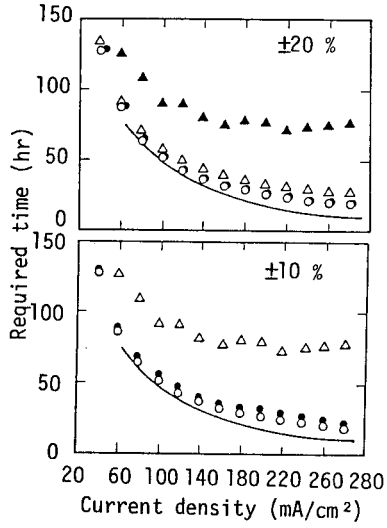

Fig. 4 Required time to pre-concentrate and count

Tritium activity level(pCi/l)
▲ 10, △ 20, • 50, ○ 100

CONCLUSION

On HTO pre-concentration, following results were obtained using multi plate electrode with effective surface area 60∿62 cm^2 and sample capacity 100 ml cell. 1) At the condition of current density 60∿140 mA/cm^2, recovery rates were more than 90% using Fe-Fe electrode, 2) Fe-Ni electrode are more corrosion-resistant, but HTO recovery rates are lower, 3) In the following conditions of the measurement with precision level of ±10% for low level Tritium sample as 10 pCi/l and current density of 180 mA/cm^2 , the required time to pre-concentrate and to count are 20 and 10 hours.

MEASUREMENT OF PLUTONIUM ISOTOPIC ACTIVITY RATIO IN LOW LEVEL PLUTONIUM SAMPLES

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INTRODUCTION

In the usual environmental plutonium monitoring around atomic energy plants, such as nuclear fuel reprocessing factories and Atmic power stations, plutonium is chemically extracted from environmental samples in question, purified radiochemically, and electrodeposited on a metallic plate, than α-ray spectrum of the electrodeposition sample is collected and α-activities of $^{239+240}Pu$ and ^{238}Pu of the environmental sample is determined. Alpha-ray energies of ^{239}Pu(5.16 MeV) and ^{238}Pu(5.17 MeV) are so close to each other that α-activities of the two nuclides can not be distinguished.

It is well known that nuclear test explosions and the burn-up of SNAP-9A have caused global plutonium contamination. Information about the plutonium isotopic ratio, $^{239}Pu/^{240}Pu$, gives possibility to presume whether plutonium found in environmental samples has originated from atomic energy plants or from global fallout. Measurements of the plutonium isotopic ratio in environmental samples using nuclear emulsion method[1], α-X ray method[2]and fission track method[3], have been attempted, but these methods were found to be insufficient in the sensitivity. Beside these methods, the Lγ/α-ray activity ratio has recently been used to measure the $^{240}Pu/^{239}Pu$ ratio in environmental samples[4].

PRINCIPLE OF THE METHOD

Plutonium extracted from an environmental sample in quesion and a standard plutonium, the isotopic ratio of which is well known, are both electrodeposited on a metallic plate, respectively, and α-ray spectra of these electrodeposition samples are collected to measure α-counting rates of $^{239+240}Pu$, $\alpha(^{239+240}Pu)_{sample}$ and $\alpha(^{239+240}Pu)_{stand.}$.

Then, these electrodeposition samples are altogether irradiated by thermal neutrons, and γ-ray counting rates of ^{99m}Tc, $\gamma(^{99m}Tc)_{sample}$ and $\gamma(^{99m}Tc)_{stand.}$, are measured.

The $^{239}Pu/^{239+240}Pu$ activity ratio in the environmental sample, $R(^{239}Pu/^{239+240}Pu)_{sample}$, is determined by the following equation.

$$R(^{239}Pu/^{239+240}Pu)_{sample} = \frac{\gamma(^{99m}Tc)_{sample}}{\alpha(^{239+240}Pu)_{sample}} \times \frac{\alpha(^{239+240}Pu)_{stand.}}{\gamma(^{99m}Tc)_{stand.}} \times R(^{239}Pu/^{239+240}Pu)_{stand.} \quad \cdots\cdots\cdots(1)$$

where $R(^{239}Pu/^{239+240}Pu)_{stand.}$ is $^{239}Pu/^{239+240}Pu$ activity ratio of the standard plutonium, and both $\gamma(^{99m}Tc)_{sample}$ and $\gamma(^{99m}Tc)_{stand.}$ are the counting rates converted to the values at the end of the neutron irradiation.

EXPERIMENTAL

Chemical procedures for plutonium extraction from environmental samples, radiochemical purification of the extract and electrodeposition on a metallic plate were carried out mainly according to the mannual issued from the Science and Technology Agency, Japan.[5)]

(I) NBS standard plutonium

Plutonium isotopic abundance of the NBS standard plutonium, which was used as the standard reference in the present study ($R(^{239}Pu/^{239+240}Pu)_{stand.} = 0.7384$).

(II) RCC standard plutonium

Another standard plutonium made by RCC was also electrodeposited on a nickel plate in the same manner as stated above. The contents of ^{238}Pu and ^{240}Pu of this standard plutonium are of negligibly low level.

(III) Environmental samples

Two environmental samples described below were investigated in the present study.

(a) Pit sample(Mud taken from a rain water pit in our laboratory. 1.6 kg. Less than 2 mm in diameter. (b) Sediment sample(Sea sediment, which was collected in the middle of the Nyu Bay, Fukui Prefecture. 240 g. Less than 0.5 mm in diameter.

It was found that, when an electrodeposition nickel plate was irradiated by neutrons, activities induced in the nickel plate made much interference to ^{99m}Tc γ-counting. In order to avoid this difficulty, deposited plutonium film on the nickel plate was wiped off with a piece of cellulose fiber filter paper immersed in hot nitric acid and the filter paper piece was neutron irradiated. Neutron irradiation was made in the Kyoto University Research Reactor(5MW, thermal neutron flux: $X10^{13}n.cm^2.sec^{-1}$, 1 hour irradiation).

RESULT AND DISCUSSION

In Table 1 are shown measured values of $\alpha(^{239+240}Pu)$ and $\gamma(^{99m}Tc)$.

Sample	α ($^{239+240}Pu$) (cpm)	γ (^{99m}Tc) (cpm)	
		Neutron irradiation I	Neutron irradiation II
P - 1	0.47(2.56)	3.61(48)	
P - 2	1.39(1.09)	4.35(17)	
P - 3	2.13(0.54)	5.10(26)	
S - 1	0.56(1.77)		5.22(27.9)
S - 2	1.22(0.66)		6.19(17.0)
S - 3	2.34(0.75)		8.72(25.5)
RCC-1	17.30(0.69)	53.6(6.5)	
RCC-2	30.46(0.56)	107.0(8.3)	
RCC-3	44.91(0.95)	150.0(5.1)	
RCC-4	88.97(0.64)	281.0(2.6)	
NBS-1	16.69(0.72)	44.9(9.9)	44.8(11.9)
NBS-2	34.94(0.59)	85.5(12.4)	102.5(5.2)
NBS-3	70.46(0.66)	173.0(3.6)	214.5(4.7)
NBS-4	89.23(0.72)	220.0(4.8)	263.8(5.0)

( ): Relative standard deviation in %

Table 1 $^{239+240}Pu$ and ^{99m}Tc counting rate of the samples.

Wipe-off samples of P-1~3, RCC-1~4 and NBS-1~4 were neutron irradiated altogether (Neutron irradiation I in Table 1). About six months later, wipe-off samples of S-1~3 and NBS-1~4 were neutron irradiated altogether (Neutron irradiation I in Table 1). About six months later, wipe-off samples of S-1~3 and NBS -1~4 were neutron irradiated altogether, when activities in the NBS samples at Neutron irradiation I had decayed to be negligibly small (Neutron irradiation II).

Relation between $\alpha(^{239+240}Pu)$ and $\gamma(^{99m}Tc)$, which were found for the NBS samples and the RCC samples are shown in Fig. 1, and those which were found for the pit samples and the sediment samples, are shown in Fig. 2.

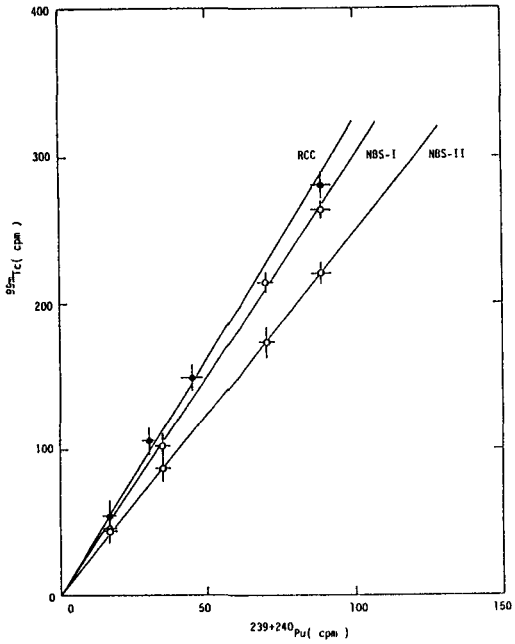

Fig. 1 Standard plutonium.

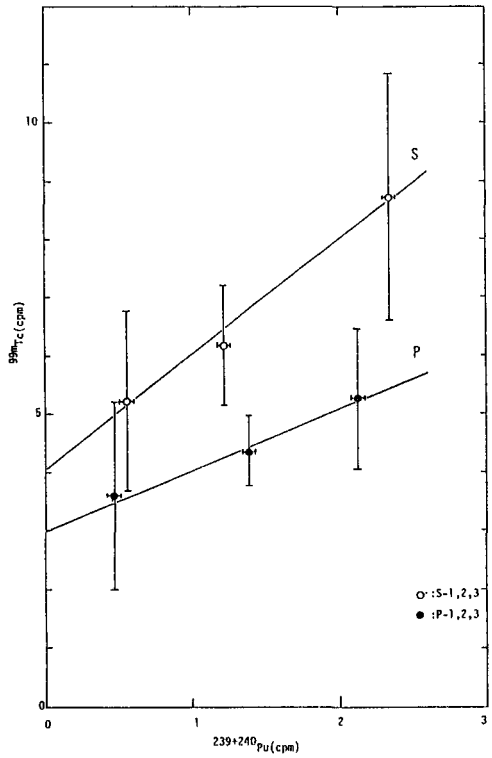

Fig. 2 Nyu Bay sediment and pit samples.

Here, it was assumed that the $\gamma(^{99m}Tc)$ values corresponding to zero $\alpha(^{239+240}Pu)$, namely, 4 and 3 cpm, for the sediment samples and the pit samples, respectively, are ascribed to uranium contamination.

In Table 2 are shown $^{239}Pu/^{239+240}Pu$ ratio values, which were calculated on the basis of the results shown in Table 1 and on the assumption stated above. Ratios of $^{238}Pu/^{239+240}Pu$ are also shown in the table.

Kray et al. reported that the $^{239}Pu/^{239+240}Pu$ ratio in the recent global fallout ranges from 0.5 to 0.8 by the use of mass spectrometer method.[6] Hisamatsu et al.[2], Nakanishi et al.[3] and Komura et al.[4] reported that the ratio in sediment and soil ranges from 0.4 to 0.6 by the use of α-X ray method and fission track method.

The ratio in the sediment sample determined in the present

study(~0.6)is in the range stated above. But, the ratio determined for the pit sample(~0.3) is somewhat smaller.

Sample	$^{239+240}Pu$ (Bq) X10^{-2}	^{238}Pu (Bq) X10^{-2}	$^{239}Pu/^{239+240}Pu$ (Bq/Bq)	$^{238}Pu/^{239+240}Pu$ (Bq/Bq)	^{137}Cs (Bq/kg)
P - 1	2.57 ± 0.07	0.17 ± 0.02	0.39 ± 0.18	0.066 ± 0.007	
P - 2	7.55 ± 0.08	0.70 ± 0.03	0.29 ± 0.05	0.093 ± 0.005	64.1 ± 3.3
P - 3	11.48 ± 0.07	1.00 ± 0.02	0.29 ± 0.08	0.087 ± 0.002	
S - 1	3.03 ± 0.05	0.07 ± 0.02	0.51 ± 0.14	0.023 ± 0.006	
S - 2	6.63 ± 0.05	0.13 ± 0.02	0.45 ± 0.08	0.020 ± 0.003	21.5 ± 3.3
S - 3	12.68 ± 0.10	0.33 ± 0.02	0.53 ± 0.15	0.026 ± 0.001	
RCC-1	93.72 ± 0.65	~ 0	0.91 ± 0.09	~ 0	
Rcc-2	164.98 ± 0.92	~ 0	1.03 ± 0.12	~ 0	------
RCC-3	274.97 ± 2.32	~ 0	0.98 ± 0.09	~ 0	
RCC-4	481.92 ± 3.08	~ 0	0.93 ± 0.08	~ 0	

Table 2 Activity ratios of $^{239}Pu/^{239+240}Pu$ and $^{239}Pu/^{239+240}Pu$ found in Nyu Bay sediment(S), pit sediment(P) and standard plutonium(PCC) samples.

The $^{238}Pu/^{239+240}Pu$ ratio found for the sediment sample(0.02 ~ 0.03) is relatively close to the ratio of 0.03 in global fallout7). Aginst this, the ratio found for the pit sample(~0.09) is considerably larger.

It appears that the main origin of the plutonium in the sediment sample is fallout from nuclear test explosions. Both of $^{239}Pu/^{239+240}Pu$ and $^{238}Pu/^{239+240}Pu$ ratios of the pit sample are different from those of global fallout. But, the main origin of the plutonium in the pit sample is also presumed to be nuclear debris. It is possible that the pit sample contain much of the old nuclear debris which were produced in Russian nuclear explosions carried out in 1961 to 1962. In our preceding study it was found that the $^{238}Pu/^{239+240}Pu$ ratios determined for fallout samples collected in 1961 to 1968 were rather high, ranging from 0.01 to 0.5[8].

It is desirable for raising accuracy of the present measurement method to get so much amount of electrodeposited plutonium that uranium contamination may be neglected. If the uranium contamination is as low as in the present study, at least 2 Bq of plutonium is necessary for getting sufficient accuracy.

The authors wish to thank Dr. M. Yosioka, Fukui Prefectural Institute of Public Health, for the kind present of valuable samples.

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ACTINIDE-IN-AIR MONITORING IN THE PRESENCE OF RADON & THORON

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This paper describes further developments of a computer code[1] which calculates the expected count rate in the plutonium channel of an actinide-in-air monitor from radon/thoron daughters. All the major pathways of radon/thoron into a building are allowed for; calculations are performed as a function of ventilation and recirculation rates.

OPERATION OF THE MONITOR

Commercially available continuous actinide-in-air monitors work by drawing air through a filter paper; this retains any particulate or free radon daughter activity. The filter paper is surveilled by a surface barrier semiconductor detector located in the incoming airstream a few millimetres above the deposited activity. In ideal counting conditions the pulse height spectrum would be composed of a number of narrow peaks corresponding to emitted α-energies. However, in the practical system the spectrum is composed of a number of broad bands because of random energy degradation: the α-particles travel across the air gap between filter paper and detector in random directions and therefore lose different amounts of energy. There are other possible sources of degradation including passage through filter fibres and any accumulated dust.

It is therefore not adequate simply to measure the count rate in a particular incident energy range; as can be seen from Figure 1, the overlap between the radon daughter spectrum and that from plutonium-239 is extensive. The only reliable way in which the actinide can be detected is to compensate, in some way, for the presence of the radon daughters. In the 3280 this is done by making a number of measurements at higher energies, to establish the radon daughter contribution, and using these to subtract an appropriate amount from the count rate measured in the actinide energy region.

Although this process works quite effectively there remains some advantage in controlling the radon level in the working environment. The model described here allows examination of the effects of various factors on the radon daughter count rate.

THE THEORETICAL BASIS OF THE MODEL

The individual radon daughter air concentrations are estimated, using source term specification procedures discussed below, with a steady-state version of the set of equations

derived by Porstendorfer[2]. These account for the effects of radon daughter attachment to particulates and to surfaces, detachment from particulates, radioactive decay and ventilation. They allow for constant sources of radon, such as direct exhalation into the room as well as entry with ventilation air. They have been extended to include recirculation and to allow for filtration of both makeup and recirculating air[1].

The steady-state activity of daughter j on the filter paper of a monitor sampling $V m^3 h^{-1}$ is:

$$S_j = \frac{V(C_j^{(a)} + C_j^{(f)}) + S_j - 1}{\lambda_j}$$

where $C_j^{(a)}$ and $C_j^{(f)}$ are the concentrations in air of attached and free particles (both are assumed collected by the filter paper and is the decay constant of daughter j. Since radon is not collected, $S_o = 0$.

The fraction of the counts from each alpha emitting daughter (of radon and thoron) which appears in the actinide energy region have been estimated using a relationship described in reference 1.

SOURCES OF RADON

The entry of radon with ventilating air, filtered or unfiltered, is included in the basic equation; the user may therefore specify the appropriate concentrations.

The model includes a number of options for describing radon ingress through floor intact slabs with and without damp proof membranes using diffusion theory. However the importance of penetrations as sources of radon has been recognised in recent years. The model includes a simple representation of these features although, as yet, it does not account explicitly for interactions with ventilation characteristics. Exhalation from walls is included, either through solution of the diffusion equation or by input of direct measurement information[3]

Thoron exhalation is included in the model; its representation is somewhat simplified because its short half life results in only a thin surface layer being effective.

THE IMPLEMENTATION OF THE MODEL

The basic computer code PLURAD has been implemented on a number of micro-computers; it is used principally on a Research Machines Ltd 380Z, a 64 Kbyte machine. The results are displayed as coloured graphs and these are reproduced on a plotter. A typical output from the program is shown in Figure 2.

CONCLUSIONS

The model includes most of the significant features with a background for actinide-in-air monitors. The computer program hsa proved useful in the evaluation of effectiveness of options for radon/thoron control in a number of active facilities. Although primarily designed for use in the analysis of problems associated with air monitoring in active facilities, where air change rates are generally high, it is adapable to a wide range of situations where airborne natural radioactivity is a concern.

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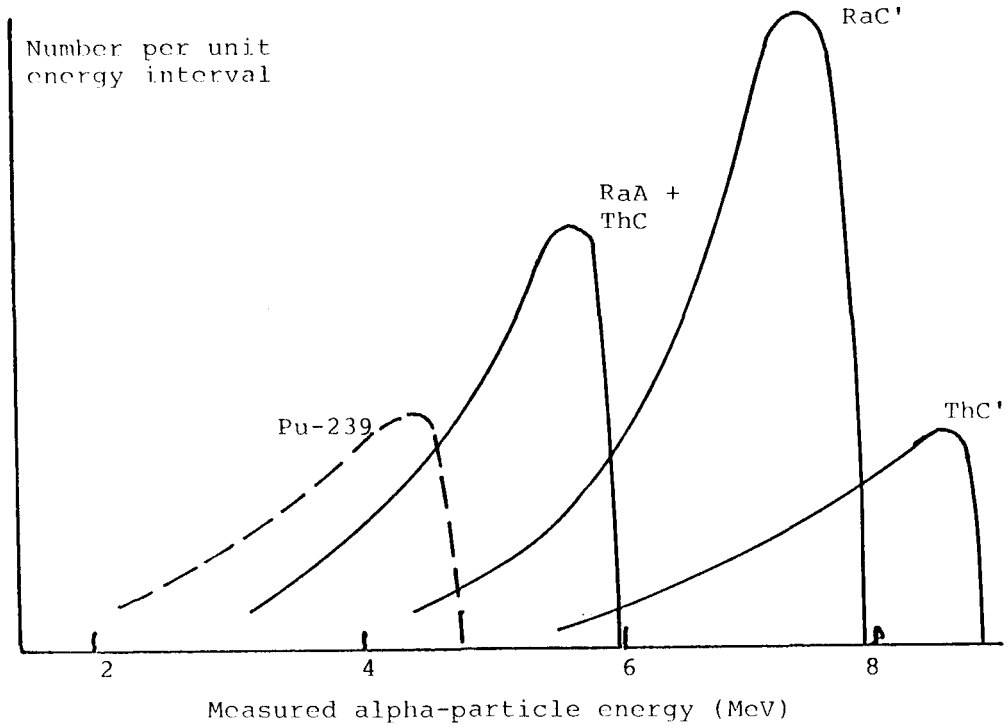
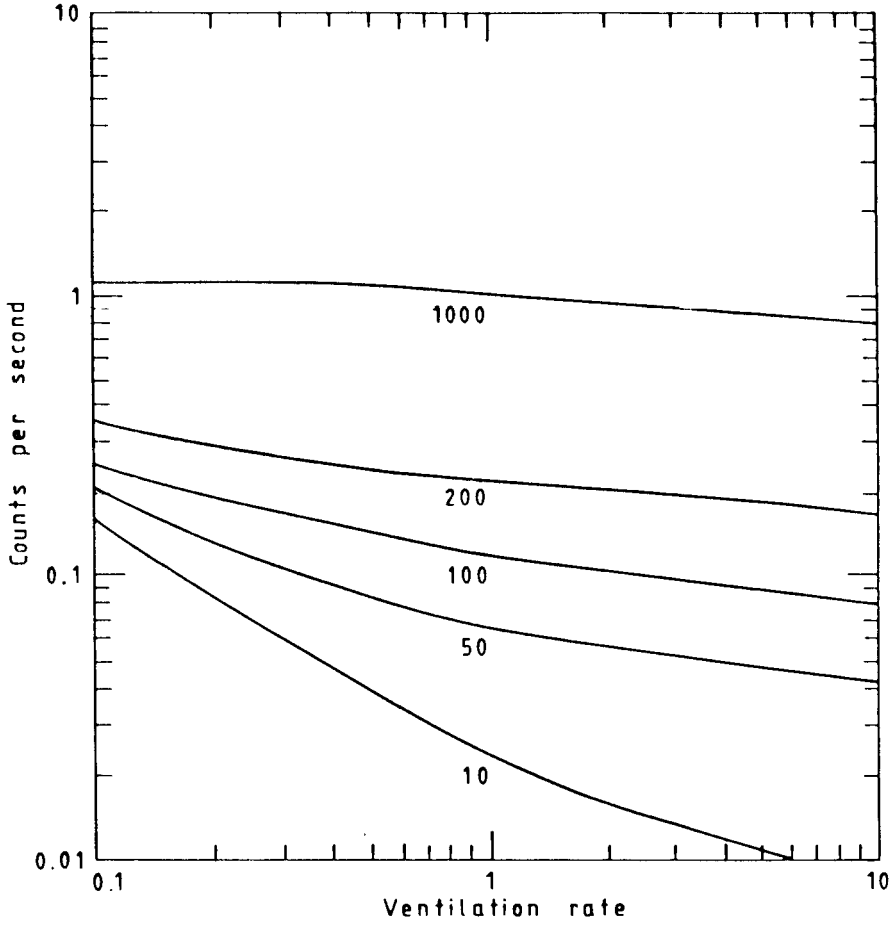
Figure 1: Illustrative Energy Distribution of Measured Alpha Particle Energies





Outdoor radon concentration = 10 − 1000 pCi m^{-3}
Outdoor thoron concentration = 10 − 1000 pCi m^{-3}
Recirculation rate = 0 hr^{-1}
Ventilated air is filtered : filter efficiency = 90%
Height of building = 10 m
Length of building = 30 m
Width of building = 20 m
Radon exhaling from walls = 200 pCi m^{-2} hr^{-1}
Thoron exhaling from walls = 400 pCi m^{-2} hr^{-1}
% radon passing through paint = 20
% thoron passing through paint = 0
Soil Ra − 226 concentration = 1 pCi g^{-1}
Concrete Ra − 226 concentration = 0.8 pCi g^{-1}
Floor thickness = 40 cm
Radon diffusion coefficient in dpm = 10^{-8} cm^2 s^{-1}
Thickness of dpm = 0.02 cm
Distance between dpm and surface = 15 cm
Thoron exhaling from floor = 200 pCi m^{-2} hr^{-1}

Figure 2: Count Rate in the Plutonium Channel of a Harwell Type 3280 Plutonium-in-Air Monitor from Radon & Thoron Daughters with Various Outdoor Radon/Thoron Levels. Incoming air is filtered.

QUALITY ASSURANCE OF ENVIRONMENT RADIOACTIVE MEASUREMENT IN CHINA

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ABSTRACT

This paper reported the work engaged in quality assurance of environmental radioactive measurements in China. It contains measurement intercomparison of environmental samples with Gamma Spectrometer for ten laboratories of Ministry of Nuclear Industry and quality control programme in investigation of natural radioactive level sponsored by EPA China. Examination on the analytical methods for soil samples and check on analyzed samples in environmental radioactivity level investigation for eleven provinces and three cities was also given in the paper.

Environmental sample reference materials which used as measurement intercomparison of environmental samples and examination samples and standard samples have been prepared in this work. The methods of intercomparison, examination and check for laboratories of MNI and many Environmental Monitor Central Station of Provinces were reported. Some questions in quality assurance were discussed.

DEVELOPMENT OF A MONITORING METHOD IN THE RADIATION FIELD UNDER THE GROUND

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*Radiation Center of Osaka Prefecture

I. INTRODUCTION

Environmental effects from a radioactive waste depository would ,if they could be occurred, be arisen extremely very slowly through water and soil. In the case, although detrimental effects on man and his environment from the depository would be negligible, it will be necessary to prove it by some measure to assure safety of the public substancially or psychologically. And it is also important to develop monitoring techniques around the facility from the view point of radiological protection.

As the first step toward the above purposes,the authors have discussed the informations on the radiation field in the soil through the measurements using TLD-set at several depth under the ground.

II. CONSTRUCTION OF DETECTOR AND MEASURING LOCATION

It is desirable for the detectors of evaluating the radiation field in the soil to fulfill the following requirements:

i) to be able to detect variations in the radiation field in the natural soil,
ii) to be able to set the detectors in the soil without a disturbance of their surrounding if possible,
iii) to keep stability of the detectors during the measurements in the soil,
iv) to be free from their maintenance during the measurements if possible,
v) to be inexpensive in cost of the detector for being distributed to many points, and so on.

To be taken these requriments into consideration, TLDs fulfill those comparatively, though they can only meausre integral dose. The detectors to be set in the soil were composed of (1)a brass tube (D_{outer}=15mmϕ, D_{inner}=12mmϕ,1 or 2m in length)for making energy dependence of the detectors flat, a methacrylate resin tube (D_{outer}=10mm , D_{inner}=5mmϕ),methacrylate resin rod as spacer and TLD-chips (denote Type A detector, hereafter) and (2)the detectors without brass tube (Type B) (see Fig.1). TLD-chips are CaSO Tm (UD110S,National Elect. Co.). The energy dependences for both types of detectors are given in Fig.2.

Both types of the detectors were buried vertically in the soil. For the detectors with 1m length, two pieces of

TLD-chips were set at 2.5,10,25,45,55,75 and 90 cm from the soil surface, respectively. And for that with 2m length, three pieces of TLD-chip were set at 5,40,80,120,160,and 195cm, respectively.

Four points, where detectors were buried, were chosen; 2 for 1m length and 1 for 2m length on the campus of the Nagoya University (denote Pts I,II and III,hereafter), and 1 point for 2m length in the field attached to the Faculty of Agriculture, Nagoya University (Pt.IV).
TLD-chips were exchanged almost every month, and monthly averaged exposure rates were measured.

A,B and C horizons were clearly found pedologically for Pts.I,II and III. For Pt.IV,however, A and B horizons were artificially removed and only C horizon was for our experimental depth.

III.RESULTS AND DISCUSSION

1)Monthly Variations in Exposure Rate

One of the examples for monthly variations in exposure rate for various depth in the soil (for Pt.I) is given in Fig.3. The general feature for each depth showed that monthly averaged exposure rates were decreased as the liquid water content in the soil increased. As seen in the Figure, the range of variations in exposure rate was 3μR/h and maximum at 2.5 cm depth. And that at 55 cm depth was almost constant. The exposure rates in the soil were generally varied in time.

From the present interval of the exchange of TLD-chips,one of the sources of variations in exposure rate was supposed to be largely related to the content of liquid water easily movable in the soil, that is, the changes in the bulk density of the soil due to the liquid water originated from precipitations.

2)Index of Liquid Water Content in the Soil

As previouly described, the authors have buried two types of the detectors (Types Aand B). For the exposure rates from both types of the detectors, the authors denoted exposure rates for type A by D_c ,and those for type B by D_N . Since the exposure rate of interest are considered to be proportional to the averaged photon flux density, the authors defined Apparent Flux Ratio (AFR)as a following equation,

$$AFR = (D_N - D_c)/D_c \times 100 \quad (\%). \qquad (1)$$

The AFR were related clearly to the amount of photon flux scattered by the liquid water in the soil. Then, there couldbe seen a negative correlation between the exposure rates in the soil D_c and the quantity AFR. This quantity was related to the amount of liquid water in the soil.

For Pt.I, the spread of the averaged values of exposure rates for each depth have been observed almost 0.7 to 1 μR/h as yet. Considering the spread originated from the change in the liquid water content, the authors estimated the spread expected by the change of liquid water content from 0 to 25% for dried soil of interest. The comparison of the above results with the observed ones is given in Fig.4. Since both results were comparatively coincided, the main source of variations in exposure rates was probably arisen from the changes in liquid water content in the soil.

The change in exposure rates due to wter in the soil could be estimated through the present study.
For fallout case, the TLD-chips set at shallow depth would first detect the change in exposure rate. On the contrary, those set deeply would show higher dose, if they would be originated from the deep sources.

The averaged exposure rates of 8 to 10 μR/h were observed in the natural background soil within our experiments. Some 20 or 25 % of seasonal variations were additional to these values. If further changes in exposure rate of 1 μR/h were assumed to be detectable, the changes originated only from ^{137}Cs would be corresponded to about 2 pCi/g of soil at present, assuming 1.25 g/cm^3 of dry soil density and 15% of liquid water content. Improveing the detector configuration, the detectable changes could be reduced.

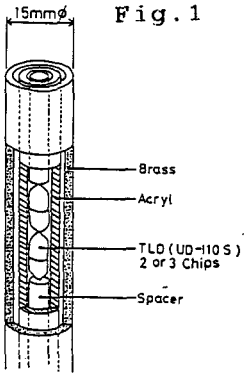

Fig.1

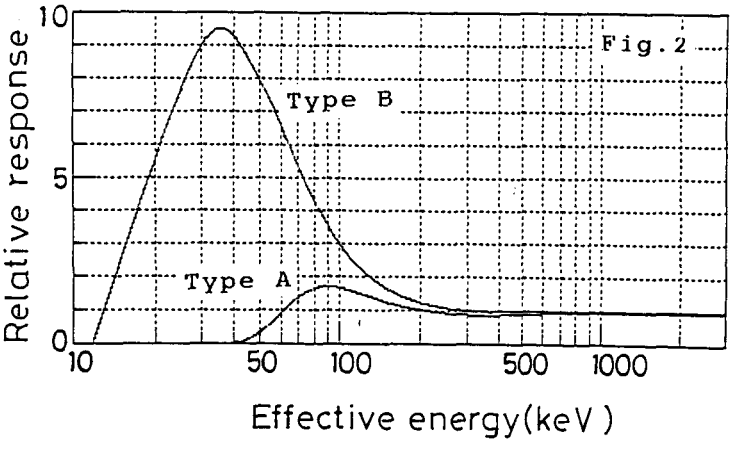

Fig.2

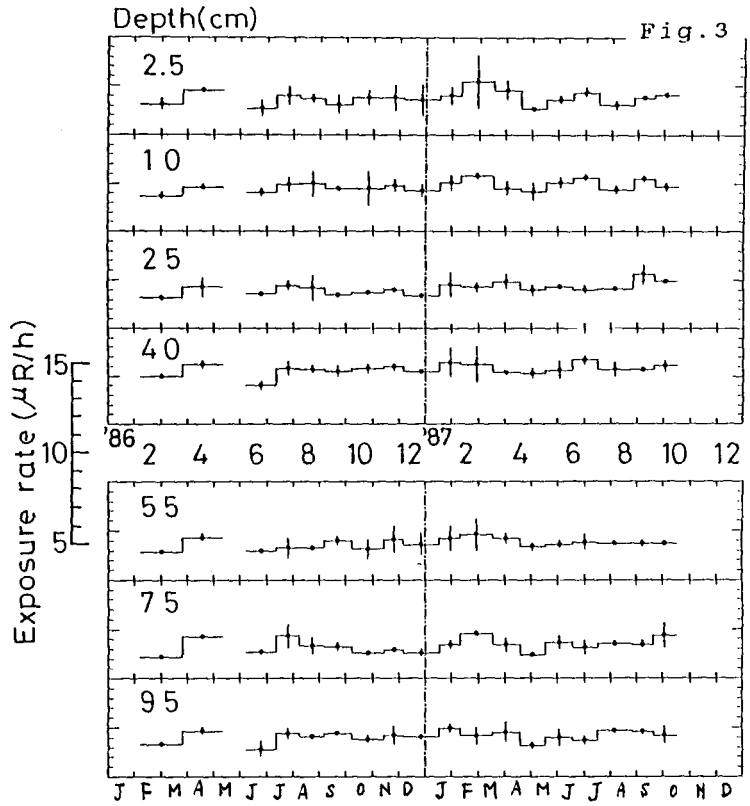
Depth(cm)
2.5
10
25
40
55
75
95
Exposure rate (μR/h)
15
10
5
'86
2 4 6 8 10 12
'87
2 4 6 8 10 12
J F M A M J J A S O N D J F M A M J J A S O N D

Fig.3

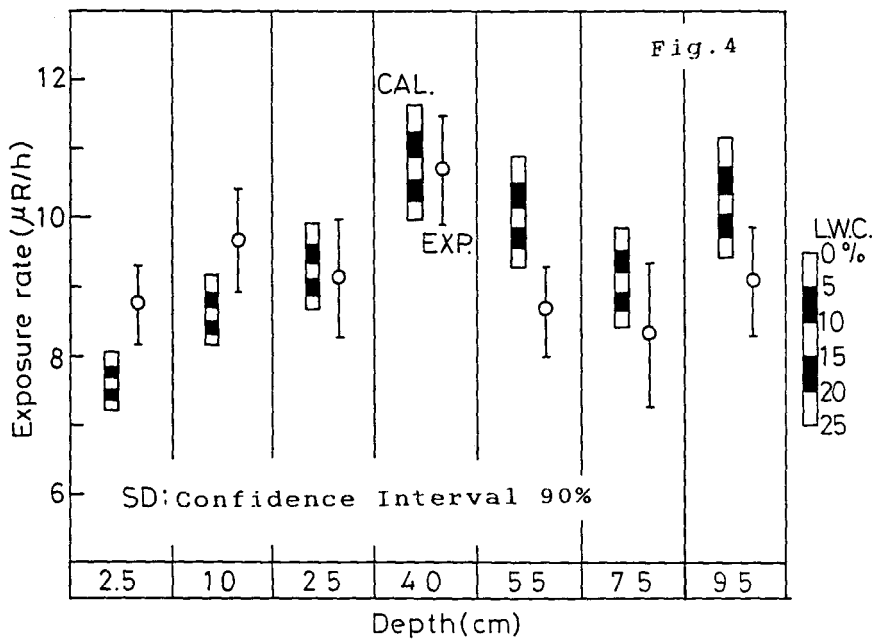
12
10
8
6
Exposure rate (μR/h)
CAL.
EXP.
SD: Confidence Interval 90%
2.5
10
25
40
55
75
95
Depth(cm)
LWC.
0 %
5
10
15
20
25

Fig.4

NETWORK OF RADIATION MONITORS (NORM) AT KEK AND ITS PRESENT STATUS

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INTRODUCTION

The NORM, Network of Radiation Monitors, had firstly constructed in 1980 with a center, 3 sub-stations and 48 radiation monitors. The basic conditions for the construction of the NORM were as follows: the first is to ensure the large extension of the number of both the sub-station and the radiation monitor, the second to invest the selection of radiation detectors with the large flexibility so as to allow stepwise development of those, the third to unify the assemblage of radiation detectors and electronic circuits, and the fourth to simplify the connection between a sub-station and a radiation monitor. The NORM has largely expanded for 7 years. Here we describe the NORM system in detail.

ROLES OF NORM AT KEK

In KEK the laboratory site is divided into 7 radiation areas. The radiation monitors are mainly installed at representative places to monitor the level of radiations and radioactivities in air or dust in such areas. The important role of the radiation monitors is to raise an audible alarm with a flashing light when the level of radiations or radioactive contaminations exceeds the assigned one. There are some radiation monitors which are installed near the border between the radiation area and the residential area and also on the site boundary. The crucial role of the radiation monitors on the border is to stop the machine operation as soon as the radiation level exceeds the assigned one. The radiation monitors on the site-boundary always verify the background level of radiations. The sub-station is located in each local control room of its facility. During the machine operation, operators sometimes need informations on the radiation monitors and on the generations of radiation alarms and interlocks in order to feed those back to the operation of their facilities. The sub-station also make the book-keeping of a short period (a week), and display the dose rates at all monitoring points and messages about the radiation interlocks. The center is located in the radiation control office and all the sub-stations are connected to this center. The center receives the status and the radiation levels of all the monitors from 7 stations. The book-keeping of all the data about the radiation monitors are also made at the center.

THE SYSTEM OF NORM

The NORM is at present composed of three different systems, a

central mini-computer (CENTER), 7 sub-stations (STATION) and 192 stand-alone radiation monitors (SARM). Each STATION is star-likely connected to the CENTER with two pairs of local telephone lines. Each SARM is also star-likely connected to one of the STATIONs with a coaxial cable through a module in a CAMAC system which is dedicated to the STATION. At each monitoring point, two SARMs are normally installed with a detector for gamma-rays and one for neutrons. In order to monitor activities, a detector for gamma-rays is used. The detectors presently used in KEK are shown in table 1.

Two types of the radiation monitor are developed. One is a table-top radiation monitor and is composed of a radiation detector with a preamplifier, a main amplifier with a discriminator, a presetable scaler with a comparison logic circuit and a 100 mA cable driver. All the circuits except the preamplifier are contained in a monitor frame with a high voltage and a biasing power supplies as shown in Fig.1. A detector can be set on the frame or separately on a wall. The other is a wall-mount radiation monitor, which is furnished with just doubled circuits of the table-top one in its frame. The detectors are separately mounted on a wall. The dose equivalent rate due to gamma-rays and also due to neutrons are separately calculated from each counting rate. The dose equivalent rate summed up every 10 sec is displayed with LED on the front panel of monitor frame. The signal, which is correspond to each count, is sent to the STATION through the 100 mA cable driver. The termination of the 100 mA current driver is made at the input of a CAMAC module with a resistor of 50 ohm. It keeps the dc level of the input 5 volt and a narrow negative going pulse correspond to each count as a signal. The dc level of 5 volt is used to detect a power failure at the SARM. In order to use the ionization chamber for the SARM, a circuit, which is used to quantize the output dc signal from a charge integrator of the chamber, is incorporated instead of the amplifier system.

The STATION is composed of a micro-computer (NOVA MP/100) with a down-line loader, a CAMAC system and a graphical display unit (Apple II) as shown in Fig.2. In the CAMAC crate, a timer module is used to generate a LAM signal every 10 sec to the computer to force it read the data in all modules. Modules used in the NORM are mainly 4-channel 24-bit scalers and interlock drivers. Each scaler registers the signals from a SARM and also watches the dc level of the input. If the dc level is zero, it outputs a special bit pattern to the CAMAC dataway. If not, it sends out the number of counts for 10 sec to the dataway. An interlock driver receives the same signals as a scaler and looks for the case that all counts in eight short successive periods exceed the counts which correspond to a level of radiation interlock. If the case is found, a signal is sent to the operator's console and the operation of accelerator is automatically stopped. The computer of the STATION reads all the data from modules in the crate every 10 sec, and sorts and puts those data in its memory. Also, it sends the data to the CRT display. It interrupts the center computer and sends the status of the SARM if it finds the SARM which generate alarms or interlock signals. Furthermore, it sends the data to the CENTER, whenever it receives a request from the CENTER. The data of hourly dose from

all the SARMs are sent to the CENTER once a day.

The CENTER is a mini-computer (NOVA 4/X) system equipped with a disk with large capacity which is shown in Fig.3. The computer is operated under the real time disk operating system which has two grounds. An on-line program, which is used for communication between the CENTER and the STATIONs, is always running at the Background. The analysis and the book-keeping of the data from all the SARMs are made at the Foreground. Furthermore, programs running at the STATION are written at the Foreground and are sent through the down-line-loader to each STATION. A main task of the on-line program is to collect all the data which are sent from all the STATIONs once a day. In the disk, recent two years data are kept in the form of hourly dose for all the monitoring points and are served to analyze the data and to make the book-keepings of those. The current data in the forms of every 6 min, hourly, and daily dose for a past week are always available at the STATION with the request of the CENTER.

PRESENT STATUS

The NORM is composed of 7 STATIONs and 192 radiation monitors at present and have grown from the composition of 3 STATIONs and 48 radiation monitors at the beginning as mentioned above. There now exist two STATIONs which have no space for adding modules further, but a mini-CAMAC system, which is connected through a register module instead of a scaler one in the STATION, is now used to install more SARMs. The computer of the STATION have a only 64 kiro-bytes memory and the program size are limited. However, we still afford to add a STATION in the NORM. About 20 monitors are always ready for installation at present. Whenever troubles at the SARM are found, it is completely replaced with a same type of SARM as soon as possible. If there are found any troubles with three boards in the MP/100, a processor, a memory and an interface one, those are easily replaceable with their alternatives. The periodical examination of all the STATIONs and SARMs is made once a year and the whole system of CENTER is examined twice a year.

DISCUSSION

This NORM system had been planned and designed about 10 years before. In this period we saw outstandingly rapid growth of mini-computers and micro-computers. However, the computers used in the NORM are still convinced to be the best choice at those times. The star-like network of the computers is still powerful in our laboratory because the local power failure or cut only gives rise to the loss of one or two STATIONs. In the case of the radiation monitor, which had been designed as the SARM at those days, we had a lots of benefit for maintenance, replacement, movement, installation and enlargement. Furthermore, the unification of communication signal makes the whole system reliable and especially the 100 mA current driver makes the connection of a monitor to the STATION easy and is free from the contamination of noise in signals.

Neutron	BF_3-Counter with 6.5cm polyethylene	15cmϕ x 180cm 2"ϕ x 14" 1"ϕ x 10" 1"ϕ x 4"
Gamma-ray (x-ray)	side window GM counter	5/8" x 8.58"
	air filled ionization chamber	10 ℓ
Gamma-ray (Activities)	NaI(Tl) scintillator	2"ϕ x 2"

Table 1 Radiation monitors

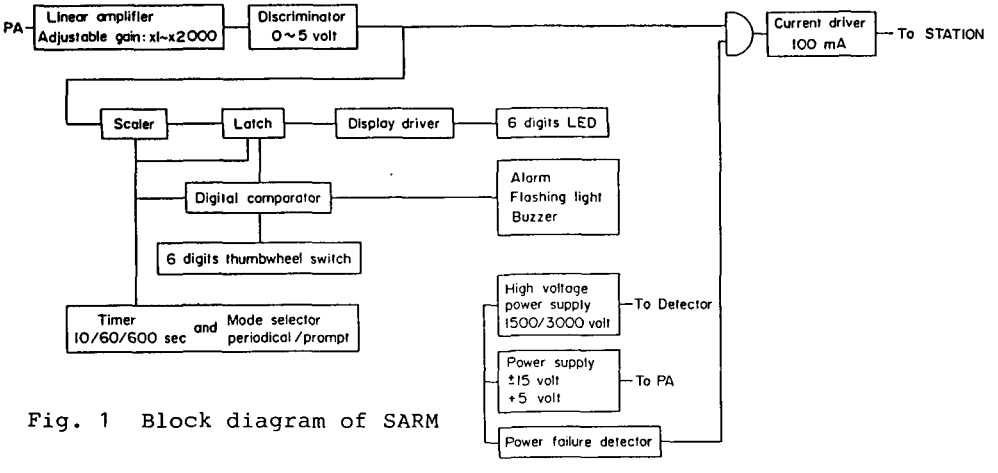

Fig. 1 Block diagram of SARM

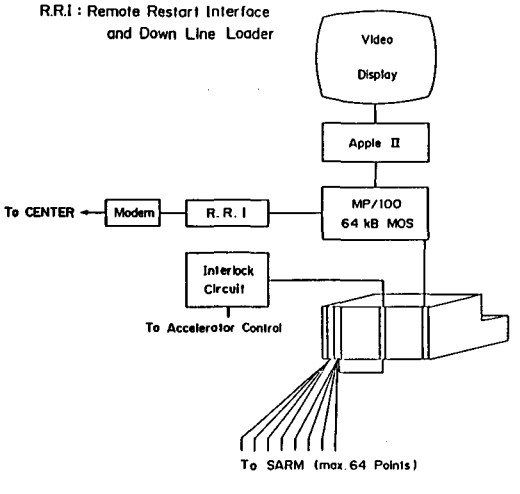

Fig. 2 Block diagram of STATION

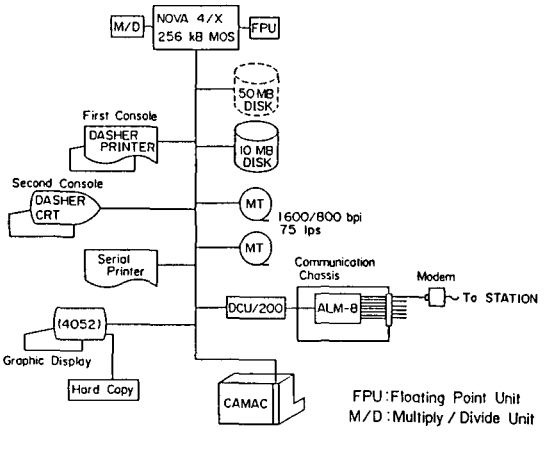

Fig. 3 Block diagram of CENTER

THE USE OF THE HELICOPTER AS A MEANS OF TRACING A RADIOACTIVE PLUME

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ABSTRACT

This paper discusses the use of the Helicopter as opposed to fixed wing aircraft as a vehicle for monitoring a plume of radioactive material released during a Nuclear Accident: advance planning and mapping, communications, instrumentation, interaction with ground monitoring teams, operational support, contamination and dose control are among the topics addressed.

A OPERATIONAL NETWORK FOR RADIOACTIVE CONTAMINATION IN FOODSTUFFS IN THE NETHERLANDS

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ABSTRACT

Almost thirty years a network of measuring devices for determination of radioactive contamination in foodstuff exists in the Netherlands. It had been designed in connection with the probability of "fall out" after an atom bomb explosion. Later on it was understood that such a system could be used in case of an accidentel release of radioactivity after an accident with a nuclear chain reactor. In the early days of May 1986 (after the Chernobyl accident) the network came in operation in order to measure milk, meat, vegetables and other products. The system improved to be operational and became very important, not only to establish the contamination degree of products but also as a tool for controlling the effectiviness of measures that had been taken.

In the planned poster presentation the following aspects will be described and shown:
- the purpose of the network related to the responsibility of the Ministry of Agriculture and Fishery in the Netherlands when radioactive contamination of animals, plants and products occurs.
- The extent of the network with special attention to the choice of the locations of the measuring devices and the products that can be measured.
- A technical description of the network. Attention will be paid to the following subjects:
  * the equipment at the different locations: detectors, multichannel analyser, calculation program. Data transfer to a centralized computer system and final presentation of arranged date;
  * special measurements with sophisticated equipment.
- The operational procedures in case of an accident:
  - starting the network and to keep it in operation;
  - call together a coordination team;
  - connections with the nationally or regionally organized emergency management centre;
  - agricultural measures and their consequenses.

SYSTEM SET UP BY THE C.E.A. GROUP FOR MONITORING ENVIRONMENTAL RADIOACTIVITY IN REAL TIME

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ABSTRACT

The Tchernobyl plant accident has led the C.E.A. group to re-examine its own network for monitoring environmental radioactivity and to define, on a national level, an organization that would, in the case of a significant radioactive pollution, enable information to be collected and employed to establish diagnostics and advise authorities so that they can act and inform.

The following points are treated in this paper :

- the general structure adopted : information From all French nuclear centers is regrouped and centralized at a control and decision making post conterred the responsibility of establishing a synthesis so that appropriate measures can be taken. The communication network is organized with the different nuclear centers at its nodes. The system is therefore completely interactive,

- the structure of the real time monitoring stations locatel around each of these centers,

- the complete characteristics of the materials used and the specific data acquisition and handling systems developed,

- the operating modes employed in order to assure the quality of the materials used and their correct operation,

- complementary equipment selected for various samplings to be batch treated afterwards in the laboratory,

- the equipment of various nuclear analysis laboratories integrated into the network,

- meteorological measurements made in real time at the various different centers,

- the cost of a monitoring system of this kind developed and realized by the C.E.A. group.

INSTRUMENTAL PROCEDURES TO CHARACTERIZE RADIOLOGICAL IMPACT ON A CONTINOUS BASIS.

F. Sagastibelza Chivite
J.E. Otamendi Carrillo
M. Fernandez García
CIEMAT - ESPAÑA

As a result of the Chernobyl accident, the CIEMAT has developed and instrumented a mobile laboratory to characterize the radiactive - cloud while still airbone.

This unit measures the most important radiological parameters, such as gamma exposure rate, activity concentration on air and on the terrain. The measurements are made on real time and while moving over preasigned roads.

The gamma exposure rate is measured continously in the range - from 1 uR/h to 100 R/h, with two G.M. detectors and an ionization chamber. These units are measure at 0.4, 1 and 6 m. over the ground.

The data are processed automatically and the results are transmitted to an Emergency Center via a microcomputer installed in the mobile unit.

The air activity concentration in measured continously through a particulates filter first, and an activated charcoal filter second. Both are integrated in a plastic scintillation counter for beta radiaction and a second one (NaI) for Iodine.

These results are also automatically procesed and transfered - to the Emergency Center as before. The filters, however, are subsequently analized via gamma spectrometry with a Ge detector, also installed in the unit. This is used to determine the overall isotope composition over each road sector travelled.

The same spectrometry detection system can also be adapted to measure deposited activity over the roads.

Experimental data obtained by these units arround a Nuclear - Power Plant will be presented.

(Poster presentation)

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MOBILE UNIT FOR OPERATIVE INTERVENTIONS IN HAZARDOUS SITUATIONS OF CONTAMINATION

D. GALVAN - G.F. MARCOALDI - A. BAZZAN

E.N.E.A.

In case of an accident having, as main effect, the release in the environment of a radioactive or conventional contamination, the consequent intervention develops, normally, in three subsequent stages.

Stage 1. - First intervention, following immediately the accidental event, having as principal aim the rescue of the operators eventually involved.

Stage 2. - This stage is very often the most important (except in case of instantaneous release) because of the major release of the contaminant.
The principal goal to achieve in this situation is to stop the event's evolution.
It becomes therefore important to achieve the following results:

a) Turning off the apparatus to avoid the risk of further evolution.
b) Safety and control of the whole facility and components liable to be involved or damaged.
c) Shielding of eventual irradiation sources.
d) Complete survey of contamination and, eventually, external irradiation.
e) Complete survey by means of TV images and collection of data about incidental site and the surrounding area.
f) First operations of decontamination or bordering, creating support areas for following interventions.

Stage 3. - During this stage, having complete information about the accident and its consequences, the decontamination operations can start to restore the original conditions of the site.

Generally, in typical accidental situations, the existing structures are able to carry out stages 1 and 3. Only in very special cases the facility has locally complex and protected apparatus suitable to manage stage 2.

Operative Health Physics Service, by E.N.E.A. Casaccia Centre (Rome-Italy), with the purpose of allowing efficient and effective intervention during a "stage 2" situation, realized a Mobile Unit, completely autonomous, able to intervene in safety conditions.

The Mobile Unit consists in a large van and a trailer where all technical equipments and instruments are located. Its design assures the safety and protection of the intervening operators and their foundamental functions: breathing, speaking, working. In fig. 1 it is shown a block diagram of the installed equipments.

The Mobile Unit, with a minimum team of three operators, is able to operate indipendently even inside a toxic or radiotoxic cloud.

The Unit provides continuously the operators with fresh air, tele-communication and TV connection.

At the end of the intervention, operators and Unit can be easily decontaminated by its own decontamination apparatus and be ready for a new timely intervention.

The Unit assures - in EMERGENCY SITUATIONS - the following results and advantages:

- Direct interventions on facilities involved
- Acquisition of TV images and data about the involved site
- Timely transmission of the obtained information to the Public Authorities, for the following civil defence actions
- First decontamination actions preparing following interventions.

In CONVENTIONAL SITUATIONS:

- Decommissioning operations
- Environmental measurements programs

CONCLUSIONS

The Mobile Unit, equipped with an absolute filter system and having long autonomy, is able to operate, inside a toxic or radiotoxic cloud, in many incidental sites.

e.g.: Nuclear or chemical plants, ship holds, tunnels, places saturated by toxic substances.

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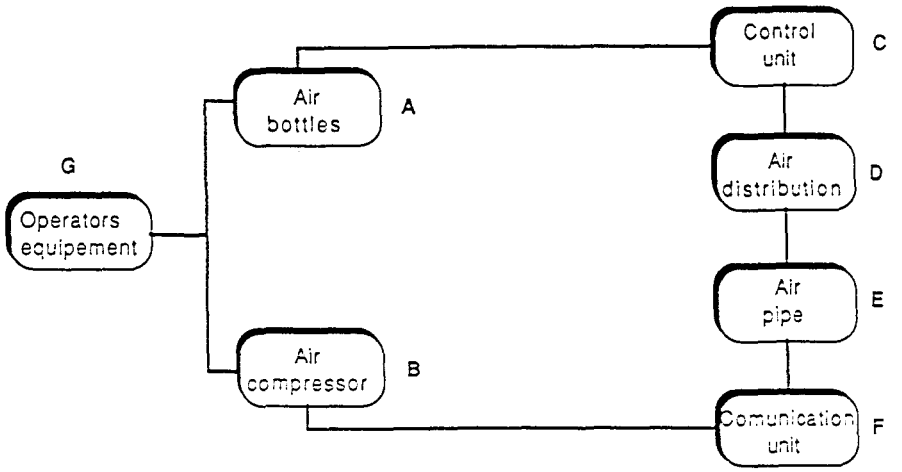
Fig. 1 - AIR DISTRIBUTION SYSTEM



To allow the UNIT FULL operation the following devices are working

H
Generators

I
Supply batteries

L
Batteries for intern.-extern. devices

M
Water group for decontam. and washing

N
External light

O
Absolute filter system

P
Supply aqualungs

Q
Radiotelephon connected with national and intern network

R
CCTV video recorder

S
Instrumentation for internal and external controls

T
Special equipments for different type of protection

RADIATION PROTECTION IN NUCLEAR INDUSTRY PRODUCTION

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ABSTRACT

China's nuclear industry got started in the mid-1950s. A decade later she had succeeded in developing her first atomic bomb, and then followed the peaceful purposes as industry, agriculture and medicine. And now, there are a relatively comprehensive nuclear industry and a large number of nuclear scientists and technicians. Nowadays, the country's first two nuclear power stations are about to be built.

Since 1960, China's nuclear industry has been taken the principles "safety first, with prevention and protection precedent", And a series effective radiation protection measures has been adopted, such as: 1) We establisbed radiation dose standards of radiation protection regulation, carring out three principles of radiation protection. 2) Nuclear reactors and reprocessing plants must have remote control devices, appropriate shielding materials, containment of airborne radioactive materials, interloking safety mechanisms, continous radiation monitoring systems and effective warning systems. 3) We have put great amount of work force and funds in the management of high-, medium- and low-level radwastes as well as in the research on final disposal methods. Bituminization and cementsolidification techniques have been used in the treatment of medium- and low-level radioactive wast liquids. 4) Nmphasises on radiation protection research; strengthens individual protection, health and medicine, prevents the hazard of radioactive materials.

From then on, there have been no deaths from over-exposure, no acute cases of radiation sickness, and no instances of large-scale environmental pollution. Between 1959 and 1984, workers working near reactors, reprocessing plants and isotope workshops, the average per capita external radiation dose equivalent has been 9.96 msv. While workers working in uranium exploration, mining, metallurgy and prodution of enriched uranium, the dose equivalent has been 5.0 msv, less one-tenth the state-stipulated standard.

RADIOLOGICAL IMPACT OF NUCLEAR POWER IN SWEDEN - PRINCIPLES AND TRENDS

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Nuclear power is today a major source of electricity in Sweden. In 1986 the consumption of electric energy in the country - with its 8.3 million inhabitants - was about 130 TWh which is among the highest per capita in the world. Since 1971 twelve light water reactors (LWRs) have been put into operation, the last two in the autumn of 1985. Altogether the reactors have been in operation slightly more than 100 reactor years. Today nuclear power accounts for about 50 per cent of the total production of electricity in Sweden while the remaining half is supplied almost entirely by hydro-electric power.

The occupational radiation exposure has during these years been comparatively low put in an international perspective with only a few exposures over the dose limits. The radiation exposure outside the four nuclear sites caused by the released radioactive substances has generally been only a few per cent of the reference value given by the national regulations. This applies to a situation in which the long term exposure effects from carbon-14 are not included.

1 OCCUPATIONAL EXPOSURE

The occupational exposure is influenced by a number of technical as well as administrative factors. The radiation sources inside a plant are design-related and depend on such factors as material selection and water chemistry. Other important factors are work-planning, education, training, dose surveillance and feed-back of experience. It is also essential that the radiation protection staff has such a position within the plant organization that it can have an overview and influence of all these factors.

The same type of thermoluminescent dosimeters are used by all Swedish plants and the recorded doses are transferred to a joint register for dose data. The intakes of radioactive substances are checked at the plants by whole-body measurements. Internal exposure has, however, been insignificant compared with external exposure.

1.1 COLLECTIVE DOSES

About 70 to 80 per cent of all the collective doses are received during the routine outage periods. The distribution of the annual collective doses has been such that the contractors' personnel received about 75 per cent of the total annual collective doses. To establish a tentative guideline for a restriction of the occupational exposure at the Swedish LWRs, the Swedish National Institute of Radiation Protection (NIRP) has suggested 2 mmanSv per installed MW electrical capacity and year as a level

of ambition to which the collective dose equivalent on average should be limited. Table 1 gives the values for the past ten years.

Table 1 NORMALIZED ANNUAL COLLECTIVE DOSES AT SWEDISH LWRs - 1977 - 1986

Year	Number of reactors	Average manSv per unit	mmanSv[a] per MW(e)·a	mmanSv[b] per MW(e) per year
1977	6	1.70	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
1983	10	1.47	3.2	2.0
1984	10	1.16	2.0	1.6
1985	12	0.92	1.7	1.2
1986	12	1.43	2.1	1.8

a) Energy generated
b) Installed capacity and year

1.2 INDIVIDUAL DOSES

The average annual dose both for plant personnel and contractors' personnel has been between 2.5 and 3.5 mSv. The distribution has been such that about 85 per cent of the workers received annual doses less than one tenth of the 50 mSv dose limit. On average only three per cent of the workers exceed 15 mSv. Only a very few overexposures have occurred and none of them has been serious.

1.3 PREDICTION OF LIFETIME DOSES

Some attempts have been made to predict the lifetime dose-equivalents for the workers who receive the highest occupational exposure. About two per cent of the exposed workers, which means 150-200 individuals, were considered. The average predicted dose over a 40-year employment period amounted to 0.5 Sv with maximum values in the order of 1 Sv. It is impossible to point out a special occupational group as all groups have a few individuals with comparatively high lifetime doses. However, the group "material testers" and some specialists among the "mechanical workers" seem to be the most interesting groups in this respect.

2 RELEASES TO THE ENVIRONMENT

The regulations regarding limitation of releases of radioactive substances from nuclear power plants (NPP) were in their present form issued by the NIRP about ten years ago. The main general provisions state that every NPP shall have such a design for normal operation conditions that the sum of the effective dose equivalents due to expected resulting releases should be less than 0.1 mSv per year. This dose applies to the critical group. An annual evenly distributed release corresponding to this dose value is named a norm release. This value shall be

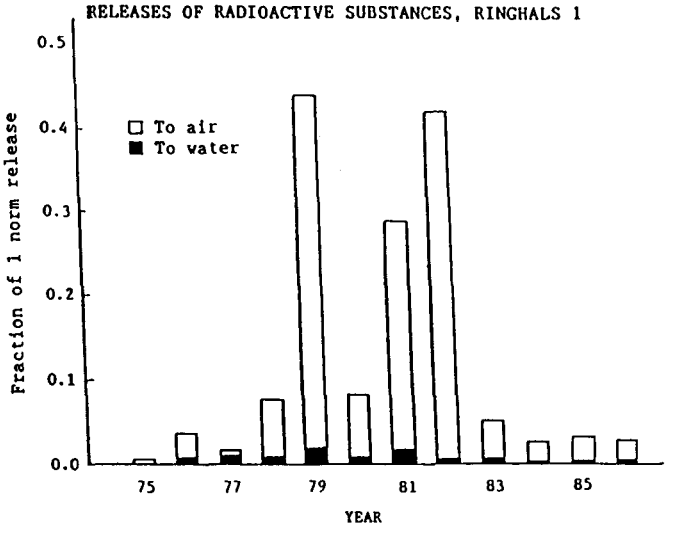
seen as a reference value and it has not the formal character of a limiting value for the permitted dose in a given operating situation.

The limit for the global collective dose equivalent is set to 5 manSv per gigawatt of installed electric power and year. For radioactive nuclides with long half-lives such as carbon-14, the time integral of the collective dose equivalent rate is over 500 years. The purpose of the limitation of the collective dose is to limit the future annual average dose equivalents to 100 μSv at a time when the number of reactors may be much greater than today. This requirement has so far been applied less stringently than was intended but that can be acceptable during the initial period of time of the nuclear power era.

2.1 ATMOSPHERIC RELEASES

The releases of radioactive nuclides have varied in time and between the various units. The dominating amount of releases comes from atmospheric emission of radioactive noble gases. This kind of release depends greatly on the condition of the fuel cladding. For example the releases of noble gases from Ringhals unit 1 have been the highest among the Swedish reactors, see figure 1. In this case the fuel cladding damage is expected to

Figure 1



have been caused during the first years of operation when different power change tests were performed. The original old fuel has been gradually replaced which has resulted in considerably lower releases. In recent years the atmospheric emission from Ringhals 1 has been below 5 per cent of a norm release which corresponds to an annual effective dose equivalent of 5 μSv or less. Furthermore the newer reactors in Forsmark and Oskarshamn have been equipped with offgas treatment systems including charcoal columns and a recombiner system. These systems have limited the emission of radioactive gases from the plant further.

With the exception of a few years, the annual atmospheric emission has been of the order of 1 to 5 per cent of a norm release for Ringhals and Oskarshamn. For Forsmark and Barsebeack the releases have been at least an order of magnitude lower. The contribution from C-14 is not included in this assessment.

2.2 AQUATIC RELEASES

As in the case of the releases to the atmosphere, the releases to water have not varied appreciably between the four nuclear power sites. In this respect Figure 1 can therefore be seen as a representative example. This means that the relative contribution from aquatic releases to the exposure of the critical group is much higher for Forsmark and Barsebaeck than for the other two Swedish sites. The releases to water, which mainly consist of radioactive corrosion products, have been of the order of 1 to 3 per cent of a norm release .

2.3 COLLECTIVE DOSE ASSESSMENT

The major contribution to the collective dose comes from carbon-14 wich is mainly released to the atmosphere. However, no measurements of releases are regularly performed at the Swedish nuclear power plants. To assess the radiological impact of released carbon-14, data are taken from national and international analyses.

The estimated doses are given in Table 2. For Ringhals site this means 3 manSv per GW(e) installed capacity per year and for the other three sites 6 manSv. The latter value is slightly above the ambition level given in the regulations.

Table 2. Annual doses from releases of C-14 from Swedish LWRs

Site	Individual dose[b] (μSv)	Collective dose (manSv)
Barsebeack	0.9	7
Forsmark	0.6	18
Oskarshamn	0.5	13
Ringhals	11	10

b) Critical group

2.4 SUMMARY

The highest releases have been at the Ringhals unit 1 and Oskarshamn unit 1, and have in some exeptional years been of the order of 40 to 50 per cent of a norm release. This corresponds to assessed doses of about 50 μSv. For most of the years the assessments of the exposure for the critical group give doses of the order of 5 üSv or less when the contribution from carbon-14 not is included.

DOSE REDUCTION AND CONTROL AT THE WINFRITH REACTOR

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1 INTRODUCTION

The Winfrith Reactor is a 100 MW(e) heavy water moderated, light water cooled, pressure tube reactor. It was designed and built in the 1960s as a prototype, using the best technology available at the time. Coolant chemistry problems initially caused fuel failures and due to material specification, significant quantities of activated corrosion products and minor quantities of fission products are transported into working areas. For several years the Reactor has been the focus of a major dose reduction programme with the emphasis on reducing individual doses. The options available in a dose reduction programme are straightforward. Remove or reduce the source and then apply the usual "distance, shielding and time". Source reduction depends upon the replacement of dose contributing alloys (eg Cobalt-rich stellite) where practicable, the refinement of chemical control of the coolant to minimise pick-up, transport and plate-out of the active species together with chemical cleaning of the circuit before major maintenance work starts in the primary containment. Gains can also be made by arranging rapid removal of active waste.

Distance usually means relocating active plant items requiring maintenance from high dose areas to low. Identification of such items is greatly facilitated by the collection of dose records specific to work areas and jobs. The costs of plant relocation are often high and in Cost Benefit terms unattractive (see 5). For an existing plant, shielding may be portable - eg lead wool blankets - or permanent, and may be cheaper than relocation. Permanent shielding need not necessarily be of sophisticated design; for example a high density concrete brick wall, erected by relatively unskilled staff, produced a dose rate reduction of about x10 in an area of high occupancy. Of all the available methods, working time reduction and dose sharing potentially offer the most cost effective option but their successful application is crucially dependent on sophisticated dose recording and analysis techniques. It is with these techniques that this paper is chiefly concerned.

2 ANALYSIS OF PERSONAL DOSE

2.1 Data Collection. Effective control of individual doses in situations where significant doses can be incurred very rapidly, requires a system which works very closely to real time. The system developed involves the issue of personal indicating dosemeters (PIDs) on entry to potentially high dose, barriered, work areas in addition to routine issue TLD badges. These PIDs are read out by Health Physics Barrier Controllers and the data transferred to the local Dose Control Computer. The flow diagram

of the system is shown in Fig 1. Although careful sensitivity checks have been made between the PIDs and the (legal) TLDs, for staff approaching local dose limits, the TLD badge read out is available within 30 minutes.

2.2 Dose to Different Work Groups. To target dose reduction measures most productively, the reactor work force was split up into about 40 different groups and the dose received by each group measured. As with most water reactors the highest dose groups were those associated with mechanical maintenance work. An example of a dose reduction measure for these groups is to use labour from other, lower dose, groups to erect temporary shielding rather than mechanics themselves. Individual doses in groups were also looked at and where wide variations were apparent, supervisors were requested to investigate the situation to share the dose more evenly.

2.3 Dose for Different Tasks. Of the several hundred planned tasks carried out during a maintenance shutdown, analysis showed that about 20 produced 60 to 70% of the total dose. Most of these high dose tasks are foreseeable and special provisions to reduce doses made, eg special tools, training, mock-ups and local shielding; however some of these jobs encounter unforeseen problems and it is essential to review task doses daily to allow remedial actions to be taken. Some initial effort was required to motivate staff to provide correct task numbers to the Barrier Controllers but the accuracy of dose allocation has steadily improved and with the issue of task number tags to personnel, has probably ensured a 90 to 95% accuracy - more than adequate for planning purposes. Further analysis highlighted the high proportion of dose incurred on some high dose tasks by those carrying out preparatory work such as scaffolding installation and removal and replacement of thermal lagging. More permanent scaffolding and demountable lagging panels are now provided.

2.4 Presentation of Dose Information. It is essential that the supervisors organising and allocating work are provided with regular, frequent and clear information on individual doses. During the annual maintenance shutdown the work force is enhanced by contractors and staff from other buildings and AEA sites and many people have different dose limits ascribed to them by their home base. A simple alphabetical list of "best estimate" doses is therefore inadequate. During the night, Health Physics shift staff produce a dose list of each supervisor's team and as well as listing "best estimate" monthly, quarterly and annual doses, a final column lists "dose to go to lowest limit". Supervisors usually use data in this last column as a basis for work allocation. Individual and task dose data also go to senior supervisors and Health Physicists.

3 SIMVIDOSE

The system is fully described elsewhere (Ref 1) and its

operational use at the reactor will be illustrated here. One job involving a potentially high dose was the replacement of a reactor pressure tube. The TV camera and lights were therefore erected in an area where dose rates of up to tens of mSv hr^{-1} were possible and the dose incurred had to be offset against potential dose savings. In the event the task was carried out for about 50/60% of the original dose estimate. Some of this saving was due to training and equipment improvements on the mock-up but SIMVIDOSE contributed the following:

(a) Supervisors were able to monitor work from low dose rate areas and act appropriately.
(b) Support staff viewed the work on TV and needed a minimum of takeover briefing.
(c) The person wearing the dosemeter effectively carried out an area radiation survey and this allowed "hot spots" to be identified and shielded. The video tape of the operation is also a useful training aid.

4 TRAINING

4.1 Training Techniques. When providing training for numerous staff the use of a video or slide/tape presentation is often preferred. However some form of personal contact must be established perhaps via a question and answer session at the end, and by highlighting special factors applicable to the particular audience. (The classic methods of dose reduction by "Time, Distance and Shielding" should be physically demonstrated.)

4.2 Refresher Training. It has been found that initial training has to be supplemented by other techniques to maintain individual interest in dose reduction measures. These other techniques involve the repeating of initial training, preferably with some variations to maintain interest, personal contact when specific tasks involving high doses are proposed or carried out or high individual doses incurred, and financial incentives, such as staff suggestion schemes, for dose saving ideas. The provision of eye catching notices highlighting areas of high dose rate and low dose rate have proved useful but are soon taken for granted and need to be revised regularly to maintain an impact.

5 COST BENEFIT ANALYSIS - CBA

CBA, as part of the ALARP (As Low As Reasonably Practicable) process, has been helpful in the selection of viable schemes but much depends upon the skill, familiarity with the plant and its history and appreciation of the future programme possessed by the practitioner. Usually however the "social" factors of ALARP outweigh the "economic", and expenditure on dose saving is frequently greater than the approximate £25K per man Sv indicated by the NRPB proposals (Ref 2). The expenditure justified by CBA is often exceeded by custom and practice in the industry.

6 ASSESSMENT OF DOSE REDUCTION PROGRAMME EFFECTIVENESS

Measurement of dose savings from the various actions described is hampered by the interaction of plant modifications and changes in work requirements or operating regimes. The general picture of dose rates in working areas can be measured by fixed point instruments (40 are used in the reactor containment) but even here the results are confused by the effects of local plant changes. Two assessment methods are therefore used:

(a) Standard tasks carried out during each annual shutdown, chosen for their independence of other work and their total dose penalty measured.

(b) The total daily personnel dose is measured for operational and shutdown periods. Table 1 shows the rapid initial improvements but the increasing difficulty in maintaining the annual reductions, which progressively need greater effort to achieve.

7 CONCLUSIONS

The dose reduction programme has been very successful with the total annual collective dose halved over 5 years but further improvements are increasingly hard to achieve. The use of comprehensive computerised data analysis has been a major factor both in targeting action areas and the operational control of dose uptake. Finally, a co-operative climate of commitment to the programme both amongst workforce and management is essential both during training and operation.

Fig 1 SYSTEM FLOW DIAGRAM

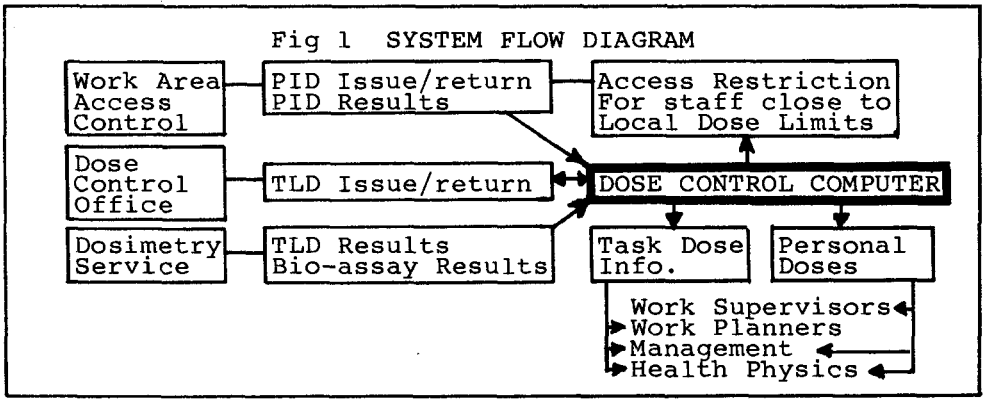

Table 1 NORMALISED SHUTDOWN AND OPERATIONAL EXTERNAL DOSE RATE

	1981	1982	1983	1984	1985	1986
Average Shutdown Dose per day	1.00	0.81	0.69	0.66	0.63	0.63
Average Operational Dose per day	1.00	0.87	0.70	0.67	0.59	0.61

Ref 1 GC Meggitt Reduction in Occupational Doses Using the UKAEA SIMVIDOSE System (Paper 292 IRPA-7).

Ref 2 National Radiological Protection Board - ASP.9.

RADIATION DOSES AT ENEA NUCLEAR RESEARCH SITES

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The ENEA (Italian Nuclear Energy Agency), through its Health Physics Department sited in Bologna, provides a radiation monitoring service by delivering and reading about 60000 film-badges and TL dosemeters in order to control the radiation doses of about 15000 workers exposed to radiation risks in many kinds of plants in the fields of industry, medicine, nuclear, research and so on.

Among these workers some 2000 work at ENEA nuclear sites, where research reactors, radiological machines, radiochemical laboratories, plutonium facilities and fuel elements fabrication and reprocessing plants are in operation.

Such a large quantity of workers and the larger quantity of dosemeters (one or more per each worker and per each month or 45-days) have imposed the use of an automated production system controlled by a computer.

The whole dosemeters management automatic system, an overall description of the dosemeter used and the main results of an in-progress study on occupational doses to ENEA radiation workers, in comparison with the other Italian radiation workers, will be now briefly outlined.

The dosemeters management system automatically performs the following operations (fig.1) by means of a local control computer: dosemeters preparation and user-match; dosemeters reading and identification; individual dose data management (storage and retrieval); statistical analysis (annual mean individual dose, standard deviation, yearly collective dose, cumulative distributions, etc.); notification and warning report production. To perform these tasks the local control computer is connected with a host computer IBM-3090 which is linked in network with ENEA's main frame systems.

The local computer also maintains all the information files and performs all technical and administrative operations for their management. The master files are:

1) Customer file (where the customer may be an establishment or a person) which contains customer addresses, number of workers who need dosemeters and type of dosemeter, and all administrative information;

2) Workers file which contains information about workers exposed to radiation such as date of birth, sex, number and type of worn dosemeters and, if possible, details concerning their employment and accumulated doses for the previous and current year;

3) Dosemeter file which contains more detailed information about dosemeters issued by the service.

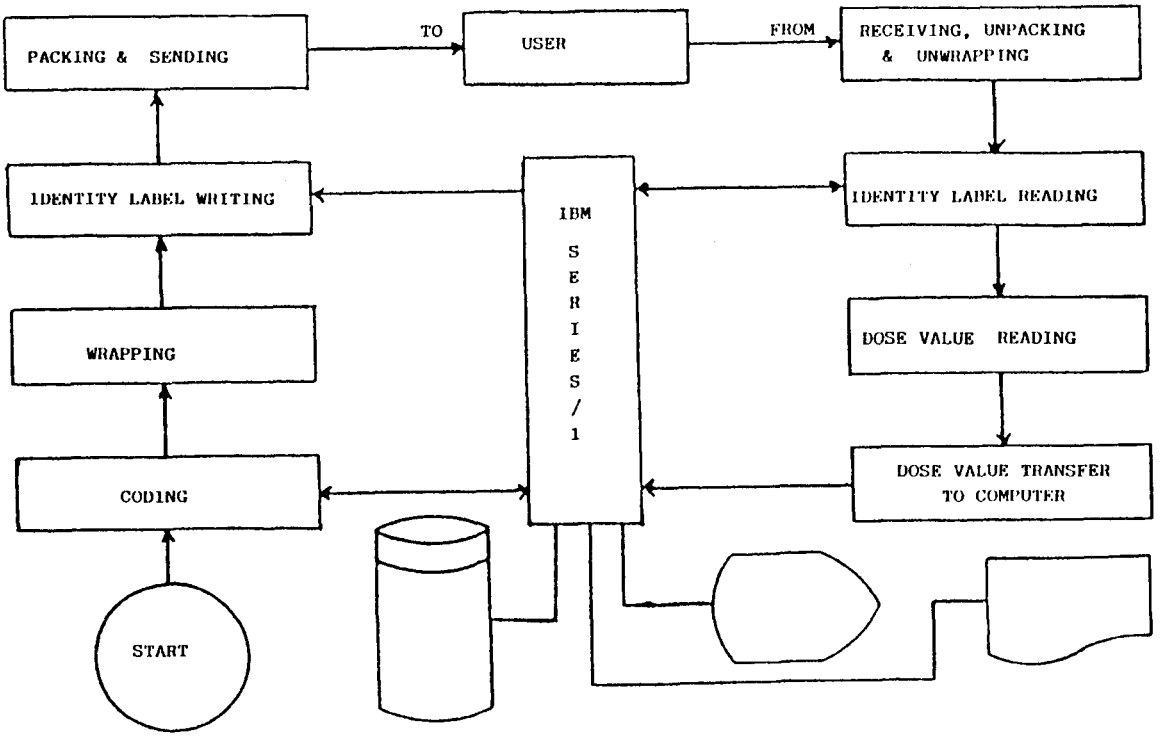

Fig.1: Dosemeters management flow-chart

As for the type of dosemeters issued, the H.P. Department in Bologna, at the present time, has just turned to use BeO TLDs instead of film-badges in view of the known advantages of accuracy, precision, cheapness and handiness of the first one, in comparison with the latter. In addition, compared with other commercial dosimetric systems, the employed technique, based on one card with two BeO detectors and two cards to filter one of them, presents the following main advantage: the BeO dosimetric card, on its return to the service, can be reassigned to any user; in fact, while an automatic packing machine puts the three cards together, the bar code is coupled to a user name and this information is printed on the front thin card and kept by the computer. On return, dosemeters are unwrapped and the BeO card is read by an automatic TL-reader (Harshaw mod. 2271) which sends the code and the two readings (filtered and unfiltered) to the computer for dose calculation. After the reading, in the case of low doses, the card can be again reassigned to any user without additional thermal annealing cycles. In this way dosemeter management is easier and faster and we can also minimize possible systematic errors taking into account the calibration of the single BeO card, previous evaluated and recorded.

As for the dose data analysis -owing to space constraints just some main and typical results are shown here- such a kind of study can be a starting point from which the Health Physicist, as a person strongly involved in a radiation protection program, can receive suggestions, perform evaluations, identify future needs and possibly make improvements as for the workers radiation protection level in order to better meet the legal requirements too. From a broader view point the use of doses from personal monitoring services can offer a useful data base for epidemiological studies. Tables I and II show two examples of typical

outputs of the computer system after the statistical treatment of the dose data.

Individual interval (mSv)	N. of workers (%)	Total collective dose inside the interval (man-mSv)(cumulative)	% of collective dose inside the interval (cumulative)	Mean individual dose (mSv)	Standard deviation (mSv)
0 ÷ 0.5	782 (80.7)	13.65 (13.65)	3.85 (3.85)	0.017	0.077
0.5 ÷ 1	38 (3.92)	28.93 (42.58)	8.17 (12.02)	0.761	0.142
1 ÷ 2.5	112 (11.56)	186.91 (229.49)	52.79 (64.81)	1.669	0.423
2.5 ÷ 5	36 (3.72)	117.23 (346.72)	33.10 (97.91)	3.256	0.562
>5	1 (0.1)	7.41 (354.13)	2.09 (100.0)	7.41	-
Overall	969 (100)	354.13	100.00	0.365	-

Table 1: Doses to all ENEA workers in 1985

Year / ENEA site	1980	1981	1982	1983	1984	1985
BOLOGNA	5.04 (0.05)	1.86 (0.02)	2.20 (0.01)	4.82 (0.03)	3.61 (0.02)	0.0 (0.0)
SALUGGIA	8.28 (0.03)	13.19 (0.08)	52.77 (0.21)	11.60 (0.05)	1.11 (0.02)	1.08 (0.0)
TRISAIA	87.65 (0.43)	36.94 (0.19)	102.39 (0.45)	72.03 (0.33)	30.01 (0.12)	347.49 (1.35)
FRASCATI	89.31 (0.31)	36.33 (0.11)	42.51 (0.13)	11.88 (0.04)	62.20 (0.18)	71.30 (0.17)
CASACCIA	214.71 (0.22)	103.35 (0.11)	40.09 (0.04)	29.95 (0.03)	31.09 (0.03)	32.17 (0.04)
All italian radiation workers	(1.00)	(0.28)	(0.30)	(0.38)	(0.20)	(0.45)

Table 2: Breakdown of collective (individual) doses (man mSv(mSv)) during recent years at ENEA centres

It can be seen that both the collective doses and the mean individual ones are not only lower than the statutory and the recommended limits but even satisfactorily low especially when compared with the mean individual doses to all the italian radiation workers. Apart from the good radiation protection level at ENEA nuclear sites, as a matter of fact the radiation risk at ENEA plants is fairly low; that is mainly due to the very kinds of the plants themselves.

A last brief consideration concerning raw data treatment and interpretation: it is well known that, in dealing with such data analysis problems, some questions arise about dosimetric readings when they are not available (because the dosemeters are lost or something like that) or they are lower than the minimum detectable dose, which is 0.1 mSv for the dosemeter used.

In the first case the "missing" periodic dose value has been calculated on the basis of the remaining three quaterly readings of the concerned worker during the concerned year. This procedure is based on the assumption, which is fairly realistic at least at ENEA, that the kind of job and operations performed by that worker during that year is fairly homogeneous(*). As far as doses lower than a minimum level are concerned, these are considered in the analysis, the recorded values being zero.

As concluding remarks the whole dosemeters management automatic system here depicted is an easy to use tool for maintaining, retrieving and updating personnel radiation exposure records in an accurate, reliable and adaptable way. The dosimetric history of any radiation worker is continuously up to date just to comply with the legal requirements too. The statistical analysis software module can provide a broad and, at the same time, an in-depth description of the situation of occupational radiation exposures at ENEA plants, showing how limited the radiation risk is at these plants.

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(*) Anyhow it should be borne in mind that these cases of lost dose values are about 1% of the total for any year, so the distorsion due to them is at a minimum.

DEBITS D'EXPOSITION DANS LES CENTRALES REP EDF : ANALYSE ET TENDANCES

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1. INTRODUCTION

Le parc nucléaire français représente à ce jour 40 unités de type REP en fonctionnement, pour 200 années réacteur.

Son importance a nécessité la mise en place d'un programme d'investigation cohérent afin de surveiller et analyser les niveaux d'exposition au voisinage des circuits.

Cette approche a permis de dégager les facteurs importants expliquant le bon comportement des tranches françaises /figure 1/.

2. LES MOYENS D'INVESTIGATION

2.1. Les mesures IN-SITU

Elles sont effectuées au cours des arrêts :

- EDF/SPT/DSRE mesure les débits d'équivalents de dose (DDD) au voisinage des circuits principaux de chaque tranche du parc, en des points et pour des situations donnés.

- le CEA/CEN CADARACHE détermine, à la demande d'EDF, l'activité déposée des divers radionuclides et plus particulièrement celle des isotopes du Cobalt, Co58 et Co60, qui contribuent pour l'essentiel aux DDD. 50 campagnes de mesures ont été effectuées à ce jour.

2.2. Les études fondamentales

Elles concernent l'étude du comportement des matériaux, du transfert des produits de corrosion et le développement des moyens d'analyse des oxydes. Elles s'effectuent sur boucles d'essai ou in-situ.

Les enseignements recueillis sont intégrés dans les modèles utilisés dans le code PACTOLE du CEA, qui constitue le lien entre les aspects théoriques et les observations sur réacteur.

FIGURES

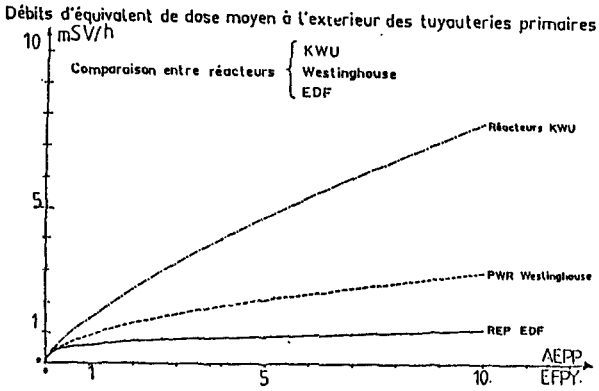

Figure 1

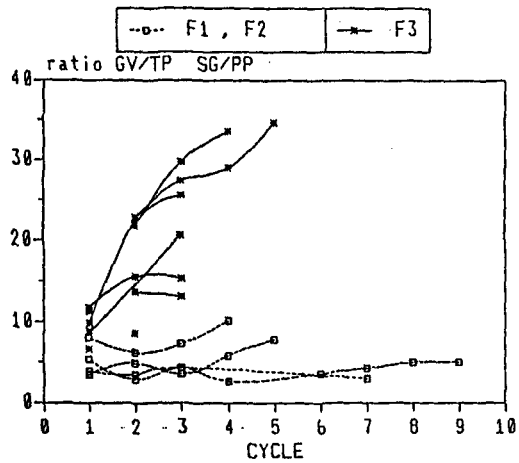

Figure 2 : REPARTITION DE L'ACTIVITE DEPOSEE ENTRE LES GV ET LES TUYAUTERIES PRIMAIRES

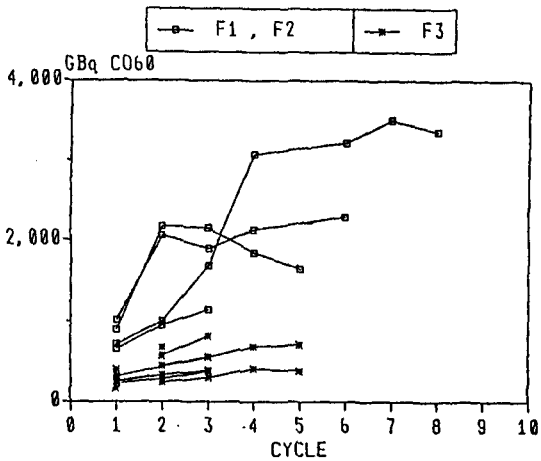

Figure 3 : ACTIVITES Co60 DEPOSEES SUR LES TUYAUTERIES PRIMAIRES

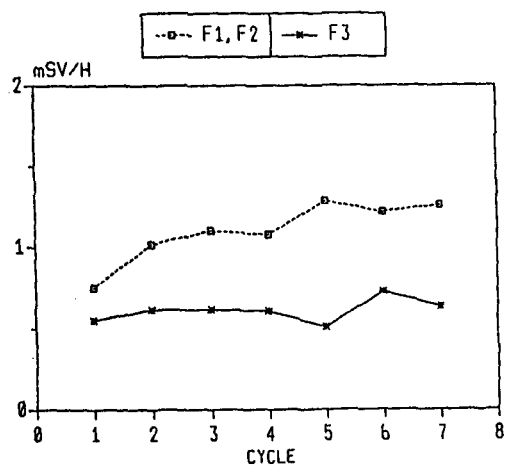

Figure 4 : DDD AU VOISINAGE DES TUYAUTERIES PRIMAIRES

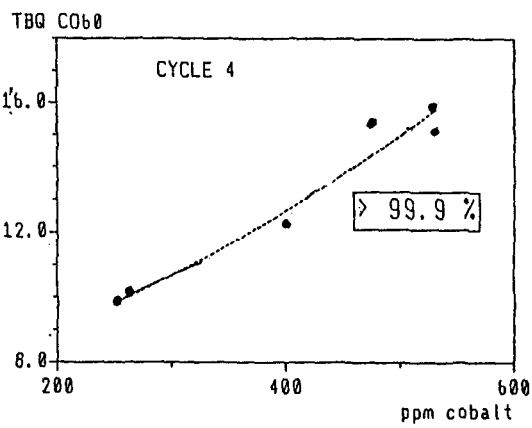

Figure 5 : ACTIVITE Co60 PRODUITE ET TENEUR EN COBALT DES GV

Leur respect par l'exploitant explique que le retour d'expérience d'EDF ne permette pas de mettre en évidence de façon nette l'influence du conditionnement chimique.

Les réflexions sur les valeurs du PH optimum (6.9 ? 7.1 ?) se poursuivent et des éléments de décision devraient être fournis par des essais sur réacteur.

4. CONCLUSIONS

Les bons résultats obtenus par EDF sur les débits d'exposition au voisinage des circuits résultent de la conception et du fonctionnement de ses réacteurs. Ils constituent un environnement favorable pour les opérations de maintenance.

Ce constat devrait être confirmé dans les prochaines années par :

- l'introduction, progressive depuis 1984 sur les tranches en fonctionnement, des grilles de maintien du combustible dites de "conception avancée" (Zircaloy) ;

- mais également pour les générateurs de vapeur de remplacement ou équipant les tranches non encore démarrées :
  - . faisceau tubulaire en alliage présentant une bonne résistance à la corrosion (Inconels traités thermiquement, Inconel 690),
  - . boites à eau électropolies, afin de limiter leur contamination par les produits d'activation de corrosion.

Néanmoins, EDF restera vigilant sur divers points :

- comportement des faisceaux tubulaires des GV de type F3 et DDD au voisinage de ces faisceaux ;

- problèmes liés au vieillissement du parc et à la tenue des matériels ;

- impact du fonctionnement (augmentation de puissance, de la manoeuvrabilité, etc...) ;

- poursuite des réflexions sur les matériaux de remplacement des stellites, le conditionnement chimique ou la passivation des circuits.

Ces efforts constants témoignent de l'intérêt accordé par EDF aux niveaux d'exposition et à la dosimétrie associée à la maintenance des réacteurs.

L'expérience acquise en matière de contamination des circuits, des débits d'équivalents de dose et des moyens de calcul associés devraient se révéler utiles pour les opérations de maintenance normales ou exceptionnelles.

3. FACTEURS IMPORTANTS REGISSANT LA CONTAMINATION DES CIRCUITS

3.1. Influence du mode de fabrication du faisceau tubulaire des générateurs de vapeur (GV)

Une corrélation a été établie entre la contamination des faisceaux tubulaires et leur origine.

Parmi les trois fournisseurs (F1, F2, F3), on note un comportement très différent pour les GV de type F3, qui équipent la majorité du parc : on constate sur un même réacteur que ces faisceaux retiennent plus l'activité des divers radionuclides, 4 à 5 fois plus que les faisceaux F1 par exemple.

Ainsi pour les tranches possédant au moins un de ces GV, on observe une activité déposée préférentiellement sur les faisceaux au détriment des tuyauteries primaires /figure 2/.

On constate également une activité totale déposée sur le circuit primaire plus importante pour le Co58, résultant d'un relâchement de Nickel plus élevé.

La combinaison de ces facteurs se traduit par :

- des DDD plus élevés au voisinage des faisceaux de type F3,

- une moindre contamination des tuyauteries, très visible pour le Co60 /figure 3/, et donc des DDD plus faibles au voisinage des tuyauteries /figure 4/ et dans les boites à eau, où ont lieu la majorité des travaux de maintenance.

3.2. Influence de la teneur en Cobalt du faisceau tubulaire des GV

L'effet bénéfique d'une faible teneur en Cobalt des faisceaux sur l'activité totale déposée en Co60 a été mise en évidence par les bilans d'activités provenant des mesures IN-SITU /figure 5/.

Ceci justifie l'attention portée par EDF à la réduction des teneurs en Cobalt de ces matériels. Les autres sources potentielles de Cobalt n'ont pas été pour autant négligées.

3.3. Influence du conditionnement chimique du circuit primaire

EDF et CEA ont toujours considéré que le conditionnement chimique était un moyen efficace permettant de limiter les niveaux d'exposition.

Aussi, des spécifications ont été imposées, depuis 1981, à l'ensemble des tranches en fonctionnement : elles consistent à maintenir, dans la limite des contraintes d'exploitation, un PH à 300°C constant et égal à 6.9.

REASONS WHY TVO HAS ONE OF THE LOWEST COLLECTIVE DOSES AMONG THE WORLD'S NUCLEAR POWER PLANTS

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ABSTRACT

Industrial Power Company Ltd. (TVO) owns and operates two boiling water reactor (BWR) units of Asea-Atom design in Olkiluoto, Finland. The installed net electric power of each unit is 710 MW. The full power operation of TVO I and TVO II began in 1979 and 1980, respectively. This paper discusses the main reasons why the annual personal doses have stayed on a very low level at the TVO power plants. The variation in annual collective doses falls between 0.3 and 0.8 Sv, and the highest annual effective dose equivalent was 16,75 mSv. The main reasons are listed below:

1. Plant design: the highly radioactive systems are separated from the low ones. The capacity of the reactor-water cleanup system is relatively high. The main components are made of materials with low cobalt concentration.

2. The radiological work permit system is computer based. The most important information about different components, systems, and rooms, like dose rates and contamination levels, is stored in the memory. Radiological work permits, which have been carried out earlier, are also stored in the memory. Good prior knowledge of characteristics in the work objects makes careful work planning possible.

3. Work dosimetry system: to be able to limit doses, a good knowledge of the work activities that contribute to doses is needed. TVO has a very up-to-date microprocessor-based work dosimetry system equipped with seven readers and 350 dosimeters. The only way to perform satisfactory as-low-as-reasonably-achievable calculations is to know the exact work doses.

4. Outage planning: outages cause ~80% of all annual doses, and that is why special emphasis must be placed on advance planning. With careful planning, the annual outage time has been shortened to 2 to 3 weeks.

1. DOSE STATISTICS

TVO has one of the lowest collective doses among world's nuclear power plants. The dose statistics of annual collective doses, doses during refuelling periods and collective doses per GWe-yr at TVO Power Plant are set out in Table I. It can be seen in the table that on average more than 80 per cent of collective doses are received during refuelling periods.

For the sake of comparison it can be mentioned that TVO's annual collective doses represent less than 10 per cent of corresponding values in US-reactors. TVO's annual collective doses are even lower than the annual doses in similiar Swedish reactors of Asea-Atom design. TVO's highest annual collective dose per plant unit ever reached has been 0,94 manSv (94 man rem) and the lowest one for a calendar year with refuelling period 0,36 manSv (36 man rem). The average collective dose for a calendar year with refuelling period is 0,54 manSv (54 man rem).
The highest annual dose for an individual has so far been 16,75 mSv (1,7 rem). The average annual personal dose per two units for personnel exposed has varied from 0,52 mSv (52 mrem) to 1,41 mSv (141 mrem) which means that corresponding doses per one unit are clearly below 1 mSv (100 mrem).

TABLE I. ANNUAL DOSE STATISTICS. DOSES ARE GIVEN IN MAN SV.

Year	Collective doses		Refuel.shutdown doses		man Sv/GWe-yr	
	TVO I	TVO II	TVO I	TVO II	TVO I	TVO II
1978	0,01	-	-	-		
1979	0,22	-	0,10	-		
1980	0,50	0,03	0,37	-		
1981	0,49	0,13	0,43	0,06	0,95	0,32
1982	0,50	0,54	0,39	0,43	0,88	1,03
1983	0,57	0,36	0,49	0,28	1,04	0,61
1984	0,44	0,80	0,34	0,70	0,70	1,32
1985	0,38	0,58	0,27	0,47	0,61	0,94
1986	0,94	0,43	0,86	0,35	1,51	0,55

2. REASONS FOR LOW DOSES

There are several factors which can contribute to low doses. In the following the subsequent factors are dealt with; Administrative Factors, Directly Affecting Factors and Technical Factors.

A. ADMINISTRATIVE FACTORS

Outage planning. An important part of keeping doses to a minimum involves carefull preplanning of annual refuelling periods. Over the years the length of the refuelling periods has been reduced from 56 days to even 13 days. The minimization of doses in short refuelling periods is based on effective scheduling of job-functions and on minimizing the size of the workforce. TVO limits the size of workforce during refuelling period to a minimum by evaluating in advance the need of man-hours and manpower in each work. Furthermore about 80 % of workers are the same from year to year which also helps to minimize the size of workforce and time as workers are already familiar with jobs to be done, and with TVO's rutines and instructions.

Work permit and radiological work permit systems are computerised. All permits as well as the radiation and contamination levels of all components and rooms are registered in the computer's memory. Health Physics Tecnicians use this information to their advantage while planning radiation protection actions. Furthermore the work-specific dose statistics, which are received by the real time dosimeter system are at HP tecnicians' disposal.

B. DIRECTLY AFFECTING FACTORS

Work-dosimeter system. To limit doses it is essential to know which operations cause the highest doses. It should also be possible to divide doses into different phases of operations, to pin down those phases where doses can be limited. Personal doses at TVO Power Plant are controlled with individual TL-dosimeters, which form the basis of the dose registration. Doses of different work activities are controlled by an ADR-100 microprocessor-based reader with digital RAD-80 dosimeters, which are equipped with a dose display, a dose limit alarm and a memory of worker's ID-code. Armed with this equipment the worker can quite independently keep a chech on his personal dose. At the TVO Power Plant the work-dosimeter system is used routinely during work activities where the total dose is estimated to exceed 1 mSv (100 mrem).

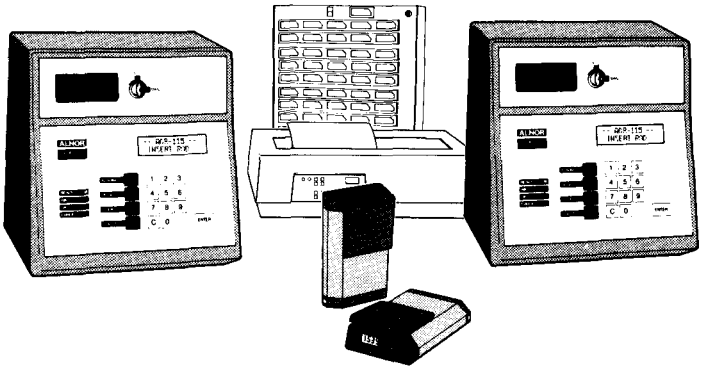

FIG 1. Work-dosimeter system

Limiting internal doses. In the case of airborne contamination TVO's policy is to limit and even avoid internal doses always when the airborne contamination exceeds 0,1 DAC (0,1 MPC_{40}). By using half face and full face respirators and even masks with fresh air supply the internal contamination has been managed to keep on a very low level, less than 0,5 per cent of ALI-values.

Good housekeeping practices. The amount of contaminated area in the plant is always kept to a minimum. At the moment there are 4-5 contaminated rooms per unit compared with the total amount of 1500 rooms. This results in greater accessibility to plant systems and lower background dose rates in those areas. Clean up of walls, floors and ceilings has been greatly helped by painting them with paint, which tolerates cleaning and decontamination.

C. TECHNICAL FACTORS

Separation. The main principle in layout design has been to instal radioactive and nonradioactive systems and components in separate rooms. Each room with radioactive components has its own access from the corridor. Valves in high radiation rooms, which must be operated manually, are equipped with remote regulating devices in the corridor through the wall. Maintenance of components has been taken into consideration by equipping rooms with necessary lifting devices as well as working platforms and ladders.

Material specification. Special attention has been paid to the selection of materials. All the internal parts of the reactor vessel are manufactured from stainless steel or Inconel with cobalt content less than 0,05 per cent. Stellite has not been used at all in the reactor. A lot of stainless steel has been used even in pipes which are in touch with reactor water. Such systems are for example Reactor Water Cleanup system, Reactor Shut Down system and Feed Water system.

Recirculation pumps and control-rod-drives. Systems which typically have very high external dose rates have been designed in such a manner that dose rates have been stayed on a very low level. By using internal recirculation pumps instead of external ones, the reactor containment volume is reduced by about 20 per cent. A continuous purge water flow through the control-rod-drives and the six recirculation pumps prevents the accumulation of reactor water impurities, i.e. ensures low contamination level. The occupational exposure per one served control-rod-drive and per one recirculation pump has been 1 mSv and 3 mSv, respectively.

Water chemistry. The primary circuit water is treated by two independent, coordinated cleanup systems, the Reactor Water Cleanup and the Condensate Cleanup system. The Condensate Cleanup flow is 100 per cent. The normal cleanup flow of Reactor Water Cleanup system is 2 per cent of steam flow but when necessary it can be doubled. High reactor water purity reduces the radioactive contamination of the primary systems, thus ensuring better accessibility and lower occupational radiation exposure.

High capacity factors. The average capacity factor of plant units has been since 1982 over 80 per cent and since 1984 even higher, 87 per cent. High capacity factors are the consequence of continuous operation without any other dose causing shutdowns than refuelling shutdown.

High integrity of uranium fuel helps to minimize contamination on surfaces of primary systems which also indirectly decreases occupational exposure. During the whole operational history of TVO there have in total been just four leaking fuel pins. All defective fuel pins have been removed from the core during the next refuelling period without any significant effect on the length of refuelling shutdowns. Jodine-131 concentration of reactor coolant is normally in order of 50-100 Bq/l ($1,4\text{-}2,7\times10^{-3}$ µCi/l).

HYBRID LOG-NORMAL ANALYSIS OF WORKER DOSES IN SPECIAL JOBS AT RESEARCH REACTORS JRR-2 AND JRR-3

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INTRODUCTION

The reduction effort proportionate to the magnitude of doses to workers is theoretically proved to result in the hybrid log-normal (HLN) type of dose distribution (1,2). The HLN distribution was applied to the analysis of distributions of worker doses in the United States(3). It showed that the shape of nationwide dose distributions by industry category changes from the lognormal (LN) to the normal via the HLN distribution according to the degree of exposure reduction.

This paper presents some properties of job-specific dose distributions rather than nationwide ones by using the HLN analysis.

HYBRID LOG-NORMAL DISTRIBUTION

The hybrid log-normal distribution is applied to the distribution of doses X so that $Y=\ln\rho X+\rho X$ ($\rho>0$) is normally distributed with parameters (μ, σ^2). The HLN distribution function, $\Omega(x)$, is given by

$$\Omega(x) = \int_0^x \frac{1}{\sqrt{2\pi}\,\sigma}\left(\frac{1}{x}+\rho\right)\exp\left(-\frac{\{\ln\rho X+\rho X-\mu\}^2}{2\sigma^2}\right)dx . \qquad (1)$$

The HLN distribution is derived from a stochastic process of reducing dose accumulation as shown in Fig. 1. The dose ΔX_j at the j-th step is proportional to the previous amount X_{j-1} and the feedback mechanism to reduce ΔX_j controls the factor ε_j so as to decrease it based on the magnitude of ΔX_j: ε_j varies randomly depending on a role and behavior of workers at workplace in terms of dose rate and working time. The sum of ε_j over n hypothetical steps in a fixed working period is

$$\sum_{j=1}^{n}\varepsilon_j = \sum_{j=1}^{n}\left(\frac{1}{X_{j-1}}+\rho\right)\Delta X_j = \int_{\rho X_0}^{\rho X_n}\left(\frac{1}{t}+1\right)dt = \left[\ln t + t\right]_{\rho X_0}^{\rho X_n} . \qquad (2)$$

According to the central limiting theory, as the standardized sum of ε_j will be normally distributed when $n\to\infty$, $\ln\rho X_n+\rho X_n$ will be normally distributed (2). This HLN genesis theory implies that the HLN type dose distribution will approach the LN for workers weakly controlled to reduce dose ($\rho\to 0$) and approach the normal for those strongly controlled ($\rho\to\infty$) as shown in Fig. 2.

APPLICATION METHOD

The HLN probability plot is used to interpret the effect of dose reduction on worker dose distribution. The LN probability plot uses a log scale of X on probability paper. Analogously the HLN probability plot uses a hybrid scale of dimensionless dose ($t=\rho X$) multiplied by ρ which means the degree of dose reduction per unit dose according to dose limitation. The hybrid scale is a function defined by lnt+t (

see in Fig. 2). It approaches a log scale for $t \ll 1$ and a linear scale for $t \gg 1$. Thus HLN probability paper is similar to LN probability paper for $t \ll 1$ and to normal one for $t \gg 1$.

The data used were obtained in a five-month series of works of repairing primary coolant valves at JRR-2 and in one year series of works of removing primary coolant equipments in the remodelling of JRR-3, as shown in Table 1. The radiation works had been well planned and prepared for a resonable goal of balancing the completion of radiation jobs and the reduction of worker doses.

RESULTS AND DISCUSSION

Minimum and maximum exposure rates of each of 74 working days at workplace of JRR-2 are plotted on HLN probability paper in Fig. 3, respectively as graphs A and B located in the region of $t \ll 1$. Their lognormality implies no effort of reducing high exposure rate day by day. However as the daily worker doses give the graph C extending $t > 1$ in Fig. 3, that hybrid lognormality implies an effective effort proportionate to the magnitude of dose to reduce the frequency of worker doses approaching the control level of 0.5 mSv/day. Besides as cumulative doses of 13 workers give the graph D located in the region of $t \gg 1$ in Fig. 3, it implies a strong effort of reducing doses not to approach the control level of 4 (later 5.5) mSv. Thus it suggests that job-specific efforts of dose reduction lead to the HLN distribution of doses to workers at workplace with the LN distribution of exposure rates.

Daily and cumulative doses to 78 workers at various workplaces of JRR-3 show parallel rows of plots extending from $t \ll 1$ to $t > 1$ on HLN probability paper in Fig. 4 (a). Their hybrid lognormality reflects a wide variety of doses and implies an explicit effort of dose reduction proportionate to the magnitude of dose approaching the control level of 1 mSv a day or 0.5～10 mSv in the whole period. Fig. 4 (b) shows HLN probability plots of four groups of daily doses according to the range of control levels of dose accumulation applied to workers. The degree of graphs extending to $t > 1$ implies the stronger effect on dose reduction due to the higher control level of dose accumulation, except the group of technical supervisors(▲) who probably took the strong self-control over the whole working period below the control level from 2.5 to 5 mSv. Another grouping of daily dose distribution, by period, exposure rate and sub-job, was also proved to be similar HLN effects on dose reduction.

CONCLUSION

The job-specific dose distribution as well as the nationwide dose distribution seems to be hybrid lognormality. The HLN probability plot depicts the effect of dose reduction according to the magnitude of doses approaching the control levels set for individual jobs.

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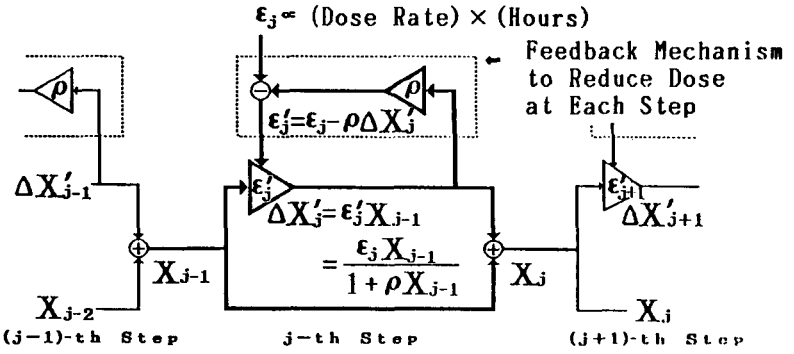


Fig. 1 A mathematical model for the control process of dose reduction

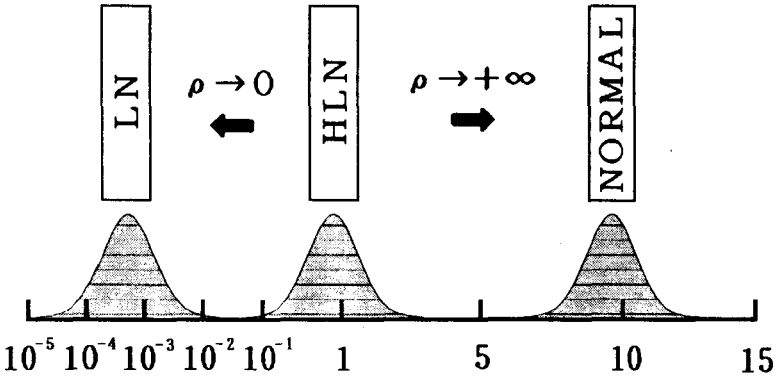


Fig. 2 Hybrid scale: hyb(t)=lnt+t. Putting t=ρX, X changes from a log scale to a linear scale according to ρ.

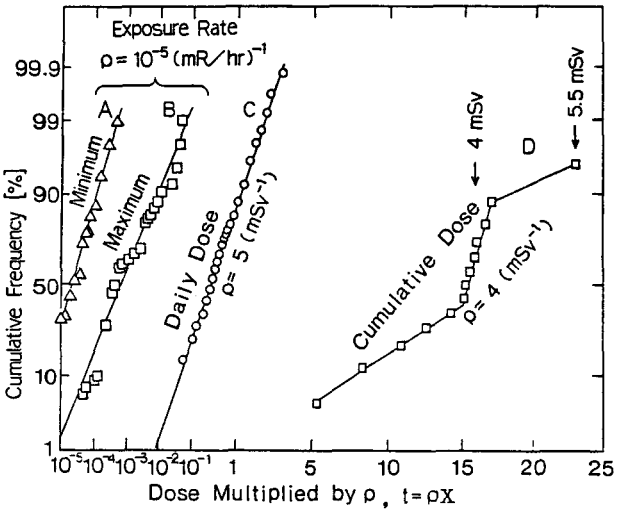
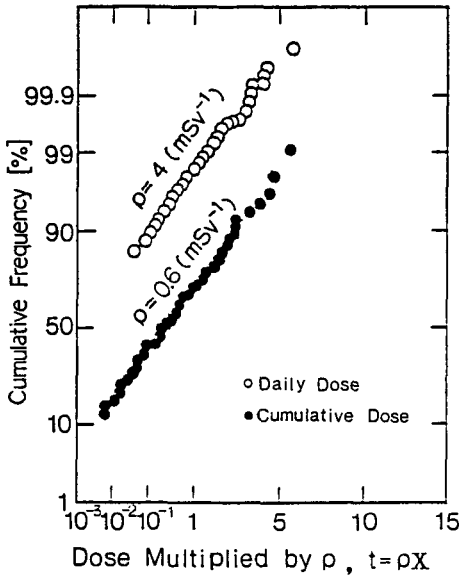


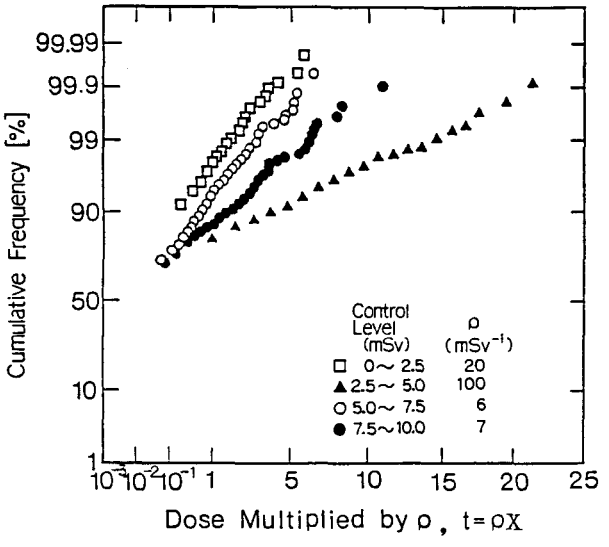
Fig. 3 HLN probability plots of exposure rates and worker doses at JRR-2.

Table I Radiation works at the research reactors to be analyzed

Reactor	JRR-2	JRR-3
Job	repairing primary coolant valves	removing primary coolant equipments
Period	May, 1980～Sept., 1980	Aug., 1985～Mar., 1986
Number of workers	13 workers about 7 workers/day total 470 worker·days	78 workers about 50 workers/day total 8000 worker·days
Control level	0.5 mSv/day 4 (later 5.5) mSv/man	1 mSv/day 0.5～10 mSv/man 200 man·mSv
Resultant dose	average 3.5 mSv total 46 man·mSv	average 15 mSv total 116 man·mSv



(a)



(b)

Fig. 4 HLN probability plots of daily and cumulative doses at JRR-3:
(a) daily doses and cumulative doses over the whole period,
(b) four groups of daily doses according to the range of control levels of dose accumulation applied to workers .

Seventh International Congress of the
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RADIATION PROTECTION PRACTICE

VOLUME II

Organised by the Australian Radiation Protection Society

ELECTROMAGNETIC ENERGY DEPOSITION AND MODELS OF HUMAN THERMOREGULATION

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ABSTRACT

Thermal energy deposition in tissue is one of the important and perhaps the most important consequence of exposure to electromagnetic fields. As our insights advance so does the recognition that both the deposition of radiofrequency energy and the resulting rise in tissue temperature is sufficiently inhomogeneous to warrant special concern. Measurement of local deposition and of resulting tissue temperature is both technically difficult, and severely invasive. As a result there has been an increasing reliance on mathematical simulation models to evaluate both the differential deposition and the local temperature rises which can result. Given the controversial nature of the interaction between non-ionizing radiation and human tissue it is important to recognize the limitations of such modeling efforts, especially in the absence of experimental validation of any of the predictions made. Specific considerations are:

1. The more detailed the model the less confidence one should have in unvalidated predictions.
2. The more detailed the model the higher the highest predicted local temperature will be.
3. The models are usually the product of a group which has recognized expertise in either electromagnetic energy absorption or in biothermal modeling, but not in both.
4. The models and their predictions are usually reviewed before publication by one of the two disciplines, but not by both. As a result claims or implications of validity of the predictions are not properly reviewed.
5. Given the controversy surrounding the issue improperly reviewed publications are likely to be given more weight than they deserve.

CONSIDERATIONS INVOLVED IN THE SPECIFIATION OF LIMITS ON HUMAN EXPOSURE TO ELECTRIC AND MAGNETIC FIELDS AT FREQUENCIES BELOW 300 GHz

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ABSTRACT

In 1982 and again in 1986 the National Radiological Protection Board of the UK issued consultative documents containing proposals for advice that it might give on limiting the exposures of workers and members of the general public to electromagnetic fields with frequencies below 300 GHz. At frequencies below a few hundred kilohertz these were based on limiting the electric currents and current densities in the body, and above these frequencies on limitation on the power absorbtion in the body. The considerations underlying the proposals will be described and the responses received during the consultation process. The points at issue are the specification of acute effects which may reasonably be regarded as hazardous and the determination of thresholds for these effects, the determination of the thresholds for harm to the developing fetus, and whether there are any long term effects, in particular cancer induction or promotion from exposures to electromagnetic fields of low intensity. The nature of experiments needed to address these issues will be outlined.

TIME VARYING MAGNETIC FIELDS AND DOSIMETRIC PRINCIPLES

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INTRODUCTION AND BASIC PRINCIPLES

In a broad sense, the term "dosimetry" is used to quantify an exposure to radiation. Quantitative descriptions of an exposure for the purpose of formulating protection standards and exposure limits require the use of adequate quantities. "Adequate" means that the quantities should represent, as well as possible, those physical processes which are closely linked to the biological effects of the fields.

From the analysis of established mechanisms it can be concluded that for time-varying magnetic fields the induction is predominant. The appropriate dosimetric quantity which represents the physical processes that are closely linked to the biological effects of time-varying magnetic fields is the induced electric field strength at the cellular level in the living tissue or - connected with the specific conductivity of the medium - the induced eddy current density (Bernhardt, 1979, 1985). By comparing the current densities, it may be possible to predict effects in human beings from those found in studies on animals and isolated cells. In this context, it is irrelevant whether the current density surrounding a cell is introduced into the body through electrodes or induced in the body by external magnetic fields. However, the current paths within the body may be different. The evaluation of human exposure using current densities is based primarily on a concept of "dose" to the critical organs. Basic protection limits can be expressed in permissible current densities; derived protection limits can be expressed as exposures to external magnetic fields. To fully assess the data obtained in bioeffects research, exposure conditions must be well controlled and measured. In this case, the "dosimetry" in bioeffects research with magnetic fields is very complex, since all relevant factors must be taken into account. Factors affecting interaction of magnetic fields are (UNEP 87):

- Parameters of the magnetic field source (frequency, modulation (pulse, AM, FM), rise and decay times (dB/dt), polarisation, field strength, field pattern (uniformity), surrounding material properties);
- Parameters related to exposure (tissue properties (conductivity, anisotropy, permeability), size, geometry, orientation relative to polarization, mode of exposure (partial; whole body));
- extraneous factors (metal implants (ferromagnetic), metal objects in the field, drugs (medications), chemical pollutants).

INDUCED ELECTRIC FIELDS AND CURRENT DENSITIES; RESPONSE OF BIOLOGICAL SYSTEMS

In accordance with Faraday's law, magnetic fields that vary in time will induce potentials and circulating currents in biological systems:

$$J = \sigma E = \frac{1}{2} r \sigma \frac{dB}{dt}$$

where

J = current density (A/m^2),
E = induced electric field strength (V/m),
r = radius of the loop (m),
σ = tissue conductivity (S/m),
$\frac{dB}{dt}$ = rate of change of magnetic flux density B (T/s).

For sinusoidal fields of frequency f the equation reduces to $J = \pi r \sigma f B_o$, where B_o is the magnetic field amplitude.

Thus, the magnitude of the induced electric fields and current densities is proportional to the radius of the loop, the tissue conductivity and rate of change of magnetic flux density. The dependence of the induced field and current on the radius of the loop through which magnetic flux linkage occurs is an important consideration for biological systems. The magnetically-induced electric field strengths and corresponding current density are greatest at the periphery of the body where the conducting paths are longest, whereas microscopic current loops anywhere within the body would have extremely small current densities. The magnitude of the current density is also influenced by tissue conductivity where the exact paths of the current flow depend in a complicated way on the conducting properties of the various tissues. Different authors give different values for the low-frequency conductivity, e.g., for the myocardial tissue and the nerve tissue (white and grey cerebral substance). Additionally, high ratios of the transverse to longitudinal impedance up to 7:1 were observed. The anisotropy of tissue conductivity makes the applicability of phantom techniques using isotropic tissue substitutes questionable. It should be remembered that the current tends to follow routes of least resistance and that the exact current paths of the magnetically induced currents and the exact magnitudes of the current densities are not known and that only rough approximations can be made.

An important factor to be considered in the response of biological systems to the time-varying magnetic field is the waveform. Many different types of magnetic field waveform are used in practice, including sinusoidal, square-wave, saw tooth, and pulsed fields. For these fields, the two parameters of key importance are the rise and decay times of the signal, which determine the maximum time rates of change of the field (dB/dt) and hence the maximum instantaneous current densities induced in tissues. Rms values are often used in the context of electric current effects to give an "effective" value for a variety of waveforms. The peak instantaneous field strengths appear to be important for nerve and muscle cell stimulation, or for perturbing cell functions. The effects strongly depend on the frequency.

A potentially important target of ELF magnetic field interactions is the nervous system. From a consideration of the naturally occurring fields in the central nervous system, it can be concluded that magnetic fields in the 1-100 Hz frequency range, which can induce current densities in tissue of approximately 1 mA/m^2 or smaller, should not have a direct effect on the brain's electrical activity (Bernhardt, 1979). Induced fields sufficient to exceed a threshold depolarisation value can result in an action potential. These effects are well understood. ELF magnetic fields inducing such large depolarisations may result in nerve stimulation or muscle contraction, or even in fibrillation. ELF magnetic fields inducing weak electric fields may also interact with, or modulate, nervous system activity in a manner less well understood, resulting in changes in electrical excitability. Such interactions may be involved in, for example, magneto-phosphenes. These interactions show frequency threshold characteristics of nervous tissue, and have been documented (Bernhardt 1979, 1985). With regard to "hazardous values" and the upper limit of the field

strength that leads to injury, the ultimate criterion for the definition of injury may be the initiation of heart fibrillation. The threshold for extrasystole induction at 60 Hz is estimated to be above 300 mT for stimulation times of 1 second or longer, and the threshold for ventricular fibrillation is higher by a factor of 3 - 5 (Bernhardt, 1985). For shorter exposure times, higher field strengths are necessary to produce similar biological effects.

CONCLUSIONS AND RECOMMENDATIONS FOR FURTHER RESEARCH

As stated above, it is difficult to correlate external magnetic field strengths with induced tissue current densities. However, using "worst case" assumptions, an estimate of the order of magnitude for "safe" and "dangerous" magnetic field strengths and their frequency dependence can be made, especially for the critical organs brain and heart (Bernhardt, 1985).

The table summarizes induced current density ranges between 3 and 300 Hz for producing biological effects (UNEP, 87) and values for the magnetic flux density for inducing approximately these current densities at 50 or 60 Hz

Current density mA/m^2	Effects	Magnetic flux density mT
> 1000	extra systoles and ventricular fibrillation possible; definite health hazards	> 500
100 - 1000	Changes in central nervous system excitability established; range, where stimulation of excitable tissue is observed; possible health hazards.	50 - 500
10 - 100	well established effects, evident visual (magnetophosphenes) and possible nervous system effects; induction of bone reunion reported	5 - 50
1 - 10	minor biological effects reported	0.5 - 5
< 1	absence of well established effects	< 0.5

More experimental data are necessary, which include

- the influence of the anisotropy and of inhomogeneities of the tissue conductivity on the induced current densities;
- influences of field gradients on the induced current densities in homogeneous and inhomogeneous tissues;
- specification of the "worst case" conditions and which current densities and current loops are maximally possible;
- the frequency dependence of threshold values for current densities which produce significant biological effects;
- threshold for modulation of nervous systems activity and excitation thresholds especially for single unidirectional pulses or pulse sequences and

connection between the duration of pulses and the rates of change of the magnetic flux density.

On the other hand, several recent epidemiological reports suggest an association of exposure to very weak magnetic field with an increase of the incidence of cancer among children, adults and occupational groups. These correlations cannot be explained by the present knowledge for interaction mechanisms of magnetic fields. Because up to now the available data are inadequate to conclude that magnetic field exposure alone is the reason for the observed cancerogenic effects there must be considerable further study before these reports can be accepted. Similar conclusions can be performed with regard to reports describing teratogenic effects of pulsed magnetic fields on chicken embryos or describing an increase in external malformations among mice exposed to weak pulsed magnetic fields. Although other groups found similar results we are far away from understanding these effects especially the causal relationship with the exposure to magnetic fields.

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RADIOFREQUENCY POWER DEPOSITION DURING MAGNETIC RESONANCE DIAGNOSTIC EXAMINATIONS

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INTRODUCTION

Magnetic Resonance Imaging and Spectroscopy (MRI, MRS) require that subjects be exposed to a radiofrequency field, and the corresponding energy absorption leads to tissue heating. The main question, thus, to be considered in connection to safety and health aspects is related to the specific absorption rate (SAR) in the imaged subject and the exposure durations which might put a practical limit on the pulse sequence which can be used (1).

This explains why current safety regulations applying to MRI and MRS examinations give information on levels of exposure to radiofrequency electromagnetic fields associated with these diagnostic procedures (2). In particular, IRPA/INIRC is issuing guidelines giving some guidance on the magnitude of temperature increases at which no adverse health effects are expected and on the corresponding acceptable cumulative exposure (SAR times Exposure duration).

In this paper some models and experimental results for radiofrequency power deposition in MRI and MRS machines are reviewed. Models show that energy dissipation is a function of the frequency, RF incident power density, exposure duration, coupling between the RF coil and the subject, and several properties of the exposed tissue, including conductivity, dielectric constant, specific gravity, size, and orientation relative to the field polarization. The ability of the body's normal thermoregulatory responses to cope with high levels of RF energy deposition must be also taken into account (3).

THEORY

Simple theoretical estimates of the average, maximum, and spatial variation of the radiofrequency power deposition in terms of SAR during proton nuclear magnetic resonance imaging have been recently deduced for homogeneous spheres and for cylinders of biological tissues (4-6). Results have been obtained with uniformly penetrating linear RF fields directed axially and transverse to the cylindrical axis. In the RF frequency range below 100 MHz and under the near-field geometric conditions presently used in MRI devices, approximately 90% or more of the absorbed energy results from tissue currents induced by the magnetic component of the field (7). The specific absorption rate is related to the average induced electric field, $E/\sqrt{2}$, the tissue conductivity, σ, the tissue den-

(o) Member, IRPA International Non Ionizing Radiation Committee

sity, ρ , and the duty cycle, D, of the applied RF field by the equation:

$$SAR = \sigma E^2 D/2\rho \qquad (1)$$

The duty cycle D is equal to t/T, where t is the pulse duration and T is the pulse repetition interval. Being the magnetic component of the field dominant in the low range of RF frequencies used for MRI, the sinusoidal induced electric field for a circular loop of tissue of radius R is $E = \pi\nu RB$, in which B is now understood to be the amplitude of magnetic component of the RF field perpendicular to the plane of the cross-section. In addition, the rotating component of the field, B_1, must satisfy the Larmor resonance condition. For an arbitrary flip angle, θ , SAR is given by the equation:

$$SAR = \frac{4kR^2\sigma\theta^2\nu^2}{\pi^2\rho tT} \qquad (2)$$

where k is equal to 6.81×10^{-19} in SI units.

The SARs predicted by Eq.(2) are plotted for a human torso model and for a human head model in Fig.1. The SAR at frequencies from 1 to 100 MHz is presented as a double logarithmic plot with $tT = t^2/D$ as the abscissa. It is evident that the SAR increases as the pulse repetition interval (T) decreases and/or the duty cycle (D) increases. At 30 MHz the SAR in the torso model equals the average body basal metabolic rate in the resting condition when t^2/D is approximately 10^{-5} s^2. The quadratic dependence of the SAR on the loop radius as shown in Eq.(2) has been confirmed experimentally (5). These diagrams provide useful interface between MRI and MRS spectrometer operating conditions and safety guidelines for RF power deposition.

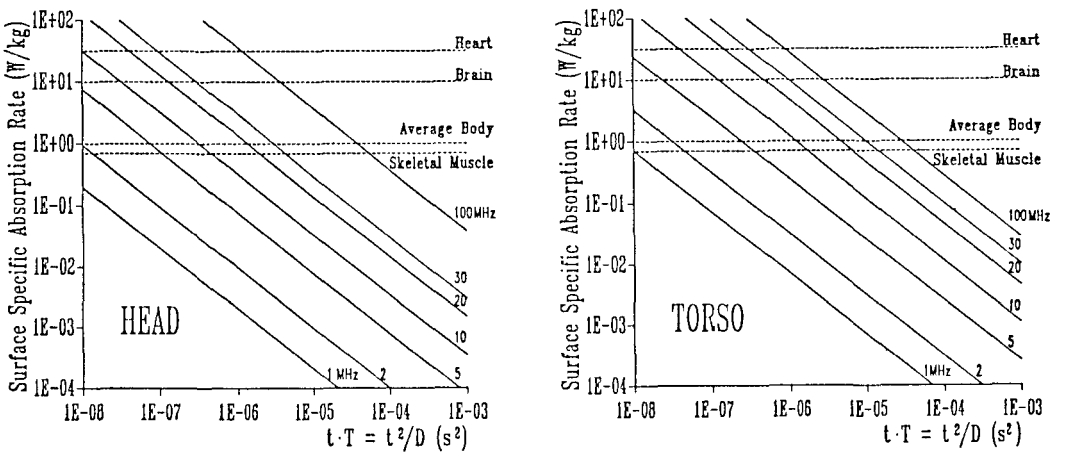

Fig.1 - Surface specific absorption rates predicted by Bottomley et al (5) for human head and torso models

A number of numerical techniques have also been developed to evaluate SAR in the human body, for which an analytical solution is impossible due to the complicate geometry. This techniques are gen-

erally based on the subdivision of the body, or parts of it, in cells small enough to assume that both the field intensity and the dielectric properties of the tissue are constant troughout each cell.

This approach in general requires large computer memories and long processing time. Recently a method has been developed which considerably reduces these requirements. This is based on the modeling of portions of the body using an impedance network. The application of circuit theory allows to determine internal currents, and therefore SAR, based on the knowledge of the dielectric constant and conductivity of each cell (8-10). The impedance method has been successfully applied to the evaluation of power deposition in magnetically induced hyperthermia. At present it is being exploited also for the evaluation of SAR in MRI, both in the ideal case of uniform magnetic field and in that of field generated by a real applicator.

EXPERIMENTAL DOSIMETRY STUDIES

Experimental measurements of SAR have been performed using a number of techniques, including infrared thermography and direct temperature measurements both in living specimens and in phantoms filled with saline or tissue-equivalent materials.

The simplest method to determine the SAR associated with the RF fields used in MRI and MRS is to measure the change in the quality factor Q of the coil upon introduction of the specimen. Bottomley et al (6) adopted this procedure to compare previous theoretical estimates with the experimental total power deposited in the bodies of nine adult male volunteers. The results for the average power deposition agree within about 20% for the exact model of the cylinder with axial field, when applied to the exposed torso volume enclosed by the RF coil. The average values predicted by the simple spherical and cylindrical models with axial fields, the exact cylindrical model with transverse field, and the simple truncated cylinder model with transverse field were about two to three times that measured, while the simple model consisting of an infinitely long cylinder with transverse field gave results about six times that measured.

Recently, Guy and Shuman (11) performed MRI dosimetry studies based on infrared thermography using a full scale phantom human body and spherical head models. The technique involved the fabrication of a phantom model with a synthetic liquid tissue for quantifying average SAR and a gel tissue for determining SAR distribution within the exposed body. The resulting change in temperature is related to the SAR by the known thermal properties of the synthetic tissue, and a useful map of the energy deposition pattern can be obtained.

Superficial and deep tissue heating during high specific absorption rate RF irradiation has been measured in dogs to see if significant temperature changes could be produced by a clinical MRI device operating at 1.5 T (12). Temperature probes were placed in the subcutaneous tissues of the dog axilla and groin, in the muscle of the neck and thigh, in the liver and mid-peritoneum, and in the urinary bladder. Temperatures were recorded before, during, and af-

ter a 26 minute spin-echo sequence. During RF exposure, in each dog all tissues experienced a linear temperature change of several degrees; maximal average change was 4.6° C in the urinary bladder. Deep tissue heating was slightly greater than that of superficial tissues.

CONCLUSIONS

Many authors have determined, both theoretically and experimentally , the RF power deposition within patients and animal models. These studies have shown that SAR levels resulting from human exposure to maximum possible MRI and MRS RF magnetic fields can exceed limits currently recommended.

It appears possible that multi-echo, multi-slice techniques may well deposit dangerous amounts of power. According to Gandhi et al (10), hot spots can also be expected due to constriction of current paths associated with some anatomically thin regions not insulated from large current paths.

The main conclusion is that when imaging infants, cardiac compromised patients, and patients with altered thermoregulation, particular attention should be paid to SAR. In addition, relatively low room temperature and humidity should be maintained along with good airflow through the magnet bore (8).

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SAFETY OF THE PATIENT DURING IN VIVO MAGNETIC RESONANCE EXAMINATIONS. RATIONALE FOR THE INIRC/IRPA GUIDELINE.

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IRPA/INIRC is developing guidelines on safety of in vivo magnetic resonance (MR) diagnostic examinations,imaging (MRI) or spectroscopy (MRS), which involve exposure of the patient to static (SMF) and switched gradient (time-varying) magnetic fields (TMF), and radio frequency (RF) fields. The aim of this paper is to relate mechanisms of interactions, clinical endpoints and exposure conditions to safety concerns. Simplifications were made; more information can be found in Blank and Findl (1987), Tenforde and Budinger (1987), WHO-UNEP-IRPA (1987); quantities and units are defined in IRPA (1985). Current MR devices use SMF of 0.5 to 1.5 T, a few up to 5 T. TMF range from 1 to 10 mT with complex wave forms at frequencies from below 300 Hz up to a few kHz, and a time rate of change in magnetic flux density (dB/dt) up to 6 T/s. RF exposures occurr mostly in the frequency range from 4 to 200 MHz and result usually in a specific absorption rate (SAR) below 0.4 W/kg, examinations at higher SARs were also performed. MR examinations can be completed under one hour, and may be repeated a few times over the patient's lifetime. The considerations below refer only to such conditions, and do not apply to occupational or public exposure.

3 classes of interactions of SMF with living systems can be distinguished: 1. electro-, and magnetohydrodynamic, 2.magnetomechanical, 3. at the atomic and nuclear levels. Electrodynamic phenomena consist in the interaction of steady ionic currents with applied SMF according to Lorentz's force law. An electric (E) field is induced when an electrolyte flows through an insulated channel (eg. blood in a blood vessel), or a container filled with electrolyte (body, or body part) moves in the presence of SMF. The magnitude of induced potentials depends on the diameter of the vessel (size of container), flow (movement) velocity, the orientation and strenght of SMF. These also can deflect ionic currents within cells, and, in particular, conduction currents in excitable membranes of nerve and muscle cells. Magnetically induced potentials lead to changes in ECG at 0.1 T which increase with SMF strenght. Concerns were raised that at 2.5 T in adult healthy humans E field may reach depolarization potential for heart fibers, other calculations indicate that 10 T are still below fibrillation response levels. These are not known for patients with altered heart conduction because of disease or medication. Experiments on isolated neurons did not reveal any effects on conduction at 2 T, theoretical estimates indicate that 24 T are needed to deflect conduction currents. However, fields below 2 T may affect impulse propagation in asymmetric conductor loops, which may explain observed transient effects on acoustic induced potentials in the brain.

Theoretically, 10 T may restrict ion transport through the cell membrane. Magnetohydrodynamic effects consist in retardation of axial velocity of an electrolyte solution flowing through SMF. This leads to an increase in arterial blood pressure. Negligible hemodynamic perturbations are expected below 2 T, a 7% reduction of aortic flow velocity is expected in an adult human at 5 T. Magnetomechanical interactions do not play a role in biological mechanisms of concern, constitute however, the basis for the most hazardous aspect of MR: effects on ferromagnetic inclusions in the body and attraction of metallic objects by the magnet (see below). Interactions at the atomic and nuclear levels lead to interference with biochemical reactions, particularly those, which involve electron transfer processes via radical pair intermediates. Under MR examination conditions this mechanism is of little concern.

TMF induce potentials and circulating currents in accord with Faraday's law, i.e. effects can be discussed in terms of E field strengths and current densities (J). Their magnitude increases with the radius of the inductive loop and dB/dt. At low frequencies the interior of the cell is shielded, and the most likely site of interaction is the cell membrane. Many cell functions are affected even at low E and J values. Under MR examination conditions effects on the heart conduction system, muscle and nerve cells (electrical neuromuscular stimulation) are of concern. Thresholds for effects depend upon electrical and electrophysiological properties of tissues, and are frequency and waveform dependent. For MR TMF exposure conditions a general statement can be made that at J between 0.1 and 1.0 A/m^2 changes in the bioelectrical activity of the brain may occur, the threshold for sensory stimulation lying somewhere in this range. At J above 1 A/m^2 immediate health hazards exist, associated with the increasing possibility of inducing cardiac fibrillation (most quoted threshold 3 A/m^2, possibly lower in pathologic states) and continous (tetanic) muscle contractions. The most widely known effect of TMF exposure, magnetophoshenes, i.e. sensation of light flashes perceived during exposure of the head, is unlikely to occur under MR examination conditions.

Several mechanisms have been proposed for the interaction of RF fields, an established and noncontroversial one is dielectric heating. There is a wide concensus that exposure limits at frequencies above 100 kHz have to be established primarily from thermal considerations. Physiological considerations indicate that an increase of $1^{o}C$ in core temperature does not pose immediate health hazards to individuals with an unimpaired blood flow from the core to the skin, and normal evaporative heat loss through respiration and sweating. The increase in temperature during MR examination depends on RF energy depositon and the rate of heat loss. Nonuniformity of RF energy absorption in the human body may lead to an elevation of temperature over a limited volume (hot spot) not accompanied by an increase in core temperature.

The above mechanistic considerations supported by a large body of experimental data allow one to identify the cardiovascular, the central and the peripheral nervous systems as critical

ones for immediate adverse reactions due to SMF and TMF interactions. Tolerance to RF exposure is dependent upon the function of the cardiovascular and respiratory systems in thermoregulatory responses induced by systemic heating. Another limiting factor is the possibility of focal thermal injury due to RF hot spots. Thus physiological considerations seem to be suitable primary criteria for the evaluation of MR exposure conditions, from which secondary criteria in terms of physical parameters of exposure fields may be derived. IRPA/INIRC proposed the following guidelines for the assesment of MR safety:

SMF: No adverse effects have been observed nor are expected from exposures of the head and/or trunk to 2 T, or of the limbs to 5 T. Exposures of the head and trunk above 2 T require an assesment of the potential risk vs the likely benefit. Whole body exposures over 5 T may pose hazards for poeple with cardiovascular disease. Monitoring of cardiovascular function must be undertaken whenever exposures above 2 T occur. All exposures must be limited to fields below 10 T.

TMF: No adverse effects are expected when the dB/dt in the region occupied by the head or trunk does not exceed 3 T/s for the duration of change in magnetic flux density longer than 10 ms. For shorter periods dB/dt may be increased, provided that the product of $(dB/dt)^2$ and the duration of the change of magnetic flux density in seconds is less than 0.09. In the region occupied by the limbs dB/dt should not exceed twice the above values.

These values incorporate a safety factor of about 100 in respect to cardiac fibrillation threshold, and an even larger one in respect to peripheral neuromuscular stimulation. In the U.K. (NRPB, 1983) and the U.S. (Czerski and Athey, 1987) 20 T/s were indentified as a level at which no adverse effects are expected. At this level the safety factor for neuromuscular stimulation is about 3, and about 10 for cardiac fibrillation.

RF: For whole body or head and trunk exposures no adverse effects are expected if the increase in body temperature does not exceed $1^{o}C$, except for infants, pregnant women, and persons with cardiovascular deficits, in whom it is desirable to limit temperature increases to $0.5^{o}C$. IRPA/INIRC derived SARs for exposures of the head (4 W/kg limited to 60 Wmin/kg), trunk (8 W/kg limited to 120 Wmin/kg) and extremities (12W/kg limited to 180 Wmin/kg) at which temperature increases remain below these values. However, users of MR devices usually do not have the resources to determine RF energy deposition. Reliable and detailed data have to be obtained from the manufacturer.

A special case is the the examination during pregnancy and early childhood. MR in vivo studies of the fetus, pregnant women, newborn and infants should be limited to cases where direct benefit tothe patient will be derived in terms of diagnostic information not obtainable by alternate methods. The fetus may be sensitive to RF and magneetic fields. MRI is not not likely to provide useful information, so that examination of pregnant women during the 1^{st} trimester should be justified by benefits to the mother.

Exposure durations should reduced to mininum, and RF energy deposition should be kept at levels which minimize temperature increases, which should not exceed 0.5^{o}C.

Because of magnetic or RF interference examination of persons who have electrically, magnetically or mechanically activated implants (eg cardiac pacemakers) or rely on life support systems is contraindicated. Patientss with ferromagnetic aneurysm clips or metallic implants (eg intrauterine contraceptive devices, large hip prostheses) are also contraindicated.

A important aspect of MR safety are ancillary (collision and electromagnetic interference) hazards. The field near the magnet may be strong enough to pull ferromagnetic objects along the axis of the field. Thus, metallic objects can become dangerous projectiles. Various medical and non-medical equipment and magnetic data carriers may be affected. The extent of zones in which such hazards exist should be established, and warning signs posted. In view of such problems institution of resuscitation and intensive care of the patient in the examination room is impracticable. Proper emergency procedures should be developed.

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THE MEASUREMENT OF BODY CURRENTS INDUCED BY RADIO FREQUENCY FIELDS

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ABSTRACT

At frequencies below about 100 MHz proposals for the limitation of exposures to electromagnetic fields proposed by the National Radiological Protection Board are based on a mixture of considerations limiting both electric current densities in the body, currents in the arms and legs and specific absorption rates. One effect of these proposals is to require the limitation of currents in the legs to about 100 mA. To give practical effect to the proposed exposure limitations it was necessary to measure these currents under realistic exposure conditions using various measurement techniques. The results of the measurements carried out in the vicinity of HF radio transmitters and industrial RF heat sealing equipment are discussed.

A SURFACE ACOUSTIC WAVE ELECTRIC FIELD STRENGTH METER FOR ENVIRONMENTAL STUDIES OF HV TRANSMISSION LINES

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INTRODUCTION

In recent years, there has been a significant increase in concern over the health and safety aspects of high voltage transmission lines (HVTL). The majority of research has focused on effects directly or indirectly involved with the central nervous system, including physiological, ultrastructural, and biochemical alterations, changes in blood composition , behaviour, reproduction, and development (1). Several recent epidemiological reports have presented preliminary data suggesting an increase in the incidence of cancer among children and adults exposed to magnetic fields through living close to various types of electrical power lines or devices (2).

With the increase in environmental concerns there has been a concomitant consideration of biological effects and health implications relative to presently existing HVTL and those planned in the future. It was concluded that the electric and magnetic field strengths and the electrical discharges are the most important electrophysical factors. Thus, it has been deemed necessary to develop measuring means to determine the field strengths in areas surrounding electric installations, in particular at ground level.

In the present paper an electric field meter, based on the use of a surface acoustic wave (SAW) delay line, is presented and the experimental results obtained are discussed.

PRINCIPLES OF OPERATION

The electric field strength meter here analyzed is based on the use of a surface acoustic wave voltage sensor (3,4). A schematic diagram of the SAW voltage sensor is shown in Fig. 1.

(o) Member, IRPA International Non Ionizing Radiation Committee

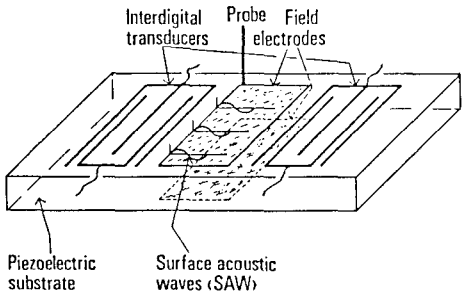

Fig.1 - Schematic diagram of the SAW ac voltage sensor

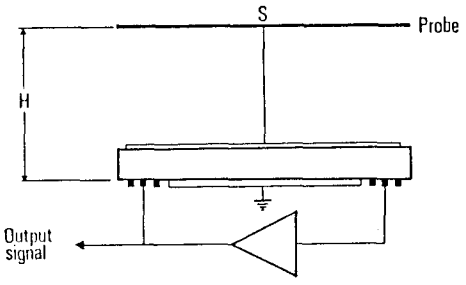

Fig.2 - Schematic diagram of the SAW E-field meter

It consists of a SAW delay line implemented on a higly piezoelectric substrate. Surface acoustic waves are generated and detected by interdigital transducers (IDT) and the acoustic propagation path is covered with a thin film electrode connected to the ground. Ground potential is also the average potential of IDTs. The voltage to be measured is applied to the field electrode deposited on the opposite surface of the substrate, and gives rise to an electric field in the substrate. This, in turn, produces a change in the acoustic propagation velocity through the nonlinear electroelastic effect.

Any change in the acoustic propagation velocity can be detected by monitoring the oscillation frequency of the acoustic delay line when connected to an amplifier to configure a SAW controlled oscillator.

The device is suitable for the measurement of ac voltages, provided that the time delay of the acoustic line is small if compared to the period of the ac voltage (5). In this case, the output of the device is frequency modulated by the applied voltage, and the voltage sensor can be employed for the detection of electric fields when connected to a proper coupling structure, as shown in Fig. 2.

EXPERIMENTAL RESULTS

Experiments were performed on a SAW voltage sensor made on a Y cut, X propagation $LiNbO_3$ substrate. The interdigital transducers consist of 10 finger pairs with a spatial periodicity equal to 35.2 μm. The distance between the transducers is 10.5 mm and the aperture of the acoustic beam is 2 mm. The device was operated both at the fundamental frequency of 106.45 MHz and at the 7th harmonic (745.15 MHz). Thickness of the $LiNbO_3$ substrate is 1 mm and the impedance of the electrical input is a capacitance of 25 pF.

The ac electric field probe was tested by applying 50 Hz sinusoidal voltages directly to the piezoelectric substrate. As a first step, the amplitude of the unmodulated carrier wave was determined; the frequency changes versus biasing voltages up to 1.4 kV were then measured. Applied voltages corresponded to internal electric field strengths up to 1.4 MV/m, being the thickness of the

substrate equal to 1 mm. Measurements were carried out both at the fundamental frequency of the probe and at the 7th overtone. A fairly linear relationship of the frequency modulation Δf versus the applied voltage has been found for both frequencies, as it is shown in Fig. 3. As it can be expected, sensivities at the two frequencies are in the ratio 1:7, suggesting the use of piezoelectric substrates oscillating at higher frequencies to enhance sensitivity.

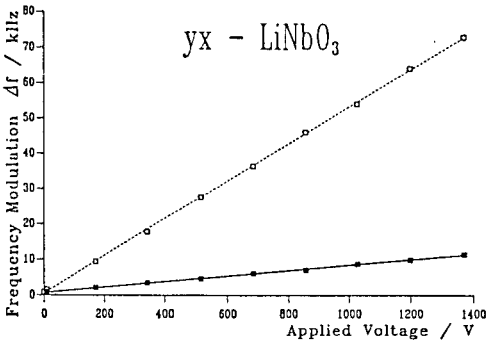

Fig.3 - Frequency modulation versus biasing 50 Hz voltages for the oscillating frequencies of (■)106.45 MHz and (□)745.15 MHz

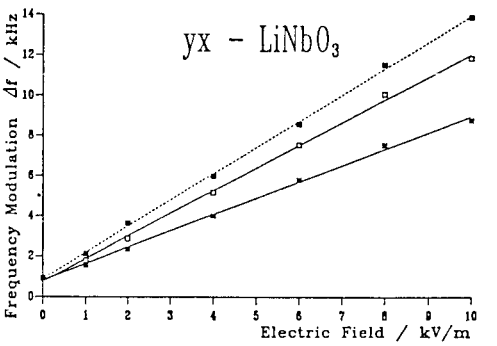

Fig.4 - Response of the meter for the different coupling structures (*S=100 cm², H=17.4 cm; ■S=100 cm², H=20.7 cm; □ S=219 cm², H=16.9 cm)

To usefully employ the piezoelectric sensor for measuring 50 Hz electric field strengths under transmission lines, a proper coupling structure must be provided. Following the analysis of Deno and Zaffanella (6), a simple conducting square plate was chosen and electrically connected to the measuring electrodes in such a way as to have the highest capacity to ground. Three plates were tested, differing for the values of the surface, S, and the height, H, with respect to the piezoelectric substrate. For the calibration of this ground-reference-type field meter, a quasi-uniform electric field of known field strength was produced by a two parallel plates system (7). In Fig.4 the frequency modulation Δf versus the external electric field strength for the three experimental arrangements are shown. Experimental results are shown up to an electric field strength of 10 kV/m, but the linear region extends up to 50 kV/m, corresponding to a frequency modulation Δf = 66 kHz.

CONCLUSIONS

The time delay of SAWs propagating in a piezoelectric substrate is affected by an externally applied extremely low frequency (ELF) electric field. The principal source of everyday exposure to ELF fields, however, is the ac electric power system which operates at 50 or 60 Hz, depending upon the country.

The signal at the output of the SAW oscillator is frequency modulated by the applied field, whose amplitude can be simply recovered by monitoring the modulation index of the signal.

The evaluation of the practical accuracy of outdoor measurements is in progress. The greatest possible sources of errors are difficulty in positioning the meter, reading errors, handle leakage in some cases, temperature effects, observer proximity effects and difficulty in defining the geometry of the boundary conditions, leading to the establishing of the true ground. Mechanical balance of the meter movement can also be a source of error. If it is not sufficiently well-balanced, the meter should be used in the same orientation with respect to the vertical as existed during the calibration procedure. An estimate of the magnitude of this type of error can be made by rotating the meter in the absence of an electric field and by observing the corresponding changes of readings.

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AN AUTOMATED DOSIMETRY SYSTEM FOR MICROWAVE AND THERMAL EXPOSURE OF BIOLOGICAL SAMPLES *IN VITRO*

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INTRODUCTION

Dielectric heating is a recognized mechanism for the induction of radiofrequency (RF) bioeffects considered to be thermal in origin - that is attributable to temperature increases. However, apart from RF-heat cell killing (e.g. Sapareto et al.,1982, Chang et al.,1987) very little information is available on quantitative relationships between RF bioeffects at the cellular and molecular level and temperature profiles over time or thermal dosage (TD). The determination of RF heating/cooling curves can be used to compute the specific absorption rate (SAR) and the specific absorption (SA) in an exposed sample (Stuchly and Stuchly, 1986). Biological variables and exposure conditions can be controlled in experiments with *in vitro* systems to a degree not achievable *in vivo*. An attractive model to study RF bioeffects is the transformation of lymphocytes *in vitro*, provided the biological variables can be related to dosimetric quantities that characterize exposure (Czerski,1975; Budd and Czerski,1985). To accomplish this, we designed an exposure system with provision for real-time temperature monitoring with RF-field non-perturbing temperature probes. The exposure system, which will be described in detail later, has multiple sample chambers. To allow on-line thermometry and dosimetry, one of these chambers is used as a site for a non-perturbing temperature probe. From the temperature (T)/time(t) history of the sample chamber, an exposure dosage can be determined and described in terms of TD, SAR, SA, or electric field strength. The system can be used for studies of RF-bioeffects in any tissue culture cell line or other *in vitro* biological sample. The monitoring component can be applied to the study of temperature-dependent effects, irrespective of the modality used for heating.

EXPOSURE SYSTEM

A schematic diagram of the system for microwave exposure of samples is shown in Fig. 1.

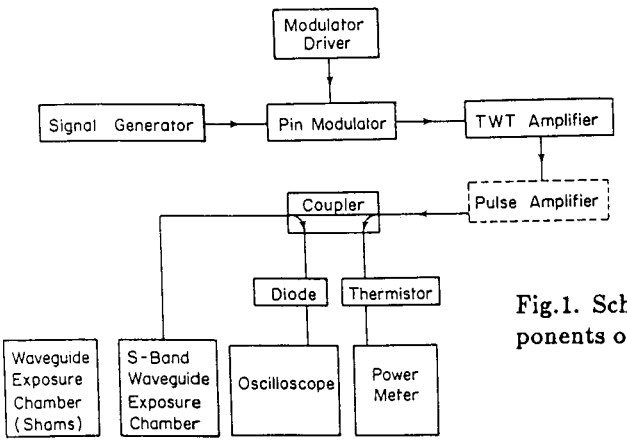

Fig.1. Schematic diagram of essential components of microwave exposure system.

The output from a CW Hewlett-Packard Model 8616A oscillator feeds a Hughes Model 1177H TWT amplifier. Pulsed or amplitude-modulated waveforms are obtained through the use of a Hewlett-Packard Model 8403A p-i-n modulator and driver. For high peak power, but low duty-cycle, pulsed exposures the signal can be further amplified with a 1kW amplifier (MCL, La Grange, Ill.) The amplified signal enters a shorted section of S-band rectangular waveguide 300 mm long through a matched coaxial feed. The waveguide sample holder receives its power through an isolator (not shown specifically in Fig. 1) and a dual-directional coupler that allows forward and reflected powers to be measured with Hewlett-Packard Model 432A power meters, and waveforms to be monitored. A sample holder for biological specimens is supported centrally in the waveguide in a block of low-density polystyrene that fills the cross-section of the guide. An identical waveguide section with an identical plastic sample holder is used for sham exposures. During microwave exposures, both waveguide assemblies were housed in a CO_2 tissue culture incubator thermostated at 37.0°C. Various sample holders were tested: a 4-chamber plastic tissue culture slide (Miles Scientific # 4804) was found satisfactory in our application. Each chamber holds 1 ml. A miniature thermistor temperature probe enters each waveguide section through a small hole in the top shorting plate. The hole is drilled near the guide wall, at the center of the shorter dimension – this is a low electric field point and the hole produces minimal disturbance. The active end of the thermistor probe enters the sample in one of the 4 chambers of the tissue culture dish. We could move the probe from chamber to chamber to check exposure uniformity. Two types of non-field perturbing probe have been used in this way: A Narda Model 8011B non-perturbing double temperature probe or two Vitek Model 101 probes. The temperatures in both the exposed and sham-exposed samples are recorded continuously during an experimental run. The temperature probes are connected to a Hewlett-Packard relay activator, Model 59306A, which is itself connected to a Keithley Model 192 DVM. Both the relay activator and DVM are under the control of a Hewlett-Packard Model 86 desk-top computer through the HPIB (IEEE-488 bus)(Fig.2).

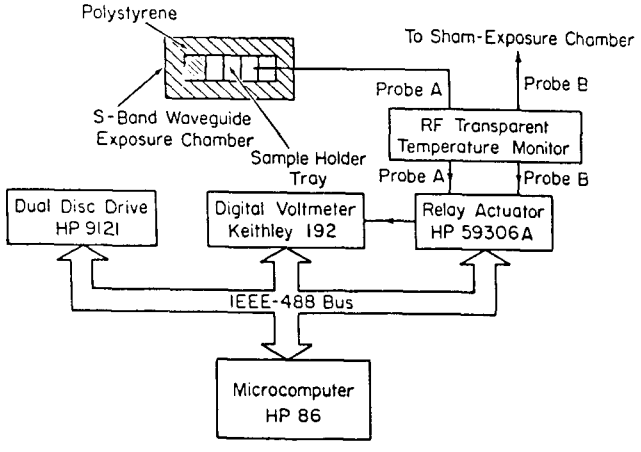

Fig.2. Schematic diagram of multiple-chamber sample holder with non-perturbing temperature measurement and automated dosimetry system.

Under computer control, the temperatures of exposed and sham-exposed samples are recorded sequentially and stored in memory. Since the temperature of sham-exposed samples is essentially constant, it is monitored less frequently than the temperature of the exposed samples. A typical experimental protocol would involve 10 temperature readings of the exposed sample for every one reading of the sham-exposed. For conventional thermal exposures, the temperature can be elevated by increasing the temperature of the incubator.

TEMPERATURE/TIME ANALYSIS

During a typical experiment, the temperature is recorded at regular intervals, with a minimum measurement interval $< 1s$, before the beginning of, during, and after the exposure. The "on" and "off" times of exposure are recorded on the computer by the operator using a "soft-key" interrupt capability. At the conclusion of a run, the T(t) behavior is analyzed to determine SAR.

If the SAR is S ($\mathrm{Wkg^{-1}}$) and the specific heat of the sample is $C(\mathrm{Jkg^{-1}K^{-1}})$, then the SAR can be determined from the change in heating rate when microwave power is applied.

$$\left(\frac{dT}{dt}\right)_S - \left(\frac{dT}{dt}\right)_{S=0} = \frac{S}{C}$$

In practice, this analysis is performed automatically. At the end of a run, the point of inflection, or turning point, in T(t) is found numerically. The temperature profile to the left and right of this point is fitted by least-squares to a linear or quadratic function and the change in slope at the beginning of the exposure period yields the SAR. Figures 3 and 4 are examples of such a procedure.

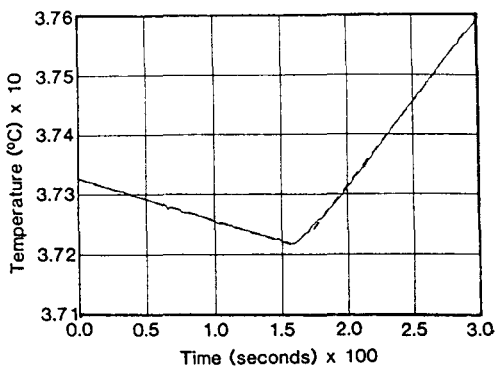

Fig.3. Temperature/time history of an exposed sample showing linear fits to cooling and heating portions of curve for dosimetry.

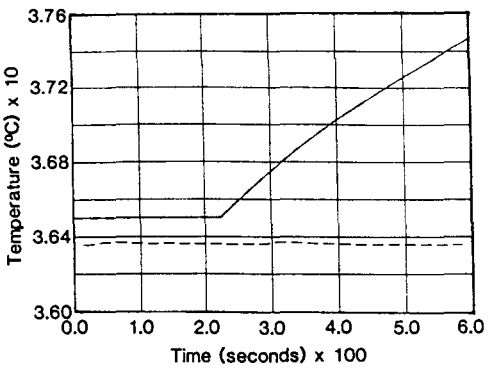

Fig.4. Temperature/time history of exposed (upper curve) and sham-exposed samples (lower curve). The dosimetry for the exposed sample has been determined from a quadratic fit to the heating curve during microwave exposure and a linear fit to the equilibrated portion of the curve prior to exposure.

Figure 3 shows the temperature/time history of a sample that was cooling prior to the start of microwave exposure. The linear fits to the cooling and heating portions of the curve yield the SAR. Figure 4. shows the heating of a sampled that was quite well equilibrated before exposure. Because the signal/noise ratio is high the SAR was determined by a quadratic fit to the heating portion of the curve. The quality of the fit near the turning point can always be examined to determine whether the slope value is realistic. However,unless the temperature/time profile has very low noise, it is generally better to use the linear fit to T(t) near the turning point or a biased estimate of the slope can result. Figure 4 also shows the temperature/time history of the sham-exposed sample to illustrate its temperature stability.

RESULTS AND DISCUSSION

The advantages of the procedure described above are severalfold. The sample need not be equilibrated before SAR is determined. Repeated determination of SAR can easily and quickly be made, which allows the SAR uniformity from one sample chamber to another to be determined. Exact knowledge of the microwave power is not required: reproducible exposures can be made at known SAR by using the measured forward power corresponding to a given SAR measurement, provided the experimental arrangement is not altered between exposures. Dosimetry is not affected by other losses in the system. The waveguide exposure system with four sample chambers was found to give an SAR uniformity from one chamber to the other within 10%. Thus, experimental samples can be exposed and examined in triplicate, the fourth chamber being used for the insertion of the temperature probe.

The system is biocompatible and, depending on cell line, cell density, and medium, allows continuous exposures of several days duration. The biocompatibility of this arrangement was proven by studying the growth of human lymphocytes under various conditions within the waveguide. As well as providing biocompatible exposure conditions, the system is flexible: it allows the exposure of tissue culture cells growing in suspension or in monolayers, and can easily be used with cell-free samples. The exposure system is relatively simple, and can be assembled from off-the-shelf components. Its capabilities can be expanded by introducing computer control of RF power input based on feedback from temperature measurements.

ACKNOWLEDGEMENTS

This research was supported by the Office of Naval Research through Contract No.N00014-86-K-0716 with the University of Maryland. K.H. Joyner is grateful for the award of a World Health Organization Fellowship and acknowledges approval by Director, Research, Telecom Australia to present this paper.

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METHODOLOGY OF MEASUREMENTS AND EVALUATIONS OF ELECTROMAGNETIC FIELDS AROUND RADIOTRANSMITTER DEVICES

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Abstract

The radiotransmitter devices are densely widespread in all the national territory, and their increasing use poses serious problems of exposure evaluation, both for the professional exposed people, and for the public. This problem is of some concern, owing to the many interrelated factors, which must be taken into account, during the measurements. In this work the authors show a procedure to evaluate the electromagnetic fields, generated by radiotransmission antennas in the frequency range of medium and short waves. The developed methodology represents a result of a great number of measurements and of many monitoring programs, which have included a wide number of different measuring points around the sources, different environmental conditions and various measurement systems. Moreover the measurements have been made in many different hours of the day, because the transmission systems (frequency and power transmission) are many and different during the day.

Source Characteristics

The monitored sources are the radiotransmitter devices in AM of the National broadcasting network (RAI), placed in Rome, and their characteristics are:

- Short Waves Centre

In this centre there are 45 antennas with 60 transmission systems. The radiation frequency of all the transmission plants (5) changes during the 24 hours. Its value goes from a minimum of 6 MHz up to a maximum of 21.69 MHz.

- Medium Waves Centre

The plant is constitued by two antennas with the following characteristics:

1) f = 846 kHz, P_{out} = 540 kW
   This is a two dipoles antenna in $\lambda/2$, and every dipole is fed by a 240 kW peak power. The transmission pattern is perpendicular to the dipole connection line. The system gain gives a 1700 kW power in the maximum propagation situation.
2) f = 1332 kHz, P_{out} = 300 kW
   This is a three dipoles antenna in $\lambda/2$. The dipoles are fed

with currents reciprocally 120° out of phase. The system gain gives a 1248 kW radiating power in the maximum propagation situation.

Measurement Methodology

Owing to the wavelengths of interest in the radiotransmitter range of medium and short waves, the monitoring surveillance program must involve, for the evaluation of the potential hazardous agent, the measurement of the electric and magnetic field strenghts in the near field region, and of the power density in the far field region. In fact in both the regions there are exposed people. By source characteristic examination, the most significant points, from a protection point of view, have been located. These points have been selected in relation to:

i) effective workplace of operators;
ii) theoretic evaluation of exposure level;
iii) possibility to find high field strengths level, owing to the presence of interference on the radiation pattern.

In addition selected points occupied by the general public and in the nearby vicinity of the radiotransmitter plants have been monitored, for the purpose to evaluate the public exposure. The measurements have been made using different broadband systems, able to detect and evaluate the field strengths in the frequency range of interest. All the data have been carried out taking into account the worst exposure conditions, and have been detected in different hours of the day, in respect of the different transmission systems. The set up methodology proceeds as follows:

1) the measurement instruments should be placed on a non conducting tripod;
2) the measurements should be made keeping a distance of at least 50 cm from the instrument;
3) the measurements should be made starting from the maximum safety distance and gradually diminishing it, with the purpose to avoid uncontrolled surveillance people exposure and damages to the probe;
4) the exposure level mean values should be determined in the previously established points (see i), ii), iii)), at a distance from the ground of about 1 m and 1.50 m (target organs) and at different times in the day;
5) the measurement points should be selected as follows:
   - at a distance of about 1 m from any possible fixed and/or mobile obstacle;
   - at very close proximity to re-irradiating points (e.g. telephone wires, electrical conducters, etc.) for the purpose of detecting "hot spots".

It is very important to select monitoring points, so that further measurements may be undertaken if and when required.

Table 1 shows an example of detected values for every selected measurement point. The measured values are after reported on the graph, and Graph 1 is an example of this one, and shows the electric field strength at the different hours of the day.

Conclusions

On the basis of the great amount of evaluations - more than 6,000 - the following recommendations may be utilized:

1) make the measurements at least along three directions, in order to detect the eventual preferential irradiating stream, even if isotropic probes are used;
2) correct the detected values due to:

- different measurement instruments;
- signal characteristics;
- exposure conditions.

In addition, in order to protect the exposed people against the possible presence of "hot spots" and to avoid high level of field strengths in the nearby of great conductor body, it is necessary:

1) earth the metal conductor, telephone wires, radiators, etc.;
2) earth water and gas mains, and waste pipes.

T A B L E 1

ELECTRIC FIELD STRENGHTS (V/m)

hour	day: n			day: n + 1			day: n + 2			mean
	$\bar{n}_1$	$\bar{n}_2$	$\bar{n}_3$	$\bar{n}_4$	$\bar{n}_5$	$\bar{n}_6$	$\bar{n}_7$	$\bar{n}_8$	$\bar{n}_9$	$\sum \bar{n}_i / N$
a.m.										
00.00	105	98	90	100	100	132	118	100	100	104.7
00.30	100	98	90	110	94	100	105	120	100	101.8
01.00	:::	:::	:::	:::	:::	:::	:::	:::	:::	:::::
:::::	:::	:::	:::	:::	:::	:::	:::	:::	:::	:::::
:::::	:::	:::	:::	:::	:::	:::	:::	:::	:::	:::::
04.00	110	115	120	100	116	110	105	100	120	110.6
05.00	62	64	68	69	54	54	54	69	66	62.0
:::::	:::	:::	:::	:::	:::	:::	:::	:::	:::	:::::
10.30	60	55	67	61	59	65	65	58	60	60.0
11.30	61	59	67	63	55	64	58	61	61	61.0
12.00	100	98	92	86	89	96	98	105	115	97.6
p.m.										
12.30	:::	:::	:::	:::	:::	:::	:::	:::	:::	:::::
:::::	:::	:::	:::	:::	:::	:::	:::	:::	:::	:::::
04.00	60	57	55	61	59	65	65	60	58	60.0
04.30	97	96	84	90	86	91	93	98	109	93,7
:::::	:::	:::	:::	:::	:::	:::	:::	:::	::::	:::::
09.00	100	98	92	86	89	96	98	105	115	97.6
:::::	:::	:::	:::	:::	:::	:::	:::	:::	::::	:::::
11.30	110	116	100	120	109	110	115	105	110	105.0

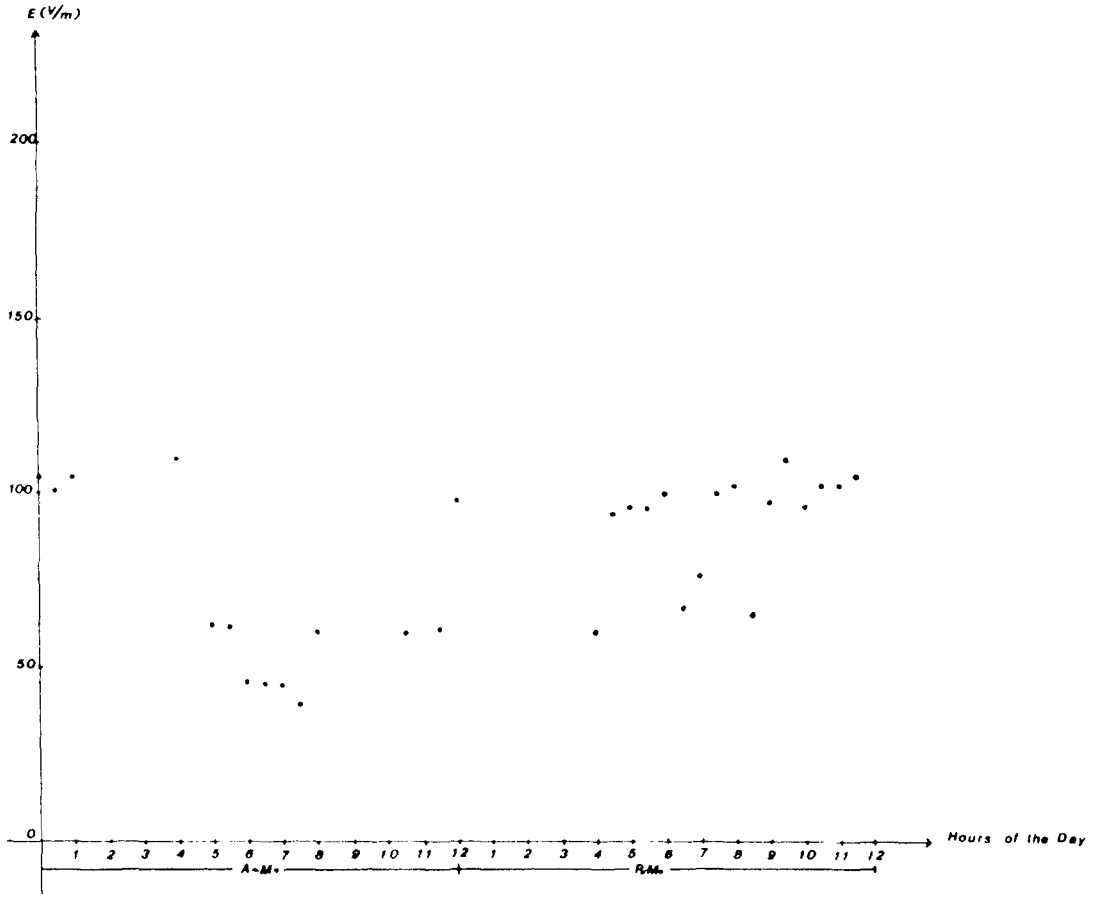
N.B. n_i = number of three series of measured values in the selected points:
(1, 2, 3) the day n;
(4, 5, 6) the day n + 1;
(7, 8, 9) the day n + 2;

$\bar{n}_i$ = each values represents the mean of other three measurements made reciprocally at 90° out of phase;

$\frac{\sum \bar{n}_i}{N}$ = the mean values relative to three measurements in three different days, but at the same hour of the day.

TABLE 1: Recording Chart of Detected Values for Every Selected Points

G R A P H 1



GRAPH 1: Electric Field Strength at the Different Hours of the Day

MEASUREMENT PROCEDURE FOR ELECTROMAGNETIC FIELDS OF RADAR SYSTEMS, USING A SPECTRUM ANALYZER APPARATUS

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Abstract

The radiofrequency (RF) and microwave (MW) emitting devices are used in many human activities and their application is continuously increasing. For this reason, the problem of evaluating their potential hazards to which people are generally exposed is of some concern, both for professional people, and for the general public. In the present report, the authors describe a measurement procedure, developed for the evaluation of electromagnetic fields produced by and around radar systems. By means of the developed measurement method the density power has been evaluated, using a spectrum recorder analyzer with the following extra-components: standard receiving antenna, attenuation nets, plotter.
The measurements have been carried out around many radar systems and the results have been compared with those obtained with other measurement systems and methods.

Measurement Method around Radar Systems

By means of the spectrum analyzer it is possible to visualize on the cathode ray tube the emitted electromagnetic signals, decomposed in their main components; the measurements provide the values of absolute or relative level (in dB). If the input signal is modulated, like the radar signal, the power level measured is not an absolute one, because the input electromagnetic energy is not a monochromatic and can not be thorougly detected by the spectrum analyzer. However the numerical value of input peak power reduction factor is given on the basis of the selected attenuation circuits and therefore it is possible to calculate the exact value of input power. Furthermore the use of spectrum analyzer makes measurement possible without stopping the emitting antenna in the direction of the measurement instrument. These measurement conditions represent the real operation conditions and make any correction unnecessary for the rotating factor or radiation exposure. This measurement procedure uses a spectrum analyzer mod. HP 853A/HP8559A, with the standard receiving antennas Singer mod. 93490-1 and 93491-2; the frequency range work is 500 MHz - 3 GHz, for input power values until 500 kW, if the set is arranged with suitable attenuation nets.

The measurement method proceeds as follows:

i) a calculation of the power density expected around the source and based on the characteristics of the emitting device;

ii) an investigation of the environment around the source;
iii) a choice of suitable measurement set, in relation to emitting device characteristics and the previous results;
iv) a monitoring program and the subsequent data recording on the plotter;
v) data evaluation

Before starting the monitoring program, all available information about the emitting source characteristics should be obtained, and at least the following: system operating frequency; peak/medium power; pulse width (PW), if any; pulse repetition frequency (PRF), if any; antenna gain; radiating antenna dimensions; beam width (BW); scan or rotation rate, if any; distance between the radiating antenna and the ground.

Estimated Value

To select the best measurement set, before starting the monitoring program, it is necessary to calculate the expected power density values in the measuring points. These values must be evaluated at a given distance from the source by means of:
- transmission power;
- radiating antenna gain;
- radar beam amplitude.

The estimated power density value represents the maximum power density at the centre of the main antenna lobe, in the given measurement point. This value is generally greater than the measured value, because the monitoring points are located, almost always, under the centre of the main lobe.

Environmental Investigation

All the available information about the geometrical and geographical characteristics of the site around the radar system should be obtained in order to ascertain if the radiation meets along its pattern metallic conducting structures or reflecting surfaces (for example mountains), and so on. These obstacles in fact may produce secondary fields whose intensity could add to that of main electromagnetic fields, leading to points of the site, where the field intensity could be very high. This investigation is important for personnel protection purpose.

Measurement Equipment Selection

The main items needed for selecting the measurement array are:
- distance between the survey points and the source must be greater than ten meters (far field conditions);
- radiating lobe corner at least 3 degrees above the ground;
- rotating antenna.

The evaluation of the power density allows, at the first stage, to estimate the expected power density values in the survey points, and moreover to select the suitable receiving antennas and the spectrum analyzer attenuation arrays. With the above mentioned receiving standard antenna it is possible to detect density for estimated values less than 831 mW/cm^2. For power density values greater than above this value, in the same frequency range, the spectrum analyzer has to be arranged with suitable attenuation arrays and located at a safety distance from the radiation source. The values measured

with this method have been compared with the data gathered using other different measurement devices, like power meter and broadband instrumentation (NARDA 8616 system). There is a very important difference between these methodologies. A spectrum analyzer provides measurements in real working condition of the radiating system (rotanting antenna); in the other cases the antenna must be stopped and the measured values are not those really present in the working condition, and represent the maximum value of power density. This is a real working condition only in very few cases (for example during radiating system maintenance program). The power meter is equipped with bolometric interchangeable devices, which, in relation to the frequency and input power range, allow the detection and the measurement of power density values up to 200 kW/cm^2.

Monitoring Program

As the radiating system is rotating during the monitoring program, the spectrum analyzer monitor shows the maximum electromagnetic field level, when the radiating antenna is oriented towards the operator. This maximum value is recorded (on the plotter), and other values are detected in all selected measurement points. These data represent the relative power level in dBm, while the absolute values of density power are obtained through suitable conversion tables, taking into account the following parameters:
- antenna factor;
- desensitation factors;
- pulse repetition frequency (PRF);
- pulse width;
- attenuation factors of the nets.

The methodology has been verified on a great number of radar systems and have been compared with the data obtained, using other methods. The results have always been compatible.
In Table 1 results obtained around radar systems, using different measurement methodologies are shown. The same values are compared with the theoretic expected values in the same monitoring points.
In Table 2 the results obtained on laboratory radar system are shown. In this case the different measurement conditions and the secondary lobe emissions have also been evaluated; the measurements have been carried out with and without ecosorb surface, in order to take into account the real operation conditions in the effective workplaces. In this case power density values have not been estimated, because they are relative to the emission, due to the main antenna lobe, while measurements have been carried out in relation to the secondary emission lobes.

Conclusions

The obtained values during various monitoring programs around many and different radar systems confirm the validity of the the methodology, also because it allows to carry out the measurements in the real operating system condition, which are the real exposure conditions for the operators.

Table 1 - Fields Measured around Radar Systems

	Plant Characteristics	Measurement Points: H_t Above the Ground (m)	Measurement Points: Distance from the Plant (m)	Estimated Power Density on the Antenna Axis (mW/cm^2)	Measured Power Density (*) (mW/cm^2): NARDA 8616 (Probe 8623b)	Measured Power Density (*) (mW/cm^2): POWER METER (HP432-HP478 HORN SINGER 93491-2)	Measured Power Density (*) (mW/cm^2): SPECTRUM ANALYZER (HP853/A-HP8559/A HORN SINGER 93491-2 PLOTTER HP7470/A)
1	P_p = 500 KW	1	20	0,06	0 (f)	0,15 (f)	0,186 (f) 83 X 10^{-6} (r)
	F = 1,3 GHz	4	35	0,02	3 (f)	2 (f)	5,88 (f) 2,94 X 10^{-3} (r)
2	P_p = 1,8 MW	39,6	2725	28,6 X 10^{-6}	0 (f)	2,34 X 10^{-3} (f)	1,8 X 10^{-3} (f) 0,452 X 10^{-6} (r)
	F = 1,3GHz	2	50	0,085	0 (r)	=	13,2 X 10^{-3} (f) 3,3 X 10^{-6} (r)

* r = rotating antenna
f = standing antenna towards the measurement instrument.
All the values have been obtained taking into account the suitable correction factors.

Table 2 - Laboratory Radar System

PLANT CHARACTERISTICS: P_p = 180 ÷ 220 KW F = 9,4 GHz

Measurement Point: H_t Above the Ground (m)	Measurement Point: Distance from the Plant (m)	Measured Power Density (*) without Ecosorb Surface (mW/cm^2): NARDA 8616 (Probe 8623 b)	Measured Power Density (*) without Ecosorb Surface (mW/cm^2): POWER METER (HP432-HP487 HORN-SINGER 93491-2)	Measured Power Density (*) without Ecosorb Surface (mW/cm^2): SPECTRUM ANALYZER (HP853/A-HP8559/A HORN SINGER 93491-2 PLOTTER HP7470/A)	Measured Power Density (*) with Ecosorb Surface (mW/cm^2): NARDA 8616 (Probe 8623 b)	Measured Power Density (*) with Ecosorb Surface (mW/cm^2): POWER METER (HP432-HP478 HORN SINGER 93491-2)	Measured Power Density (*) with Ecosorb Surface (mW/cm^2): SPECTRUM ANALYZER (HP853/A-HP8559/A HORN SINGER 93491-2 PLOTTER HP7470/A)
0,90	0,50	185X10^{-3}	100X10^{-3}	74X10^{-3}	555X10^{-3}	150X10^{-3}	74X10^{-3}
1,70	0,50	423X10^{-3}	225X10^{-3}	93X10^{-3}	925X10^{-3}	280X10^{-3}	93X10^{-3}
0,90	2	116X10^{-3}	29X10^{-3}	23X10^{-3}	231X10^{-3}	125X10^{-3}	29X10^{-3}
1,70	2	185X10^{-3}	40X10^{-3}	29X10^{-3}	462X10^{-3}	125X10^{-3}	145X10^{-3}
0,90	3	N.R.	6,2X10^{-3}	930X10^{-6}	N.R.	7,9X10^{-3}	930X10^{-6}
1,70	3	N.R.	N.R.	2,9X10^{-3}	46X10^{-3}	9,9X10^{-3}	4,7X10^{-3}

* All the values have been obtained taking into account the suitable correction factors.

THE EFFECTS OF ULTRAVIOLET RADIATION ON SOME PLASTIC DETECTORS

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INTRODUCTION

Exposure to ultraviolet (U.V.) radiation could lead to the development of phototoxicity, photoallergy and enhancement of photocarcinogenesis (IRPA 1979). For the measurement of the dose of ultraviolet radiation harmful to human bodies, it is desirable to use a detector with a response curve similar to the human action spectra for erythema and for ultraviolet radiation (Diffey 1982). It is not easy to obtain a detector which satisfies the requirement unless a very complicated setup of spectrometer with suitable photodetectors is employed (Wong and Fleming 1984). For the purpose of measuring the personal dose, a passive type of detector is preferred. Davis et al (1976) have developed a detector using polysulphone for this purpose. The response curve shows a broad peak extending from 260 to 325 nm. Other detectors (Spectrosonics, Partridge and Barton 1978, Fanslow et al 1983), which were developed for similar purposes, have a sharp peak around 280 to 320 nm. These response curves are not very good approximations to the human action spectrum which has a sharp cut-off at around 300 nm. They tend to overestimate the contribution on the long wavelength region (300-320 nm) of UV-B. The integrated response in the UV-B region could be twice the total area of the human action spectrum in the same region of wavelength. The irradiance of these long wavelengths is at least ten times that of the short wavelengths (less than 300 nm) in the solar spectrum. A search for materials with a better approximation than the three types of detectors mentioned above would be useful in the development of a more accurate dosimeter. In this paper, we will report our preliminary results on a new type of plastic material.

EXPERIMENTAL

It was found that ultraviolet exposure can change the etching properties in some plastic materials (Wong et al 1982). In this report we will present our result on allyl diglycol carbonate. The detectors were prepared in discs of about 1.0 cm in diameter and about 0.6 mm in thickness. Accelerated ultraviolet radiation effects were induced in the detectors using a 200 W xenon-mercury lamp source. The detectors were mounted on a circular holder placed at a distance at least 16 cm from the source in order to avoid excessive thermal loading. Radiation from the source was filtered through by narrow-band filters which cover a range of wavelengths from 250 to 340 nm. Each filter has the

bandwidth (i.e. the full width at half maxima) of about 10 nm. The peak transmittance of the filters is not less than 15%. The leakage through the filter at wavelengths other than the transmitted band was checked with a spectrophotometer. It was found that the leakage was negligible. A pyroelectric radiometer was used to measured the irradiance. The distance between the filters and the source was adjusted such that the irradiance through the filters was approximately the same initially. After the exposure, the irradiance through the filters was measured again with the radiometer. It was found that the irradiance through each filter did not change more than 5%.

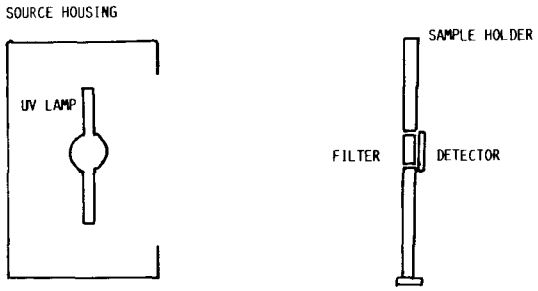

Figure 1. Experimental Arrangement for Irradiation

The exposure took place for at least 48 hours continuously. After exposure, the detectors were etched in strong alkaline solutions to enhance the effect of damage by ultraviolet radiation. An optimum etching time of about 180 minutes was used in the present experiment. The opacity through the detector was measured using an optical microscope and a spectrophotometer. At least ten measurements at different areas on each detector were averaged to determine the average response.

RESULT

After chemical etching, the detectors which had been exposed to short wavelengths ranging from 250 to 290 nm exhibited a certain degree of opacity in the irradiated area. The detectors which had been irradiated with wavelengths longer than 290 nm showed no, or insignificantly small, change as compared with those exposed to shorter wavelengths. A photographic record of these detectors is shown in Figure 2. The experiments were repeated several times to confirm the finding. It was noted that the effect may be amplified using suitable experimental conditions. In order to quantify the effects, the change in optical density has been determined for these detectors in the

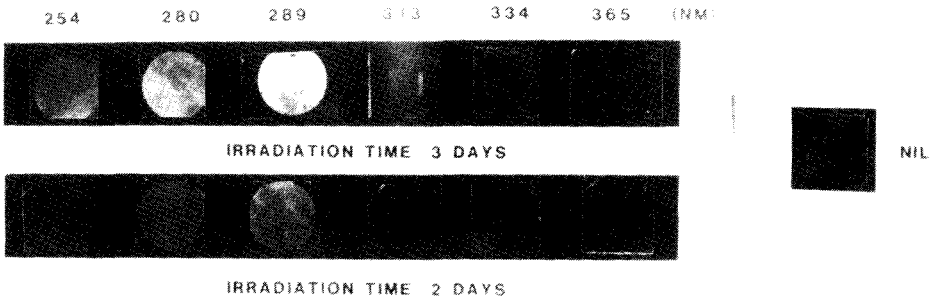

Figure 2. Induced Change by U.V. in Carbonates

manner as described previously. The aggregated results of several experiments with normalisation (Wong and Fleming 1984) of integrated irradiance and filter-transmittance is presented in Figure 3. The uncertainty in the relative response is due to statistical errors. The band width for each filter is also included in the diagram. The response curve shows a sharp cut-off at around 290 nm, while the response is almost a constant in the region from 280 to 290 nm. A comparison of the present result with the human action spectrum for erythema is shown in Figure 4. The line is given by the experimental points in Figure 3. A good agreement was obtained between the two curves. Also included in the diagram is the response curve of polysulphone (Davis, 1976). In view of these results, the detector could be used as a dosimeter for measurements of ultraviolet radiation. Further experiments on the response of the detector to ultraviolet radiation in the solar spectrum are being conducted.

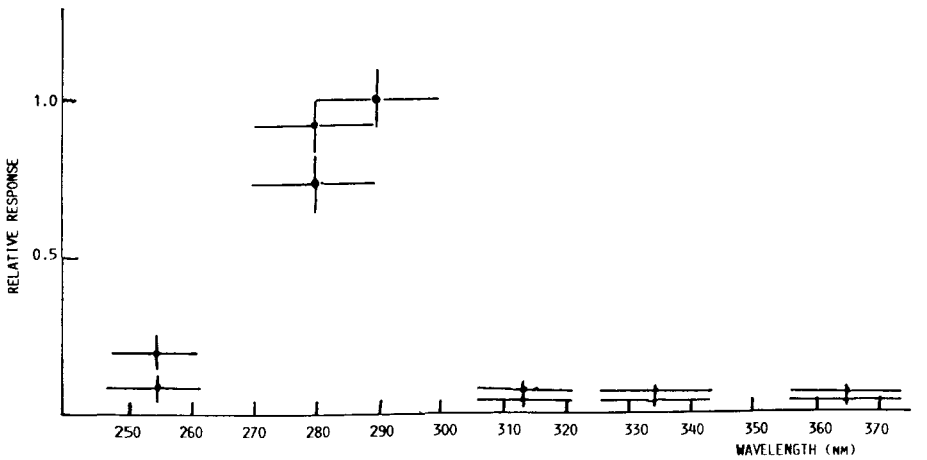

Figure 3. U.V. Response Curve of Carbonates

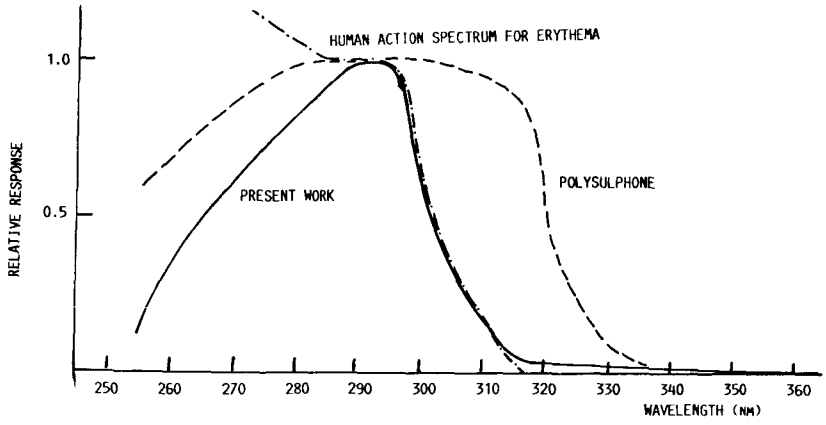

Figure 4. A Comparison of Carbonates and polysulphone

DISCUSSION AND CONCLUSION

The response curve for ultraviolet radiation has been determined in the region between 250 nm and 370 nm for a new type of plastic material. Although no measurement is available in the region between 290 nm and 313 nm, due to the limitation of the experimental set-up, the trend of the results obviously indicates negligible effects in the region between 313 nm and 370 nm. From previous experiments on the same material, it was found that the plastic has sufficient sensitivity for solar spectrum measurements (Wong and Hoberg 1982). In view of these results, it is not unreasonable to suggest that the relative response of the material decreases in a manner as shown in Figure 2. Measurements of the absorbance on the same material support this conclusion. In view of these results, it is suggested that the plastic could be useful for measurement of ultraviolet radiation harmful to human bodies. Further studies are planned to develop the material for personal UV monitors.

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CONTRIBUTION DE LA RADIOECOLOGIE A LA RADIOPROTECTION BILAN DE 30 ANS DE RECHERCHE EVOLUTION DES PROGRAMMES NOTAMMENT DANS LE DOMAINE DES ACCIDENTS

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La Radioécologie a trente ans. Trente ans de recherche au service de la radioprotection. Suivant les pays, les contributions sont d'une importance variable. De tous les continents l'Europe a le plus fourni du fait de nombreuses installations nucléaires dans un tissu urbain, agricole et industriel dense. Les priorités de recherches en radioécologie ont été fonction des programmes industriels nationaux. Parmi ces priorités l'étude de l'impact des accidents nucléaires a été il y a trente ans (avec l'incident de Windscale en 1958) un des éléments moteurs du développement de la radioécologie. Le bilan des recherches, malgré le nombre imposant de publications produites, invite à la modestie.

L'analyse des facteurs de transfert publiés et utilisés pour l'évaluation des doses, révèle qu'un certain nombre de valeurs utilisées ne sont pas suffisamment explicites et sont difficilement exploitables. Aujourd'hui, avec le retour d'expériences provenant de l'exploitation des installations industrielles nucléaires on doit reconnaître que certaines valeurs sont trop conservatrices du fait des conditions expérimentales. L'analyse des facteurs de transfert montre que leur emploi dans les modèles d'évaluation du risque n'est pas toujours facile quand on veut réduire les marges d'incertitude. La sélection des paramètres et des valeurs fixés pour ces paramètres exige une révision quelquefois déchirante. Ainsi la mise au point d'un nouveau manuel de radioécologie (Regulatory Guide) à usage des responsables sécurité des installations du cycle du combustible nucléaire nous a amené à qualifier plusieurs milliers de facteurs de transferts ou de concentration des radionucléides dans l'air, l'eau, le sol, la production agro-alimentaire. Le résultat obtenu est assez surprenant, près de 25% des travaux publiés sont inexploitables soit du fait des incertitudes sur les conditions d'obtention des résultats soit de conditions expérimentales trop éloignées des conditions naturelles ou simplement d'une estimation trop simpliste des rejets radioactifs pour le calcul d'un facteur de tranfert "in situ".

Malgré cette épuration importante, il faut reconnaître que les données fournies aux hygiénistes nucléaires par les Radioécologistes ont permis de répondre aux demandes formulées et l'établissement des dossiers règlementaires quelles que soient les législations nationales a bénéficié largement de cette contribution.

Ainsi une des réussites de la radioécologie c'est d'avoir codifié les études de sites nucléaires et de permettre ainsi d'implanter et de qualifier un site avec une approche en matière

de protection de l'environnement exceptionnelle si on la compare aux sites industriels non nucléaires.

Certains ont même pensé que la Radioécologie avait définitivement répondu aux questions soulevées par les radioprotectionnistes au point de considérer cette discipline comme arrivant au terme de son existence. Heureusement quelques scientifiques avaient attiré l'attention, il y a quelques années, sur les insuffisances de connaissance sur les transferts dans les chaînes alimentaires pour des situations particulières que l'on rencontre essentiellement dans les pays où le développement industriel nucléaire couvre l'ensemble du cycle du combustible.

On peut citer plusieurs thèmes de recherches qui méritent un effort de développement. :

- L'étude du comportement des actinides dans les sites de stockage de haute activité (sites profonds : granit, sel, argile, schiste etc...) mérite une attention particulière.

- L'examen des rejets lié au vieillissement des centrales électronucléaires conduit à des études de transfert sur les chaînes alimentaires de radionucléides dont la forme physicochimique et les facteurs de concentration ne sont pas connus.

- De même la toxicologie chimique des éléments stables associés aux rejets radioactifs posent des problèmes nouveaux.

- Enfin les études de transfert des radionucléides dans l'environnement en situations accidentelles représentent à elles seules un programme important en matière de Radioécologie.

La simulation de situations accidentelles et l'évaluation de leurs impacts dans l'environnement n'ont pas toujours suscité dans le passé de l'intérêt parmi les responsables des programmes nucléaires. Aujourd'hui on mesure davantage la nécessité de ces programmes et les premiers résultats obtenus depuis 5 ans sont utiles pour tenter de comprendre et d'utiliser les observations qui sont faites dans l'environnement proche ou lointain de Tchernobyl. De plus ils sont un argument pour témoigner de l'esprit de prévoyance et d'anticipation des autorités en matière de Radioprotection. Ils n'ont pas toujours empêché des propos maladroits et quelquefois contradictoires quant au transfert des radioéléments déposés. L'exemple bien connu du lavage des légumes qui devaient "éliminer" la plus grosse part des dépôts alors que nous savions déjà expérimentalement, et depuis 4 ans, que pour les césium , les iodes, le tellure et dans une certaine mesure les ruthénium, l'absorption foliaire de 80 % et 90 % du dépôt était effectuée en quelques heures. Aujourd'hui plus personne ne conteste l'intérêt de ces recherches qui sont en même temps confortées par les observations effectuées "in situ".

Après Tchernobyl, il est évident qu'une meilleure connaissance de la forme physico-chimique des radionucléides rejetés à haute température, serait utile pour évaluer l'impact et définir des procédures d'intervention et de récupération de l'environnement.

Des travaux ont été lancés en France depuis plusieurs années. Le programme radioécologique "RESSAC", qui vise à simuler expérimentalement les dépôts, le transfert et la récupération des sols après accident, en est un exemple. Le thème de cette étude est la mise au point de techniques de réhabilitation d'un environnement notamment des sols et des nappes après un accident nucléaire sévère qui entraîne des dépôts conséquents de radioactivité dans un rayon de 10 à 50 km autour de l'installation. A partir de la simulation expérimentale du terme source dans un bâtiment spécialement conçu pour réaliser la fusion à haute température de matériaux nucléaires à haute activité nous étudions les formes physico-chimiques des aérosols émis, le dépôt et le transfert dans les végétaux et le sol des radionucléides. Ces installations permettent sur des milieux naturels provenant de sites électronucléaires de créer des conditions climatiques variables et de mettre en oeuvre avec le maximum de réalisme différents types de contre-mesures. Ces contre-mesures peuvent aller du simple traitement du sol à la mise en oeuvre de technologies agroalimentaires qui autorisent la commercialisation du produit.

En effet le récent accident de Tchernobyl a montré que les critères d'évaluation du détriment n'étaient pas uniquement des critères sanitaires, mais que se superposaient des considérations économiques et politiques. L'environnement doit être considéré non pas seulement comme une interface entre le rejet et l'homme, mais comme le support de notre économie.

Ce domaine particulièrement important ouvre pour la Radioécologie de nouveaux espaces de recherche. Ainsi au niveau Européen des programmes expérimentaux sont lancés pour proposer des mesures spécifiques de protection de l'environnement qui prolongent les premières mesures sanitaires.

La fertilisation des sols, le blocage chimique de la migration des radionucléides, les technologies agroalimentaires sont des exemples de ces mesures de radioprotection à finalité socio-économique.

Ainsi, après 30 ans de recherches, la Radioécologie subit le même type d'évolution que la plupart des disciplines qui forment la Radioprotection. Les connaissances de base sont en grande partie acquises, mais il existe des besoins scientifiques de connaissances associées à l'évolution des programmes nucléaires industriels : le stockage à long terme, l'impact lié au vieillissement des installations, l'étude des situations accidentelles, la contribution de la radioactivité naturelle dans l'environnement humain.

Un domaine toutefois mérite une attention particulière car il a été négligé dans le passé à la suite de conclusions un peu hâtives publiées à partir d'expériences limitées, il s'agit de l'effet du rayonnement ionisant sur les écosystèmes, effet lié à la fois au rayonnement externe, au dépôt et aux radionucléides métabolisés. Sans tomber dans des études fondamentales des faibles doses, il semble que l'observation, en situation accidentelle,

de la réaction du monde végétal en particulier à ces effets d'irradiations associés devraient nous apporter des moyens d'évaluation de l'impact dans l'environnement au moyen d'une dosimétrie dérivée d'une lecture visuelle des écosystèmes.

Ce dernier exemple choisi parmi d'autres confirme qu'en effort important reste à accomplir et que la Radioécologie doit maintenir et mobiliser ses moyens pour apporter à la radioprotection les éléments indispensables à l'évaluation exacte du risque.

A PLEA FOR CONSISTENT CRITERIA FOR INTERPRETING RADIOLOGICAL ENVIRONMENTAL DATA

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ABSTRACT

Current reductions in "allowable" levels of radiation and radioactive materials and an increased public awareness of naturally occurring radioactive materials have reinforced the need for consistency in evaluating the radiological environment. A key problem is in interpreting environmental levels resulting from nuclear facility operations. If these levels can be detected and their ultimate source(s) can be identified, then corrective actions can be taken to eliminate or greatly reduce the environmental impacts of such sources. In this paper we address the lack of definitive guidance necessary to determine incremental levels of significance or of insignificance (i.e., de minimis levels or those "below regulatory concern").

This paper proposes criteria for more consistent implementation of sampling and statistical procedures used to evaluate environmental conditions. Some of the more important issues needing definitive guidance include:

- determining "background" levels by consistent methods
- assuring representativeness of samples
- instituting uniform sampling procedures
- evaluating environmental data on a "population" (statistical) rather than on a single sample/measurement basis
- establishing internationally accepted criteria for determining at what level of contamination remedial action is required.

Work supported in part by the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

LE CONCEPT DE RETOUR D'EXPERIENCE EN RADIOECOLOGIE APPLICATION AU CAS D'UN FLEUVE A FORTE IMPLANTATION NUCLEAIRE : LE RHONE

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Depuis 30 ans, l'équipement nucléaire du Rhône se poursuit : la première centrale date de 1956 et depuis ont été implantés 17 réacteurs appartenant aux filières UNGG, REP et RNR (15000 MWe), des usines d'enrichissement et de retraitement du combustible. Ces installations rejettent des effluents liquides dont la radioactivité s'ajoute à celle des retombées des explosions nucléaires et plus récemment à celle résultant de l'accident de Tchernobyl. Le Rhône est ainsi un lieu privilégié d'observations sur les processus de fixation et de transfert des radionucléides.

1. LE RHONE

Long de 812 km, dont 522 en France, le Rhône a un bassin versant de 98845 km^2. Le fleuve est équipé de 21 aménagements hydroélectriques qui, outre la production d'électricité, le rendent navigable sur 310 km et permettent l'irrigation de 200000 ha de terres cultivées. Les installations nucléaires sont regroupées dans 7 sites.

En fonction des implantations nucléaires, plusieurs zones peuvent être définies :
- en amont de Creys-Malville, une zone soumise uniquement aux retombées des explosions atomiques,
- de Creys à Marcoule une zone permettant de mesurer l'impact des effluents des centrales,
- en aval de Marcoule une zone totalisant l'ensemble des termes sources y compris celui de l'usine de retraitement.

A partir de mai 1986 l'ensemble du Rhône est soumis aux retombées de Tchernobyl.

Des échantillons d'eau, de sédiment, de végétaux aquatiques et de poissons sont prélevés dans une soixantaine de stations. Après conditionnement ils sont mesurés par spectrométrie Ge ou par radiochimie. Les résultats sont stockés dans une base de données. Elle permet de sortir les bilans radioécologiques du fleuve, d'analyser l'impact des différents termes sources, de comparer le Rhône aux autres fleuves.

2. LES TERMES SOURCES

21. Les retombées des essais militaires ont libéré en 30 ans dans l'atmosphère $4,4.10^{24}$ Bq de produits de fission dont 1,32 EBq de Cs-137 et 1 EBq de Sr-90. Les retombées sont estimées, en France à 320 kBq/m^2, soit 31,6 PBq sur l'ensemble du bassin rhôdanien. Dans ces retombées il y avait essentiellement du Cs-137 et du Sr-90 (DOURY, 1986).

22. Les rejets des centrales sont de faible activité en comparaison des retombées. La limite annuelle autorisée par tranche (hors tritium) est de 0,56 TBq/an (PELLERIN & MORONI, 1987). Ces rejets sont variables suivant les années et les centrales (BIDARD, 1987). Ils contiennent principalement du Co-58 (40 à

75%), du Co-60 (17 à 25%), du Mn-54 (2 à 4%), Cs-137 (1 à 3%) et en quantité moindre du Cs-134, de l'Ag-110m, du Sb-124+125 etc...)

23. C'est au niveau de l'usine de retraitement qu'ont lieu les rejets les plus importants de tout le cycle du combustible. A titre d'exemple, en 1983 l'usine de Marcoule a rejeté environ 40 TBq de nucléides soit 80 fois plus que la centrale du Tricastin. Ces rejets contiennent du Ru-106 (<30 TBq/an), du Sr-90 et du Cs-137 (<6 TBq/an et en quantité plus faible d'autres nucléides tels que Ce-144, Mn-95, Zr-95,SR-89, Co-58+60, Ag-110m, Sb-125, Pu-238+239+240, Am-241, Eu-151... (CALMET & coll. 1985).

24. Le 26 avril 1986 l'accident de Tchernobyl a libéré dans l'atmosphère 37 PBq de produits de fission. Le 1er Mai 1986 un panache radioactif a survolé le bassin rhôdanien. Les dépôts au sol sont au maximum de 10 kBq/m^2 en Ru-103, 7,8 kBq/m^2 en Ru-106, 8,9 kBq/m^2 en Cs-134, 24 kBq/m^2 en Cs-137 (UIR,1987).

3. IMPACT DES TERMES SOURCES SUR LE RHONE

La radioactivité naturelle des compartiments du fleuve, due au K-40, aux 14 éléments de la famille de l'uranium et aux 10 éléments de celle du Thorium est stable sur l'ensemble du Rhône (FOULQUIER & coll. 1987) :

- 1 Bq/l d'eau
- 2250 Bq/kg (sec) de sédiment
- 1700 Bq/kg (sec) de végétaux immergés
- 110 Bq/kg (frais) de poissons.

A ces valeurs viennent s'ajouter l'impact des différents termes sources pour tous les radionucléides hors H-3) :

Activité en fonction du terme source dans les différentes zones du fleuve	Eau Bq/l	Sédiment Bq/kg sec	Végétaux Bq/kg sec	Poissons Bq/kg frais
Retombées	0,003	9	10	1,5
Centrales	0,02	40	70	4
Marcoule	0,5	1000	2000	20
Tchernobyl (mai 1986)	0,5	600 à 1500	1700 à 4000	10 à 40

31. Dans la portion du fleuve soumise uniquement aux retombées des explosions atmosphériques on retrouve systématiquement dans tous les échantillons le Cs-137. Il est à noter que les poissons du Léman ont des concentrations en Cs-137 trois fois plus élevées (0,98 Bq/kg frais) que ceux du haut-Rhône, montrant le rôle particulier des grands lacs et de leur bassin versant dans la concentration de ce nucléide (FOULQUIER, 1979). Le Sr-90 est visible dans quelques poissons et végétaux. Ainsi, après des décennies de retombées puis l'arrêt des essais atmosphériques la radioactivité artificielle est plus faible d'un facteur 100 à 1000 à la radioactivité naturelle.

32. En aval des centrales se retrouvent les nucléides les plus abondants dans les rejets (Cs-137, Cs-134, Co-58, Co-60, Ag-110m, Mn-54, Sr-90, H-3...). L'eau, du fait de son renouvellement est très peu chargée en radionucléides ; le sédiment concentre ceux qui s'adsorbent sur les particules (Cs, Co, Ag...).

Ce sont les végétaux qui sont les meilleurs radioindicateurs par la quantité et le niveau de radioactivité des éléments fixés. Les poissons permettent également de caractériser les rejets des centrales et parfois de mettre en évidence des nucléides qui n'étaient pas prévus par l'exploitant tel que le Zn-65 en aval du Bugey (LAMBRECHTS & FOULQUIER, 1987).

33. Les prélèvements effectués en aval de Marcoule montrent que les concentrations en radionucléides y sont les plus élevées. Ceux qui préexistaient en amont de l'usine augmentent de façon significative, d'autres n'apparaissent qu'à ce niveau (Ru+Rh-106, Am-241, Ce+Pr144, Pu 238+239+240...).

Après 30 années de fonctionnement de l'usine de retraitement, la radioactivité artificielle du Rhône en aval de Marcoule est toujours inférieure ou de l'ordre de grandeur de la radioactivité naturelle (FOULQUIER & coll., 1987).

34. Des campagnes de prélèvements ont été entreprises dans les divers secteurs du Rhône depuis Tchernobyl. Les radionucléides à vie courte issus de l'accident disparaissent en quelques semaines. Certains à vie moyenne tels que les Cobalt, le Mn-54, l'Ag-110m, le Sb-125 demeurent à des niveaux d'activité sensiblement identiques à ceux qu'ils avaient avant l'accident. Les césium et les ruthénium sont présents sur l'ensemble du fleuve. Le Cs-137 a des concentrations de 100 à 1000 Bq/kg sec de sédiment, de 100 à 400 Bq/kg sec de végétaux et de 5 à 30 Bq/kg frais dans les poissons, soit des niveaux supérieurs à ceux qui existaient en aval de Marcoule. Le Ru-103 apparaît dans tout le Rhône avec des concentrations importantes. Les retombées de Tchernobyl masquent l'effet de plusieurs années de rejets des installations nucléaires. Toutefois les concentrations mesurées restent basses et sans signification en terme de santé publique. Les prélèvements réalisés en juillet, novembre 1986 et mars 1987 montrent la disparition du Ru-103, une décroissance rapide du Ru-106 et beaucoup plus lente pour le radiocésium, en particulier chez les poissons.

4. CONCEPTION ET INTERET DES ETUDES RADIOECOLOGIQUES

Les exemples précédents démontrent que le suivi radioécologique régulier du fleuve permet de mesurer clairement l'impact des différents termes sources. Toute situation nouvelle peut être comparée à la situation antérieure et donner des informations utiles. L'information des données permet de manier des milliers d'informations et de mettre en évidence l'existence de paramètres qui régissent le devenir des nucléides dans l'environnement. Les données de terrain soulèvent des questions concernant les mécanismes de transfert, les cinétiques de fixation et d'élimination des radioéléments, les facteurs qui agissent sur ces processus. Chaque terme source induit un besoin de connaissance particulier : ainsi les retombées des explosions posaient les problèmes concernant le Cs-137 et le Sr-90, les centrales ceux des produits de fission et d'activation, Marcoule nous interroge sur les transuraniens et le ruthénium, Tchernobyl fait apparaître la question de la vitesse de fixation des nucléides en situation accidentelle et de l'épuration du fleuve. Les expériences de transfert en laboratoire apportent des réponses à ces questions. On peut simuler des radiocontaminations du milieu aquatique, étudier les échanges entre les divers compartiments des chaînes trophiques. L'analyse compartimentale permet de modéliser chaque

cinétique de transfert. Un modèle dynamique, cumulant les différentes équations, synthétise l'ensemble des transferts et explique les constatations faites in situ. Ce travail a déjà été réalisé pour le Cs-137 (LAMBRECHTS, 1984) ; il est actuellement en cours pour le Co-60 et pourra être étendu à d'autres nucléides.

CONCLUSION

Le suivi radioécologique du Rhône mesure l'impact des différents termes sources. Les observations faites in situ induisent des expériences de laboratoire qui permettent de connaître les cinétiques et les facteurs de transfert des radionucléides. Le faisceau des connaissances accumulées peut être confronté aux hypothèses de prévisions de rejets faites par l'exploitant des installations, en corriger les modalités et les faire évoluer en fonction de la situation radioécologique du fleuve afin de préserver l'environnement et la santé des populations (BOVARD & coll, 1975).

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RADIOECOLOGICAL PATHWAY OF ELEMENTARY TRITIUM STUDIED IN THE LABORATORY AND DURING RELEASE EXPERIMENTS

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ABSTRACT

The radiotoxicity of elementary tritium HT is 10^4-fold lower compared to tritiated water HTO. HT now released by reactors and reprocessing plants will be very important for the fusion technology where it is handled in amounts comparable to the world's natural T inventary. Biogeochemical studies as well as soil exposures in the laboratory have demonstrated that soil is a sink of HT (microorganisms, hydrogenase). Our studies have shown that HTO is the primary product, the organically bound form OBT a secondary one during usual biosynthesis (about 1 °/oo uptake per week). The uptake of HT by the soil is described by the deposition velocity v_d (calculated from the HT decrease in the air space above the soil core, HT measured by an I-chamber in a closed circuit, reemission by cool trapping HTO).
HT is converted in the surface layers of soil (50 % within first 1-3 cm). V_d is not governed by soil temperature but by its water content (free porous space). Therefore a distinct annual cycle can be observed (minimum during northern winter), also an influence of human use (v_d of field, pasture and forest ≙ 1:2:10, dependent on its content of organic material wich renders the soil light.
It is proposed to collect a set of soil samples around a nuclear facility and measure v_d and the reemission rate in the lab (demonstrated around the KFA Jülich and a French site). During the French release the applicability of the method was demonstrated under field conditions (comparison to v_d from release data using HT concentration of air and HTO content of soil) (v_d about 2×10^{-4} /reemission 1-5 % h^{-1}).
Agriculturally used soils of a site behave quite uniformly. The HT pathway then depends on the local water ecology, but may be predicted very well from v_d and reemission data.

TRANSFER FACTORS ACROSS THE HUMAN GUT FOR PLUTONIUM AND AMERICIUM IN SHELLFISH FROM NEAR SELLAFIELD

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INTRODUCTION

Data on gut transfer factors for environmental forms of radionuclides are essential for estimates of public radiation exposures following ingestion, and thus in decisions on controlling waste disposals. Dose estimates for transuranic nuclides are particularly sensitive to uncertainties stemming from gut transfer data being related to non-environmental forms and/or derived from animal experiments. The main parameter in question is f_1, the fraction of intake reaching human body fluids following ingestion, as applied in the model of the gastro-intestinal tract used by the ICRP (ICRP, 1979). The ICRP have recently reviewed the metabolism of plutonium and related elements (ICRP, 1986). Values of f_1 were derived from animal data; limited verification was provided by the only human data then available which was based on the low levels of fallout in foodstuffs (Mussalo-Rauhamaa *et al.*, 1984). The ICRP proposed a cautious value for f_1 of 10^{-3} for unknown or mixed compounds of Pu and for other actinides. However, it was recognised that this cautious value may not be appropriate in all situations where a best estimate of absorption is required; in such cases, if a different value more suitable to the specific situation can be justified, it should be employed.

A particular case in which estimated doses are significantly contributed by Pu and Am, and therefore dependent upon the values of f_1, is that of the critical group of fish and shellfish consumers near Sellafield. Two methods of measuring f_1 values, relevant in this situation, have been afforded by the use of sensitive techniques to detect plutonium and americium in the urine of consumers of shellfish collected near Sellafield.

EXCRETION BY CRITICAL GROUP CONSUMERS NEAR SELLAFIELD

Adult members of this group, who were known to eat molluscan shellfish from near Sellafield at relatively high rates (Leonard and Hunt, 1985) were asked if they were willing to provide samples of their urine. Ideally, these samples would be over 24 h, and for a number of days. Two males (consumers 117 and 128, ages 19 and 45 respectively) agreed to participate. A total of 15 days' samples were obtained. For intakes, data from our regular habits surveys, which include diary studies (Leonard, 1984) provided some data, but more detailed information was sought before and during the period of sampling, and also for years prior to those covered by our surveys. Concentrations of Pu and Am in the molluscs eaten were obtained by analysing appropriate sub-samples or from our regular monitoring data. Analyses of shellfish and urine were carried out at Lowestoft by chemical separation of Pu and Am and plating onto

stainless-steel discs, followed by alpha spectrometry. Long counting times (~ 8 weeks) were used for urine samples. Bulked samples of urine from Lowestoft-based donors were also analysed as controls.

Interpretation of the result for each day's sample in terms of f_1 was carried out using intakes combined with expected excretion rates; for Pu, these were derived using the function fitted by Jones (1985) to the classic volunteer data. For Am, the function due to Takada et al. (1984) was used. Detailed results are presented in Hunt et al. (1986). For three of the days of sampling by one of the volunteers, sample contamination was suspected and the results were excluded. For each consumer, results were combined using weightings dependent on counting errors. Results are shown in Table 1. An overall geometric mean is given, in consideration of likely metabolic variations between individuals.

Table 1 Gut transfer factors for critical group consumers

Consumer number	Days of sampling	Gut transfer factor (weighted arithmetic mean) x 10^{-4}	
		^{239}Pu + ^{240}Pu	^{241}Am
117	6	1.1 ± 0.4	4.2 ± 0.2
128	6	0.7 ± 0.2	0.7 ± 0.2
(Weighted geometric mean)		0.8 ± 0.1	3.9 ± 1.3

Errors represent 1σ based on counting statistics only.

EXCRETION BY VOLUNTEER CONSUMERS

The above experiment was subject to a number of difficulties due to control of intakes, obtaining 24 h samples, and the problem of estimating long-past intakes. An alternative experiment was therefore devised using Lowestoft-based adult volunteers with no, or minimal, prior consumption foods labelled with Pu and Am. Six males and two females provided urine samples after eating flesh from winkles collected near Sellafield. These winkles are freely available and their use in this way is not subject to potential ethical issues raised by the use of deliberately labelled foods.

Supplies of winkles were collected from near Sellafield, most of them by critical group members. The method of preparation replicated that used by the critical group. The approach used, also involving the least inconvenience to volunteers, was for them to eat the largest reasonable amount of subsampled winkles (about 450 g) during 1 day; excretion was followed by 24 h urine sampling for 2 days before and up to 12 days after consumption. To reduce counting errors, urine samples from a number of volunteers were often bulked. Four experimental runs were carried out as given below, the numbers of volunteers and bulking depending upon experience gained.

The first run used one volunteer (LT9: age 37) and was intended to check whether, and over what period, any enhanced excretion was indeed detectable. There was clear evidence of an increase in excretion above the background level after the day of winkle consumption. The result for each day (or group of days in later runs) was treated as an observation for the purpose of estimating f_1. Expected excretion was derived from the equations due to Durbin (1972) and Takada et al. (1984) for Pu and Am respectively. In the second run which used two male volunteers (LT13 and LT23: ages 43 and 44 respectively), we attempted to improve sensitivity by the use of winkles of higher concentration and by bulking each day's urine. To ensure representativeness, each volunteer's intake was about the same. The third run used two female volunteers (LT7 and LT30: ages 39 and 29 respectively) and, in an attempt to improve sensitivity further, bulking of urine was carried out for both volunteers and also for some of the later days. The excretion on day 1 appeared very high; contamination was suspected and this result was excluded from our interpretation. The fourth run was designed to be the most sensitive, using four volunteers (LT9, LT10, LT24 and LT36: ages 37, 44, 35 and 41 respectively) and bulking samples for later days. In this run no contamination was obvious in either pre- or post-intake samples, and counting statistics were improved. Detailed results are presented in Hunt et al., 1986; summarised results of all the runs are shown in Table 2. The f_1 values derived for each run are the arithmetic means of those for each day's (or bulked group of days') samples, weighted for counting errors. An overall geometric mean is also presented.

Table 2 Gut transfer factors for volunteer consumers

Experimental run	Consumers	Gut transfer factor (weighted arithmetic mean) x 10^{-4}	
		$^{239}Pu + ^{240}Pu$	^{241}Am
1	LT9	2.4 ± 1.2	0.6 ± 0.3
2	LT13, LT23	0.6 ± 0.2	0.36 ± 0.06
3	LT7, LT30	2.3 ± 1.7	2.2 ± 1.4
4	LT9, LT10 LT24, LT36	0.6 ± 0.2	0.65 ± 0.05
(Weighted geometric mean)		0.8 ± 0.3	0.6 ± 0.1

CONCLUSIONS

For Pu, there is good agreement between overall results for critical group members and for volunteer consumers; for Am, agreement is poorer but this may be due to metabolic factors influencing the result for consumer 117; the result for consumer 128 is in good agreement. Whilst the results obtained for the critical group members are important as described below, the uncertainties are greater than for the Lowestoft-based volunteers and for f_1 values

we would wish to place greater confidence on the results of the latter experiment.

The range of results are not inconsistent with the cautious f_1 value for Pu and Am of 10^{-3} recommended by ICRP (ICRP, 1986) for use in the absence of specific measurements such as described here; in the case of consumption of winkles from near Sellafield, the results would support a lower f_1 value. Our current method of estimation of dose to the critical group considers the effect of f_1 values of 10^{-4} and 5×10^{-4}, and on the basis of the present results this would appear to provide adequate protection. In the small sample of critical group consumers observed there was no evidence of significant excretion in excess of that predicted using these factors, for example, such as would have been the case if they were too low or values of intake from habits survey data were underestimated. Thus the results provide confidence in our habits survey data and methods. Further, the results also provide evidence that for these people there is no significant additional intake of actinides from any pathways not accounted for.

Because of the limited data from this investigation a second experiment using volunteer consumers is being undertaken to confirm these conclusions, and the results will be published as soon as practicable.

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THE USE OF THERMOLUMINESCENT DOSEMETERS IN MEASURING EXTERNAL EXPOSURE OF POTENTIAL MEMBERS OF A CRITICAL GROUP NEAR SELLAFIELD TO VERIFY DATA FROM HABITS SURVEYS

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INTRODUCTION

The pathways for radiation exposure of the public due to authorised discharges of low level liquid radioactive wastes from the major UK nuclear establishments are kept under review by means of habits surveys. These surveys focus upon members of the public who are potentially the most exposed, i.e. the critical groups. Most of the information comes from talking to these people and requesting their assistance in completing logging or diary records (Leonard et al., 1982; Leonard, 1984). In important cases, we have endeavoured to verify data in other appropriate ways and this paper describes one such study. This involved the critical group of boat dwellers in Whitehaven, Cumbria near the British Nuclear Fuels plc (BNFL) site at Sellafield. These people are exposed to gamma radiation whilst on their boats when the underlying silt is exposed by the receding tide (Hunt, 1984). Our normal method of dose assessment makes use of occupancy data from habits surveys combined with regular measurements of gamma dose rates using portable instruments. In this comparison, thermoluminescent dosemeters (TLDs) were also used to provide a measure of integrated doses received by a number of volunteer members of the group.

METHODS

Four boat dwellers on different boats agreed to wear TLDs and also have them placed on their boats in areas of high occupancy such as their berth (sleeping accommodation), cabin, wheelhouse or deck. $CaSO_4$(Tm) TLDs were used because of their sensitivity at low dose rates with minimal fading; they were calibrated against a ^{137}Cs standard source. TLDs on the boats were used in triplicate but as they were bulky due to a number of filters, only one TLD was worn, typically in a trouser or jacket pocket. The exposure period was 46 days. A number of TLDs were exposed as controls in suitably unaffected locations.

Gamma dose rates were measured on the boats while they were aground on the silt in their usual mooring locations, using a Mini Instruments meter, type 6/80, with energy compensated Geiger Müller tube (MC-70). Using our usual procedure, four 300 second scaler readings were made at each TLD location. These scaler readings were converted to absorbed dose rates "free in air" using calibration curves obtained using a ^{226}Ra standard source. A factor of 0.87 Sv Gy^{-1} was used to convert these absorbed dose rates to effective dose equivalent rates (Spiers et al., 1981). In addition, a continuous recording of the gamma dose rate was made in a spare berth of one of the vessels using a meter of the above type connected to a data logger. Details of each vessel's

movements during the exposure period were obtained from its log book. By consulting tide tables, the total amount of time that each boat was aground on the silt was calculated, and from details of each boat dweller's time at sea, on board in the harbour and on land, his dose was established.

RESULTS AND DISCUSSION

Table 1 shows how exposures in the berths of the vessels were derived from occupancy and dose rate data; the results are compared with the doses measured using TLDs. Table 2 presents similar information for doses to the boat dwellers; here, the maximising assumption was made in the calculation that all of their time on board was spent in their berths, which is the area of highest occupancy and dose. The absorbed dose rate "free in air" on the vessels when afloat was consistently found to be close to 0.10 $\mu Gy\ h^{-1}$, and this value was used in the calculations.

Table 1. Doses in the berths.

Boat	Hours aground on silt	Dose rate in berth ($\mu Gy\ h^{-1}$)	Calculated exposure in berth (μSv)	Dose measured using TLD (μSv)
A	270	0.30	143	125
B	248	0.37	154	182
C	360	0.38	184	321
D	320	0.44	191	310

Table 2. Doses to boat dwellers.

Dweller on boat	Hours spent on boat when aground	Dose rate in berth ($\mu Gy\ h^{-1}$)	Calculated dose to person (μSv)	Dose measured using TLD on person (μSv)
A	79	0.30	110	126
B	182	0.37	139	174
C	266	0.38	161	241
D	236	0.44	166	198

In discussing these results, it is convenient first to deal with the relationship between doses in the berths (Table 1) and to the persons (Table 2). The latter may be expected to be lower mainly because of time spent off the boat, and this is generally reflected in the results. However, the measured results for persons A and B are rather similar to the measured and calculated exposures in the berths. Boat dweller A said that he had been reluctant to wear his TLD ashore, and boat dweller B admitted to sometimes forgetting to transfer his TLD to his shore-going clothes. Hence, there is fair agreement between the supposedly-measured dose to the person and the calculated and measured doses in the berths. We believe, therefore, that for these persons the calculated exposures give a better indication of those actually

received. In this case, the vagaries of human nature would seem to render the calculational approach generally more reliable than one in which people were regularly asked to wear TLDs.

Comparing the calculated and measured results in the berths, and for boat dwellers C and D, the TLD results appear generally higher than the calculated exposures by up to 70%. In order to assess any discrepancies which might be expected, a series of investigations was carried out (A. Bolsover, personal communication). The calibrations of the TLDs and the portable dosemeter were both checked against a standard source of ^{137}Cs, the predominant radionuclide giving rise to external exposure over silt in Whitehaven harbour. Any pre-existing differences were shown to lead to comparatively little error at photon energies characteristic of ^{137}Cs and ^{226}Ra but it was recognised that much of the dose in the environment is due to scattered photons of lower mean energy. The response of the $CaSO_4$ TLD is relatively higher than the portable dosemeter at these lower energies. Estimates based on the energy responses of both detectors and the characteristics of the energy spectrum showed that the TLDs could produce measured doses enhanced by up to 20% over those using the portable dosemeter. Further, the response of the TLD was found to be sensitive to orientation; unidirectional irradiation from the side produced a reading enhanced by up to a factor of 2. Scattering from objects on the boat could have influenced the anisotropy of the radiation field. No control could reasonably be exercised over the wearer and the TLDs' orientation would have changed repeatedly (e.g. when asleep). Bearing these uncertainties in mind, we

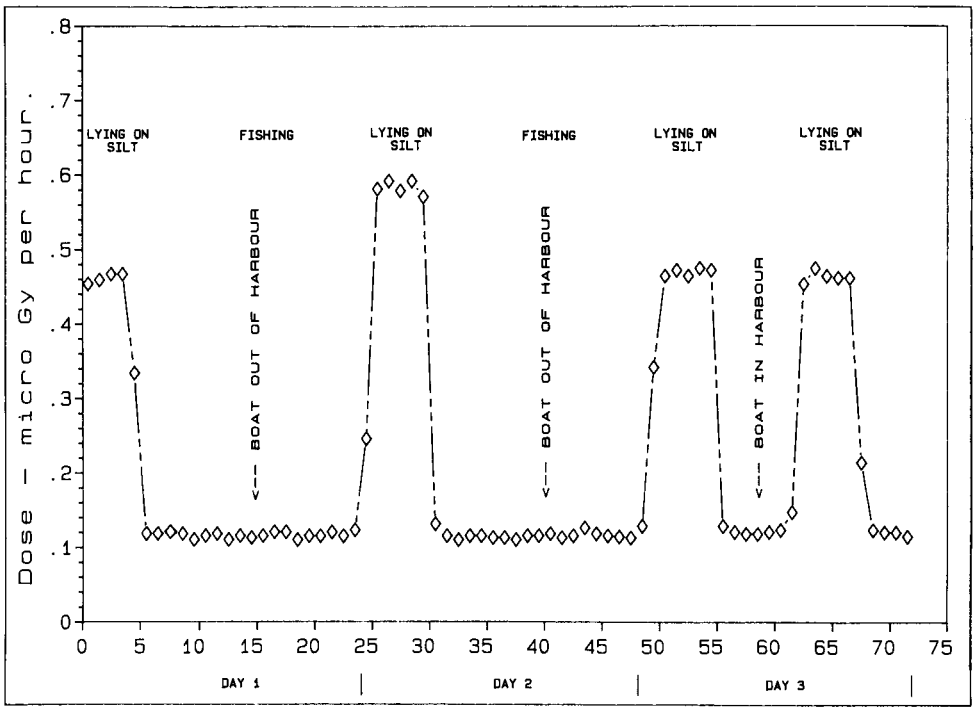

Figure 1. Hourly gamma dose rates on board a fishing vessel measured using our portable dosemeter and data logger.

regard the comparison as sufficient to support the results for calculated exposures and, hence, the habits survey data on which they are based.

Clearly, the habits survey data also contain generalisations and it is important during regular monitoring to take account of any unusual circumstances. This was illustrated during this work by the use of the dose rate recording logger. Usually, consistent results were obtained when the boat on which it was installed was aground on the harbour silt in its usual mooring position or afloat. The changes in gamma dose rate due to the tide are shown in Figure 1. On one occasion, the boat's mooring was changed, and this gave rise to a higher reading. The integrated dose over the whole exposure period was not significantly affected but this experience indicates the need, during regular monitoring using spot measurements, to carry out representative observations at appropriate locations established from habits survey data.

CONCLUSIONS

The TLDs provided a useful means whereby external exposures to a critical group could be measured and compared with our usual method of estimation, based on data from habits surveys. The results of the comparison reported here were satisfactory, taking account of the uncertainties inherent in both methods and the differences in responses of the two types of detector. We would not recommend the TLDs for general use: apart from their intrusive nature and presentational implications, they suffer from drawbacks due to human error and technical disadvantages such as variation of response with orientation and energy. The results of this work generally support the results for calculated exposures from the habits survey data on which they are based, and we intend to continue with the habits survey method for general use.

ACKNOWLEDGEMENTS

We gratefully acknowledge the help of the boat dwellers, as well as Mr A. Bolsover and Dr P. Grimwood of BNFL for arranging for the supply and analysis of the TLDs.

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DOSE TO MAN FROM THE CONSUMPTION OF MARINE SEAFOODS: A COMPARISON OF NATURALLY-OCCURRING AND ARTIFICIAL RADIONUCLIDES IN FISH AND SHELLFISH

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ABSTRACT

The consumption of above average quantities of marine seafoods has for a long time been a characteristic of critical groups associated with the authorised discharge of low-level liquid radioactive wastes into UK coastal waters. Such consumers are also of special interest in other countries discharging low-level radioactive wastes into the sea. The concentrations of artificial radionuclides in these seafoods are routinely monitored, and the resultant committed effective dose equivalent calculated for the purposes of radiological protection.

There is an increasing interest in the epidemiological aspects of risk and dose associated with identified critical groups and those living in the neighbourhood of nuclear facilities, but little attempt is made to consider their total exposure to radiation. In this respect it is of interest to compare such dose assessment with those which could arise from naturally-occurring radionuclides present in seafoods. For example, the concentrations of ^{210}Po in fish alone could result in about 0.1 mSv a^{-1}, depending on the species eaten. In addition there are contributions from, in particular, ^{226}Ra and the uranium isotopes. The dose resulting from shellfish consumption is less readily calculated. Shellfish have higher concentrations of ^{210}Po than fish, but the quantities and range of species consumed is much more variable. These factors are discussed in relation to the committed dose equivalents which have been estimated for transuranium and fission product nuclides ingested in the neighbourhood of major nuclear establishments in the UK.

APPLICATION OF PARAMETER UNCERTAINTY ANALYSIS TO THE ACCUMULATION OF Cs-137 IN FISH

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ABSTRACT

A metabolic model was used to predict the accumulation of ^{137}Cs in plaice (Pleuronectes platessa L.). Model verification was executed using published results on the accumulation of ^{137}Cs from both from in situ measurements and from experimental setups. Uncertainty analysis was performed in order to determine the contribution from the separate model parameters to the total uncertainty for both the separate exposure pathways and for the body burden of ^{137}Cs in fish. Generally, the metabolic regulation of ^{137}Cs in the gills creates a large variation in the final outcome, i.e. the concentration of ^{137}Cs in the fish. The model assumption regarding the uptake from water is also loosely supported, and priority should be given the study of this mechanism. The next most important parameters contributing to the variation in the body burden are the concentration of ^{137}Cs in the food, and the exponent relating metabolism to body weight.

CONCENTRATION FACTORS OF STABLE ELEMENTS AND RADIONUCLIDES IN PO RIVER FISH

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INTRODUCTION

The concentration factors (CF) of stable Co, Cs, Mn, Fe, Zn and Sr in different fish from six stretches in the middle course of the Po river (N. Italy) have been investigated. The space-time variation in water has been followed for 14 months. The investigation has been undertaken to study CF variations in the same fish species as a function of the physico-chemical form of the different elements in water (dissolved, dissolved and exchangeable fraction of the particulate, total). CF values of ^{103}Ru, ^{131}I and $^{134-137}Cs$ were also investigated for **Cyprinus carpio** reared, with artificial food, in two semi-natural environments.

EXPERIMENTAL

Trace elements concentration in water was determined by ICP-AES or AAS either directly (Sr) or after enrichment with NH_4 form chelex-100[1] (Mn,Fe,Zn,Co) or copper ferrocyanide[2](Cs). Radionuclides were determined by gamma-ray spectrometry (Ge-Li detector) after preconcentration on NCFC columns ^{137}Cs or on ion exchange resins. The exchangeable fraction and the total content of elements in the particulate (>0.4 μm) was determined after leaching with 0.3 M HCl, or total acid digestion[3]. Trace elements in fish were determined by ICP-AES (Fe, Mn, Zn, Sr) or AAS (Co, Cs) following wet digestion with HNO_3-HF-$HClO_4$ mixture. Cs was separated before analysis, with copper ferrocyanide. Radionuclides were determined by direct gamma-ray spectrometry.

CF FROM FIELD OBSERVATIONS

Po river water (<0.4 μm) quality and trace elements content are reported in tab. 1. The analyzed species and their subdivision in different diets[4] are : **Planctivorus** = Alburnus alburnus alborella (sample size=7): **Carnivorus** = Barbus barbus plebejus(8), Cyprinus carpio(1), Perca fluviatilis(4), Lepomis gibbosus(5), Ictalurus melas(3), Tinca tinca(1), Rutilus rubilio(4); **Omnivorus** = Carassius carassius(1), Leuciscus cephalus cabeda(11), Chondrostoma sp.(1), Chondrostoma toxostoma(9), Chondrostoma soetta(9), Gobio gobio(3), Scardinius erythrophthalmus (1), Padogobius martensi(1); **Predator** = Esox lucius (1), Micropterus salmoides(3).

Notwithstanding a space-time variability of some stable element concentrations in water (Co, Zn, Sr) was observed (analysis of variance; $P<0.05$), the geometric mean on the all samples and

for each element was used to compute CF.

CF values (wet weight) were computed for each species and trace element both for edible and not edible part. The Duncan's test showed differences among species (for the same element) of little significance; for this reason CF values were analysed only considering the feeding habits (tab. 2). Duncan's test on these data showed difference only for Zn, ^{137}Cs and Fe (edible part); higher values for planctivorus (Zn and Fe) and both planctivorus and predator (^{137}Cs) were obtained. For planctivorus fish (Alburnus a.a.) the data refers to the total sample. The Duncan's test was also applied to search differences between edible and not edible parts for each element. In this case differences were observed only for Sr (omnivorus and carnivorus) and Mn (omnivorus). CF values for ^{137}Cs are generally lower than those for stable Cs, thus suggesting the occurrence of a non equilibrium condition. CF values (edible part) were also evaluated (tab.3) according to the different physico-chemical forms of each element in water (dissolved: <0.4 μm; dissolved plus exchangeable fraction of the particulate; total). Only for Sr quite similar values were obtained; for the other elements higher CF values are obtained when the dissolved elements concentrations are considered. The latter confirms the difficulty to compare experimental and literature data, also considering that, for some elements (e.g. Cs and Sr), CF are greatly depending on the water and sediment quality (e.g. trophic state, K and Ca concentration etc.)[5].

CF FOR RADIONUCLIDES IN CYPRINUS CARPIO AFTER CHERNOBYL ACCIDENT

Cyprinus carpio were periodically collected after Chernobyl accident. Space-time variability of ^{103}Ru, ^{131}I and ^{137}Cs concentrations in edible and not edible parts as well as those in water (dissolved: <0.4 μm, solid phase: >0.4 μm) are shown in fig. 1 for ^{137}Cs. Only fish and water samples collected after May 12th. 1986 were considered in computing CF (quasi-steady state condition of radionuclides concentration in water). Geometric mean of CF data was computed,as suggested by De Bortoli et al.[6], for two different period of time. In the first period (30 days) equilibrium was not reached yet and ^{137}Cs CF is only due to the direct contamination from water; the obtained value (CF=29) is in good agreement with homogeneous literature data[7]. Also in the second period (90 days), equilibrium was not yet completely reached, but the value for ^{137}Cs (CF=74) include contamination both from water and food[7]. CF values for ^{103}Ru and ^{131}I, computed for the same period of time, agree fairly well with literature data [8] and, as the case of ^{137}Cs, point out the occurrence of a non equilibrium condition.

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Tab. 1 : Po river water (<0.4µm) quality and trace elements content

quality parameters	min	max	trace elements	sample size	geometric mean	σ	min	max
O_2 (ppm)	7.5	10.5	Co (ppb)	28	0.099	3.03	0.022	0.95
T(°C)	7.7	20.9	Cs (")	20	0.010	1.69	0.004	0.027
cond.(µScm^{-1})	74	437	Mn (")	26	15.2	2.11	5.46	54.0
pH	7.02	8.25	Fe (")	28	4.15	2.48	1.17	4.01
T.O.C. (ppm)	2.2	3.7	Zn (")	28	4.36	1.94	1.67	22.5
Cl^- (")	3.0	17.6	Sr (")	28	339	1.32	238	591
SO_4 (")	23.0	71.6						
PO_4 (")	0.011	0.228	^{137}Cs (Bq/m^3)	22	14.5	1.77	7.5	43.7
NO_3 (")	4.4	18.4						
Na (")	3.7	20.5						
K (")	1.6	4.7						
Ca (")	32.5	67.8						
Mg (")	6.0	16.7						
NH_4 (")	0.02	0.69						

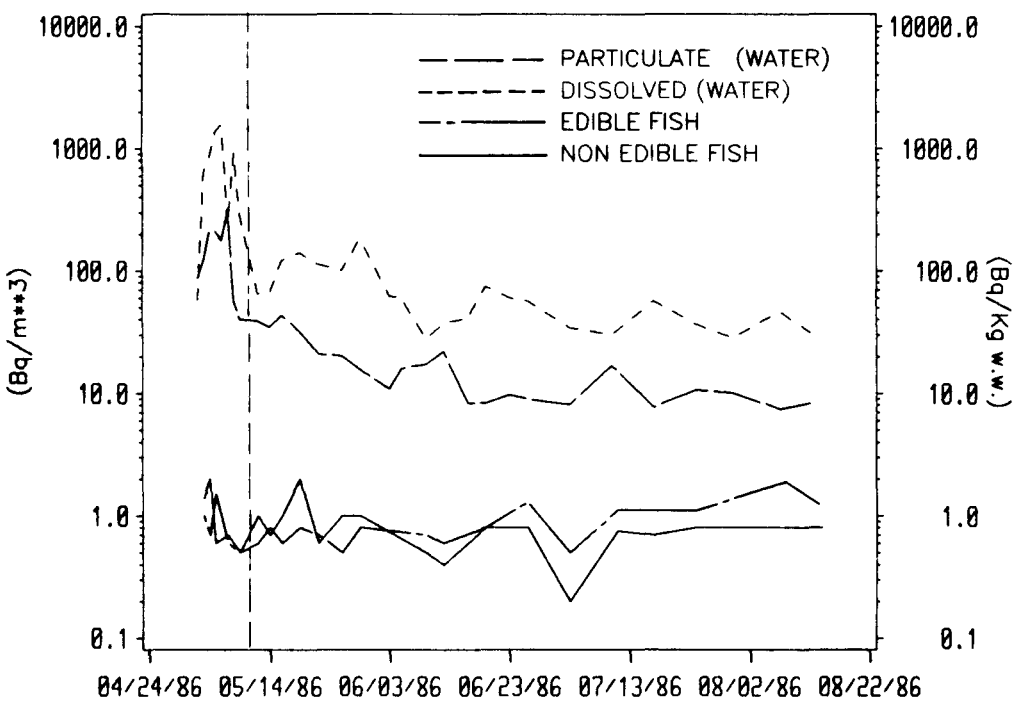

Fig. 1: Temporal variations of the ^{137}Cs concentration in Cyprinus carpio and in the Po river water

Tab. 2 - CF values (wet weight) for different feeding habits and elements

Element	PLANCTIVORUS		CARNIVORUS				OMNIVORUS				PREDATOR			
	TOTAL		EDIBLE		NOT EDIBLE		EDIBLE		NOT EDIBLE		EDIBLE		NOT EDIBLE	
	n.	mean	n.	mean	n.	mean	n.	mean	n.	mean	n.	mean	n.	mean
Co	7	2420	23	3840	23	4140	33	3840	30	3840	4	4240	2	2320
Cs	7	1000	24	680	19	970	32	900	30	760	4	330	2	980
Mn	7	370	26	185	23	360	32	900	30	570	4	60	2	170
Fe	7	10390	27	3950	24	6070	36	250	33	6190	4	2320	2	3570
Sr	7	69	27	47	24	121	36	3930	33	160	4	24	2	110
Zn	7	14700	27	3810	24	5410	36	4771	33	8170	4	2150	2	6580
^{137}Cs	3	2050	10	360	9	310	12	510	9	521	1	(4870)	1	3060

Tab. 3 - CF values (wet weight) for feeding habits with respect to different physico-chemical form of each element in water. 1 (dissolved), 2 (dissolved plus exchangeable fraction of the particulate matter), 3 (total).

Element	PLANCTIVORUS			CARNIVORUS			OMNIVORUS			PREDATOR		
	1	2	3	1	2	3	1	2	3	1	2	3
Co	2420	510	330	3840	810	520	3840	810	520	4240	890	380
Cs	1000	530	38	680	360	26	900	470	35	330	170	13
Mn	370	88	71	185	44	35	250	61	40	60	14	11
Fe	10390	140	33	3950	53	13	3930	52	13	2320	31	7
Sr	69	62	62	47	42	42	57	51	51	24	22	22
Zn	14700	2850	2110	3810	1020	760	4771	1280	950	2150	580	430
^{137}Cs	2050	410	-	360	73	-	510	100	-	4870	970	-

THE TRANSPORT AND DEPOSITION OF RADIONUCLIDES DISCHARGED INTO CREEK WATERS FROM THE RANGER URANIUM MINE

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INTRODUCTION

The Ranger uranium mine in the Alligator Rivers Region of Australia's Northern Territory has been operational since 1981. Critical pathway analysis has shown that water transport of radionuclides from the mine site could be an important route for radiation exposure of members of the public; in the short-term such exposure could be as a result of the direct discharge of effluent waters into the nearby Magela Creek and in the long-term as a result of seepage of water from tailings repositories or the transport of erosion products from the rehabilitated site. In order to estimate the radiation exposure of the public from such processes it is necessary to model the transport of radionuclides in the creek and its associated flood plain.

Such a model has been developed for radionuclides in the particulate phase by studying the transport and deposition of naturally occurring radionuclides. The two major components of this program consisted of measurements of concentrations of all the long-lived nuclides of the uranium series in the waters of Magela Creek and in the sediments of the flood plain.

RADIOACTIVE DISEQUILIBRIUM IN SURFACE WATERS

The tropical monsoonal climate of the region is characterised by contrasting Wet and Dry seasons. The Magela Creek flows seasonally and during the Wet season flow can exceed $1000 m^3/s$ in periods of cyclonic depression. The significance of floods in the transport of suspended matter has been well documented for a number of rivers of the world and, in particular, for Magela Creek (Hart et al. 1982). The extent to which radioactive disequilibrium occurs in the particulate matter carried by the flood waters of Magela Creek has been used in this study as a means of identification of the source of sediments that have been deposited on the flood plain. Two floods were studied in detail; 19-21 February 1985 when a maximum instantaneous flow rate of 250 m^3/s was recorded, and 10 April 1986 when the maximum flow rate was 60 m^3/s. Water samples were collected every 2 hours throughout the floods and both filtered and unfiltered ($<0.45\mu m$) samples were analysed by alpha-particle spectroscopy (Martin and Hancock 1987).

The variation of total concentrations with time through the 1985 flood is shown in figure 1 for the isotopes ^{238}U, ^{230}Th, ^{226}Ra, ^{210}Pb, and ^{210}Po. A pronounced peak is observed in the concentrations of ^{226}Ra, ^{210}Pb and ^{210}Po about 10 hours after the start of the sampling program; this peak was in advance of the hydrograph which reached its maximum about 7 hours later. No such peak is observed for ^{238}U, and for ^{230}Th and ^{234}U (omitted for clarity) there was only a slight enhancement of concentrations. The concentrations of all nuclides in the filtrate remained essentially constant throughout the flood.

Hart et al. (1982) observed a similar peak of total concentrations in advance of the hydrograph for suspended solids and the metals Mn, Zn and Cu in their study of flood events in Magela Creek during the 1978-79 Wet season. In the present context the main significance of the current data, however, is the demonstration of a substantial disequilibrium in the uranium series radionuclides between ^{230}Th and ^{226}Ra in the suspended matter carried by Magela Creek.

The concentrations of ^{226}Ra in the particulate matter of samples collected during the flood of 10 April 1986 were obtained both with respect to water volume and with respect to the mass of suspended matter. The former data again show a peak in the radionuclide concentrations in advance of the hydrograph but the latter remain approximately constant throughout the flood; the mean value and standard deviation were 220 Bq/kg and 40 Bq/kg respectively. Thus the peak in radionuclide concentrations during floods is attributed solely to the increased concentrations of suspended solids and not to any significant variation in the concentration of radium in the suspended material. Analysis of similar material collected outside of flood periods shows similar constancy of radium concentrations at about 210 Bq/kg and concentrations of ^{230}Th of about 50 Bq/kg deduced from thorium systematics.

RADIONUCLIDE CONCENTRATIONS IN SEDIMENTS

Some 12 km downstream from Ranger the Magela Creek enters a floodplain system; in a typical Wet season the inundated area is about 200 km^2. Average values of radionuclide concentrations were obtained on 19 transects of the creek and floodplain by combining two measurement techniques. A rapid quantitative assessment of the distribution of natural radionuclides (primarily ^{226}Ra, ^{228}Ra, and ^{40}K) was provided by gamma-ray dose rate measurements. These dose rate data were then used to select a typical site on each transect from which sediment cores were obtained and subdivided into 3 cm sections. These sections were analysed by gamma-ray spectroscopy (Murray et al. 1987) to obtain concentrations of ^{238}U, ^{226}Ra, ^{210}Pb, ^{228}Ra, ^{228}Th and ^{40}K; the standard error in the dose rate data for each transect was then used as an estimate of the variability of individual concentrations. The reliability of this technique was confirmed by detailed study of one complete transect.

Concentrations of ^{238}U and ^{226}Ra in the top 3 cm of sediment are plotted as a function of distance downstream from the beginning of the floodplain in figure 2. In the sandy creek region ($x < 0$) concentrations are, as expected, low. The absolute concentrations and the extent of radioactive disequilibrium observed at the southern end of the floodplain ($0 < x < 5$ km) agree very well with observations on the suspended matter carried by the creek. However, disequilibrium is not observed on the northern part of the floodplain ($x > 20$ km). Therefore, since sedimentation rates are (from geomorphological evidence) approximately independent of location, sedimentation in the northern region (which accounts for about 80% of the floodplain area) must have arisen from inputs other than Magela Creek even though this creek provides about half the volume of water entering the floodplain.

INTERPRETATION

These data have been interpreted within the context of a two-compartment model of sedimentation; input of particulate matter from Magela Creek with a concentration of radium C_m, and input from the remainder of the catchment with concentration C_r. The average concentration of radium in surface sediments at distance x, C(x), is then given by

$$C(x) = (C_m - C_r)f(x) + C_r \qquad (1)$$

where f(x) is the fraction of sediment at x that arises from deposition of particulate matter from Magela Creek. The areal rate of deposition of radium, R(x), is related to both the concentration C(x) and to the rate of change of the radium load, L(x), in the water column; that is

$$R(x) = \rho r(x)\ C(x) \qquad (2)$$

and

$$R(x) = -\ (dL/dx)/w \qquad (3)$$

where ρ is the density of sediment, w is the floodplain width, and r(x) is the total sedimentation rate at x.

The form of f(x) and the values of the variables C_m and C_r have been determined from the radium data in figure 2; f(x) is given by $\exp(-x/\lambda)$ where the attenuation length λ is 8.5 km, C_m = 190 Bq/kg and C_r = 50 Bq/kg. Thus C_m is in good agreement with the observed value of radium concentrations in particulate matter of Magela Creek, namely about 220 Bq/kg. Assuming r(x) does not vary substantially over the region $0 < x < 3$ km, equations (2) and (3) can be solved to determine the current sedimentation rate, r, near the southern end of the floodplain. Thus $r = L_s/(w\lambda\rho)$ where L_s is the annual average load of suspended solids carried by Magela Creek. Using known values of w and L_s, 1 km and 5000 tonnes/y respectively, and the value of λ deduced above gives r = 0.4 mm/y. This value is in good agreement with independent estimates using thorium-radium dating of floodplain sediments from the same location.

CONCLUSION

The principal conclusion of this work is that particulate matter from Magela Creek appears to contribute to sedimentation on only a small fraction of the 200 km^2 flood plain. Using the value of λ determined above, about 90% is deposited in the first 18 km of floodplain with an area of about 30 km^2. Thus radionuclides released from Ranger in the particulate phase or which partition into this phase soon after discharge have a high probability of being retained within a small region of the floodplain system.

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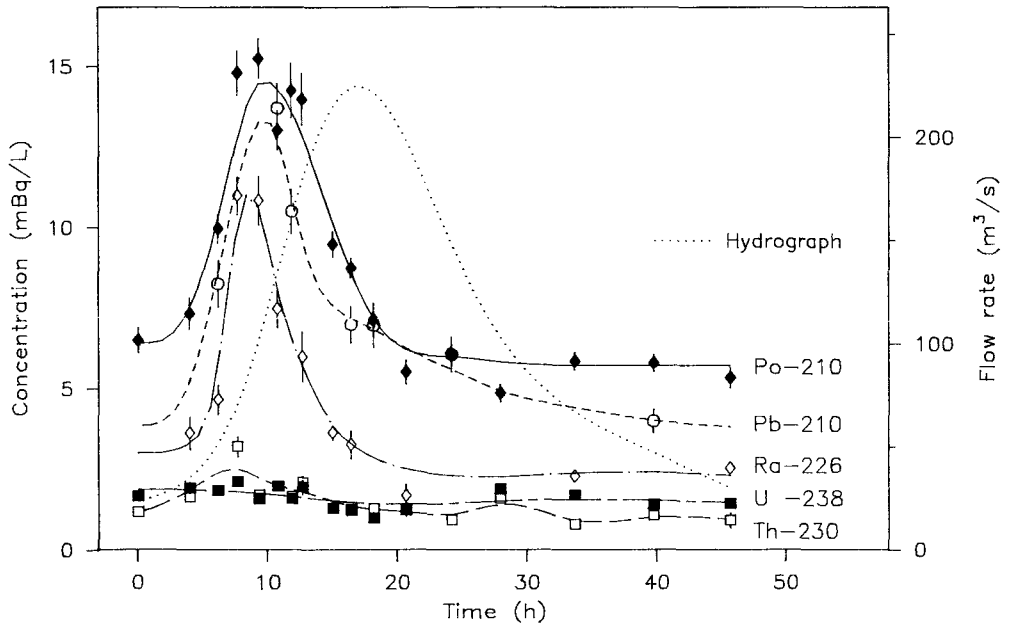

Figure 1 Variation of the concentrations of uranium series radionuclides in unfiltered water from Magela Creek during the flood of 19–21/2/85

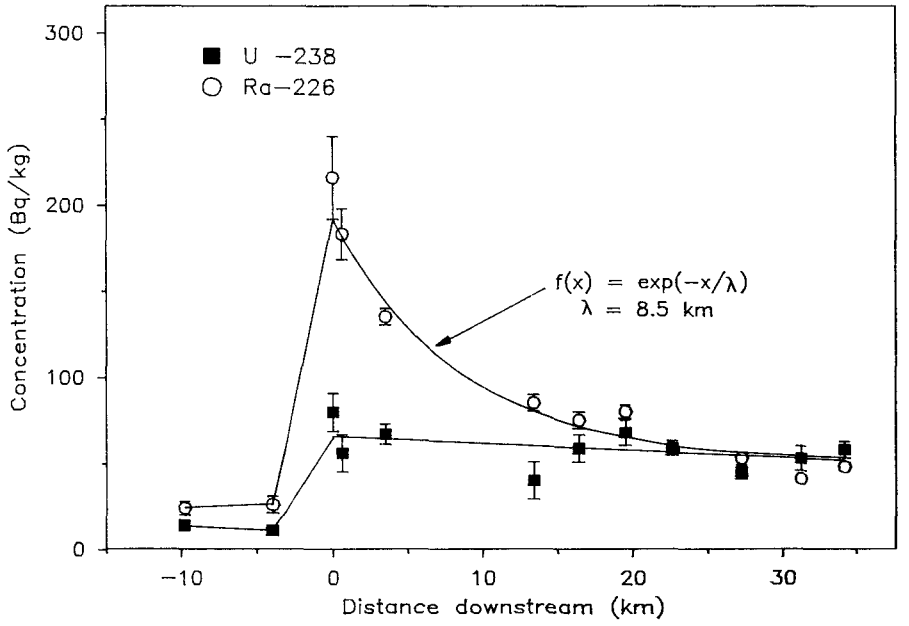

Figure 2 Radionuclide concentrations in the sediments of the Magela Creek and flood plain as a function of distance downstream from the beginning of the floodplain

CELLULAR AND SUBCELLULAR DISTRIBUTION OF URANIUM AND TRANSURANIC RADIONUCLIDES IN MARINE ORGANISMS.

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Since radionuclides have been introduced in the marine environment by several ways (nuclear weapons tests, nuclear fuel cycle operations, accidental releases), it is a topic of interest to assess radionuclides bioavailability to marine organisms, in particular to those of economic interest such as shellfish.

Cellular and subcellular distribution of ^{238}U, ^{239}Pu and ^{241}Am was examined by means of microanalytical techniques in several organisms: oysters, mussels, shrimps, crabs and sea spiders collected from the French coastal waters (Channel, Mediterranean Sea and Atlantic Ocean). Secondary Ion Mass Spectrometry (Ion Microscope and Ion Microprobe) made possible to obtain isotopic measurements and cellular images of the radionuclides distribution. A post-acquisition image processing system was used in association with SIMS for multi-image correlation. X Ray Spectrometry (Camebax: Castaing Microprobe associated with a Transmission Electron Microscope) enabled investigations at the subcellular level.

Using both of these techniques, we were able to detect in every species, target organs (Fig.1 to 5), cells and organelles (Fig.6,7 and table) of radionuclides bioaccumulation and to elucidate some of the physiological strategies involved in the uptake, storage and elimination of these radioactive elements, concentration factors ranging from 10 to 2×10^3.

Compared to other analytical techniques (histoautoradiography, radiochemistry etc...), microanalysis, for which not much biological material is needed, provides accurate advantages such as a high sensitivity and very short term investigations, for the ecotoxicological control and watch of radionuclide pollution in the marine environment.

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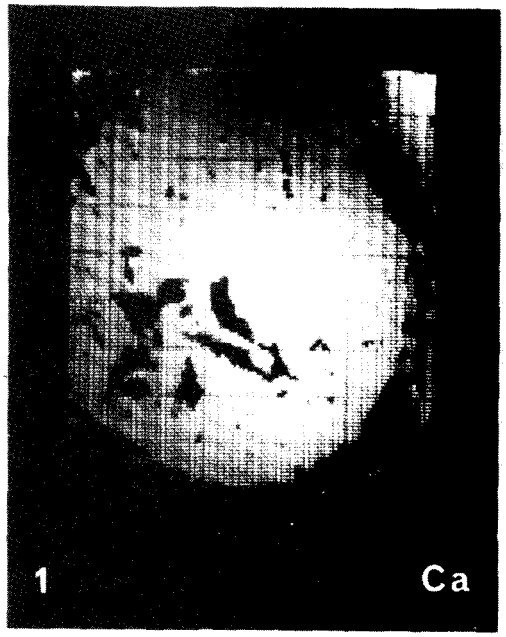
This work was supported by CNRS, INSERM (S.C.27) and the French Ministry of the Environment (contract 84244) with the collaboration of ISA RIBER Instruments (M.Outrequin, R.L.Inglebert), CEA (Institut de Protection et de Sureté Nucléaire) and the technical assistance of F.Escaig, P.Boumati, F.Kleinbauer and J.Brissard. We thank Dr.J.F.Cavellier for his assistance to SIMS Image Processing.

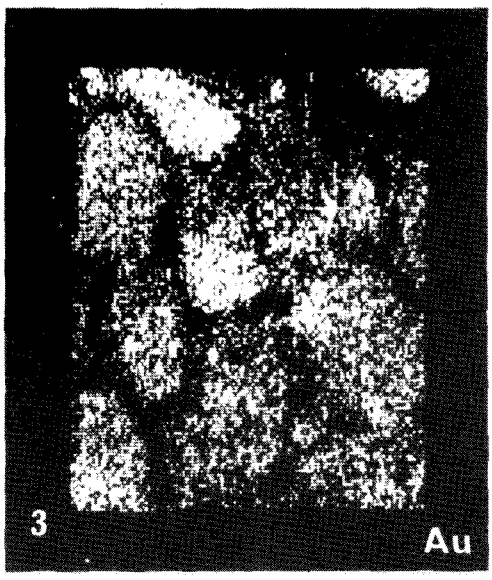


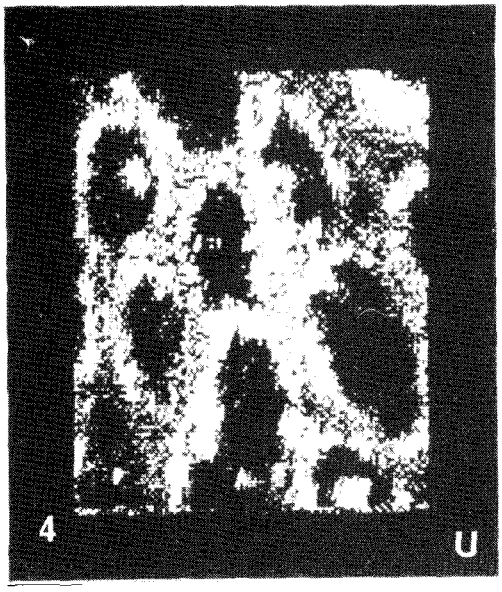
Sea Spider *Maia squinado* (Channel), Digestive gland.(200 µm full horizontal scale)
Ion probe images obtained from the same area (RIBER MIQ 156), 1: $^{40}Ca^{+}$,2:$^{239}Pu^{+}$
Elements appear as bright points in the biological section.

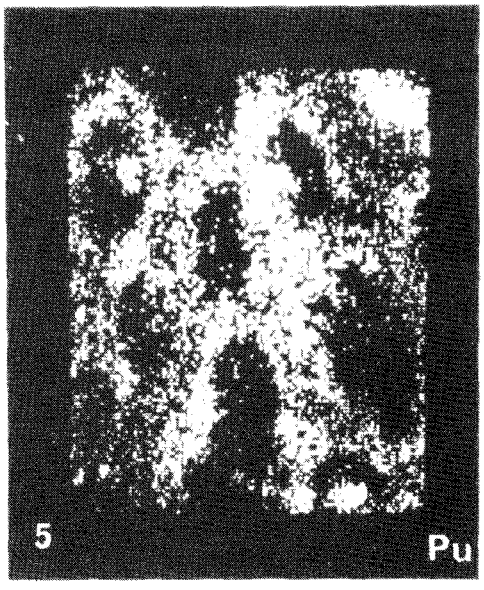
ORGANELLES	LYSOSOME (*DIGESTIVE GLAND*)	SPHEROCRYSTAL (*KIDNEY*)
ELEMENTS		
U	2750 ± 221	1820 ± 217
P	3020 ± 247	1937 ± 215

TABLE

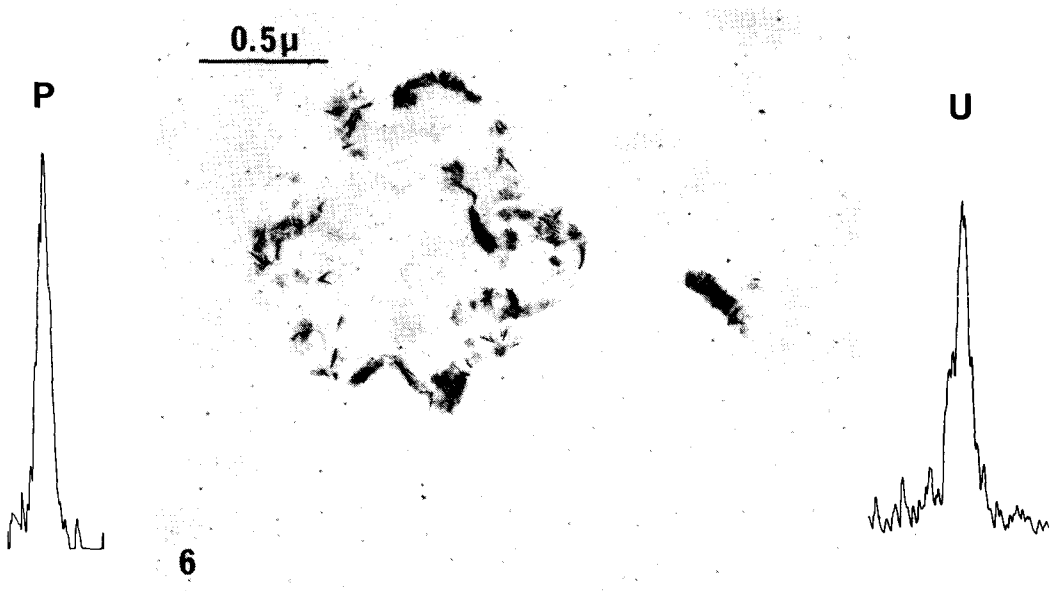
Mytilus edulis
*Electron probe X Ray microanalysis (CAMEBAX) of the organelles shown in Figures 6 and 7 (counting time : 100 sec.).*In these organelles,uranium is associated with phosphorus in the form of an unsoluble phosphate.

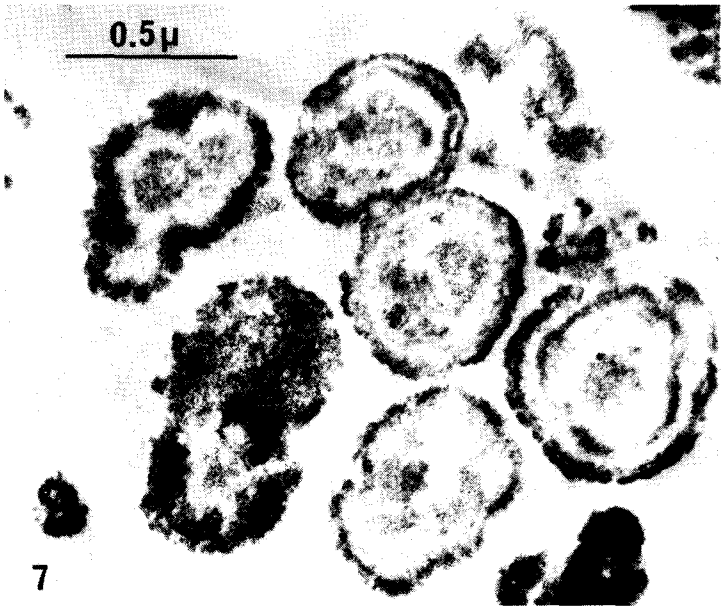






Mussel *Mytilus edulis* (Channel),Digestive gland.(200 μm full horizontal scale) Ion probe images obtained from the same area (RIBER MIQ 156),3: $^{197}Au^{+}$, 4: $^{238}U^{+}$, 5: $^{239}Pu^{+}$. On the Au image,biological section appears dark and the gold of the specimen holder appears as bright points.On the U and Pu images,the elements appear as bright points in the biological section.





Mussel *Mytilus edulis* (Channel).

6:Electron image showing a lysosome with microneedles of uranium phosphate(unstained and non osmicated section).X Ray emission spectra of P and U,obtained from this lysosome are shown on each side.

7.Electron image showing spherocrystals where uranium phosphate is concentrated.

MEASURED DEPOSITION VELOCITIES AND RAINOUT COEFFICIENTS AFTER THE CHERNOBYL ACCIDENT COMPARED WITH THEORETICAL MODELS AND EXPERIMENTAL DATA

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INTRODUCTION

The determination of different radioecological parameters for the calculation of the transport of radionuclides in the environment was, besides radiation protection measurements, an important aim of the post Chernobyl measurements in Aachen. As the growth had just begun, the only vegetations which allowed extensive deposition measurements during the first days were grass and stinging nettles.

DRY DEPOSITION OF PARTICLEBOUND NUCLIDES AND RADIOIODINE ON GRASS

Fig. 1 shows the time integrated activity concentration in the air at Aachen for Cs 137 and I 131; the concentrations of other important nuclides are published in /1/. About 35% of the total amount of iodine was bound to particles. The elemental fraction of the gazeous iodine was derived from /2/, /3/. Besides iodine, all nuclides detected in Aachen were particlebound. The activity size-distributions of the particlebound nuclides were measured with an impactor. The graphs (also given in Fig. 1) are the result of these measurements. They are corrected for the separation curves and the wall losses of the instrument. The deposition of airborne radionuclides on a green near the institute was measured hourly. Until the first rainfall (6.30 p.m., 3rd of May) the deposited activity of particlebound nuclides (e.g. Cs 137, fitted curve on the left hand side of Fig. 2) increased continuously. The dry deposition on extremely short or

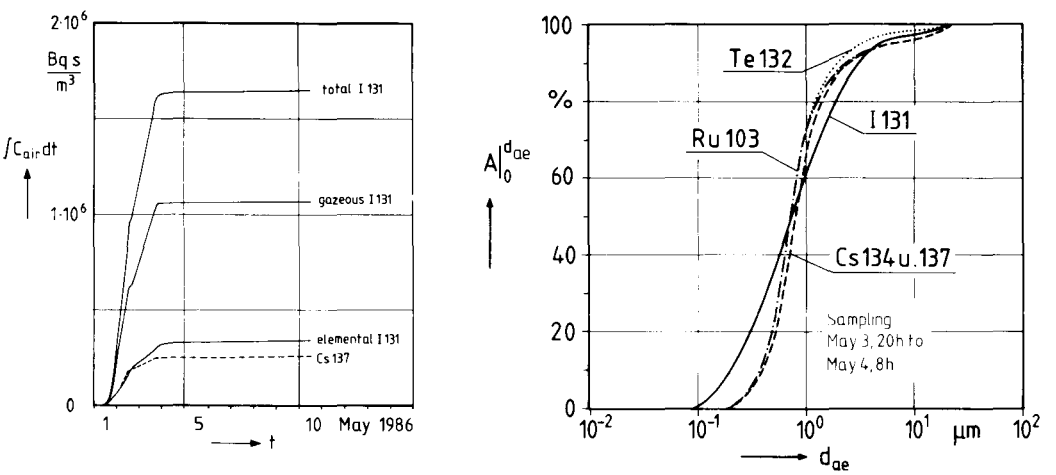

Fig.1: Time-integrated activity concentration of Cs 137 and I 131 and activity size distribution for some particlebound radionuclides in the air at Aachen /1/.

long grass yields to strong variations in the deposition per unit ground area (points in brackets in Fig.2). The strong variation of the deposition of Cs 137 per unit ground area after the rain is mainly caused by the inhomogeneous distribution of the particles on the grass and the small sampling areas. The variation of the specific activity is much smaller /1/. In contrary to the dry deposition of particlebound nuclides, the concentration of I 131 on grass showed two peaks before the rainfall (right hand side of Fig. 2). This peaks can be explained with desorption of that part of the gazeous iodine, which was not taken up into the plants' tissue. The desorption after sunrise could have been caused by the evaporation of the intense overnight dewfall. The extreme decrease of the iodine concentration in the air in the afternoon of May 2 (after the passage of the maximum concentration) could have influenced the desorption too. In addition to the deposition measurements on grass, a very early deposition on stinging nettles is shown in Fig.2, too.

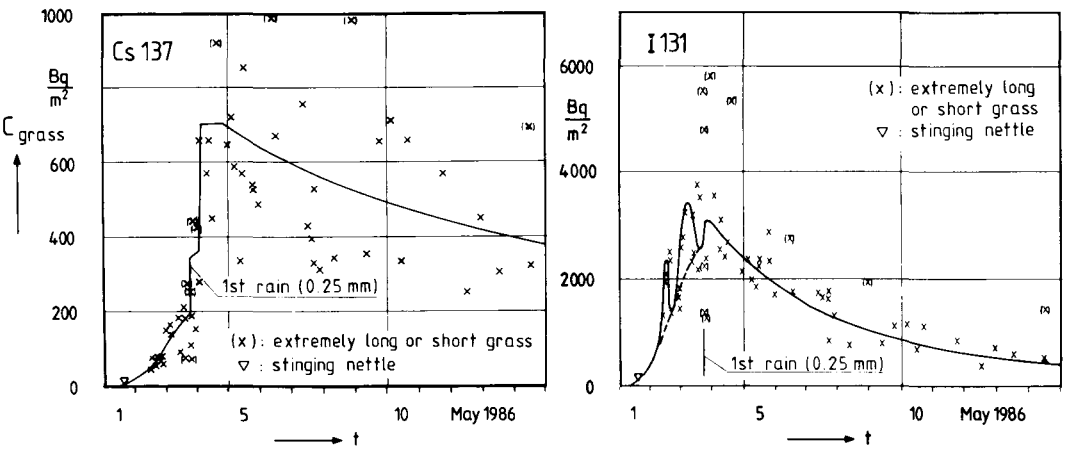

Fig. 2: Measured depositions per unit ground area for Cs 137 and I 131 on grass at Aachen, specific activity see /1/.

The measurements allow the calculation of the deposition velocity by dividing the deposited activity per unit ground area (from Fig. 2) by the time integrated activity concentration in the air (from Fig. 1). The result of this calculation -for the single measurements as well as for the fitted curves- is shown in Fig. 3. In the case of Cs 137, the deposition velocities range from 0.025 to 0.11 cm/s, including the deposition on extremely long and short grass. The mean is approximately 0.07 cm/s. As the separation of deposited particlebound iodine from gazeous species was not possible, the measurements only allow the derivation of a deposition velocity for the total amount of iodine. Ignoring the peaks in Fig. 2, the deposition velocity is around 0.15 cm/s (crosses and dashed curve in Fig. 3). If the peaks of the iodine deposition are included, the deposition velocities range from 0.13 to 0.28 cm/s. In order to predict the dry deposition, models have been developed during the recent years /4/. These models take account of the activity size distribution of particlebound radionuclides, the wind speed and the kind of vegetation. On the

right hand side of Fig. 3, the deposition velocity for grass resulting from the model is shown for three different wind speeds. Additionally, the means and ranges of published experimental data /4/ are shown. With the aid of the measured activity size-distributions (Fig. 1) and the wind speeds during the period of dry deposition, deposition velocities can be calculated. As the monitored wind speed ranged from 2 to 4 m/s, the calculated deposition velocity for Cs 137 lies between 0.03 and 0.07 cm/s for grass of 10 cm height. This corresponds very well with the

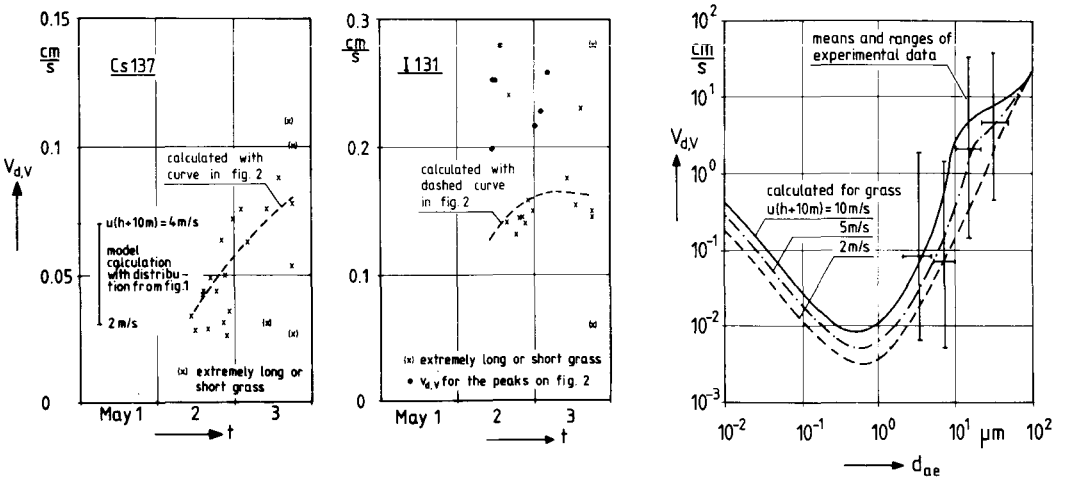

Fig. 3: Deposition Velocities for Cs 137 and I 131 on grass as results of the measurements and model calculations /4/.

deposition velocities derived from the post-Chernobyl measurements. Model calculations with the activity size-distribution of particlebound I 131 from Fig. 1 yield to nearly the same deposition velocity for particlebound I 131 as for Cs 137. With a relationship between particlebound, elemental and organical iodine of 35:20:45 /2/, /3/, thus the deposition velocity for elemental iodine can be estimated. It turns out to be approximately 0.7 cm/s (without the peeks of grass contamination). Considering all measured grass contaminations, the deposition velocity for elemental iodine ranges from 0.5 to 1.2 cm/s. The specific activity of I 131 on leaves was nearly the same and the specific activity of the particlebound radionuclides nearly two times higher than the respective specific activity on grass on May 2.

WET DEPOSITION OF PARTICLEBOUND NUCLIDES AND RADIOIODINE

During the first rainfalls, the activity concentration in the rainwater was measured as a function of the rainfall intensity. As there was no information about the distribution of the radionuclides with height, a rainout coefficient can only be estimated. The washout was small compared with the rainout. Assuming a homogenous distribution from ground level to the top of the clouds and taking account of the extension of the cloud layer (from 1400 to 12000 m; reported from meteorologists for the first rainfall on 3rd of May in Aachen) leads to the x-values

plotted in Fig. 4. These x-values correspond to the rainout coefficient if the height depending activity concentration is constant. They are within the usually expected range. Mean washout coefficients $\bar{\Lambda}(I)$ can be calculated with the aid of the measured activity size-distributions and the modelled $\Lambda(d,I)$ shown in Fig. 4. For the first rain, these mean washout coefficients have nearly the same proportionality to the rainfall intensity as the estimated x-values for rainout. From this fact, it can be concluded that the rainout was dominated by incloud scavenging processes while nucleation was not of importance. For the second rainfall, the proportionality shifted to a smaller exponent (a≈0.56). This can either be the result of a change in particle size distribution or a greater importance of nucleation.

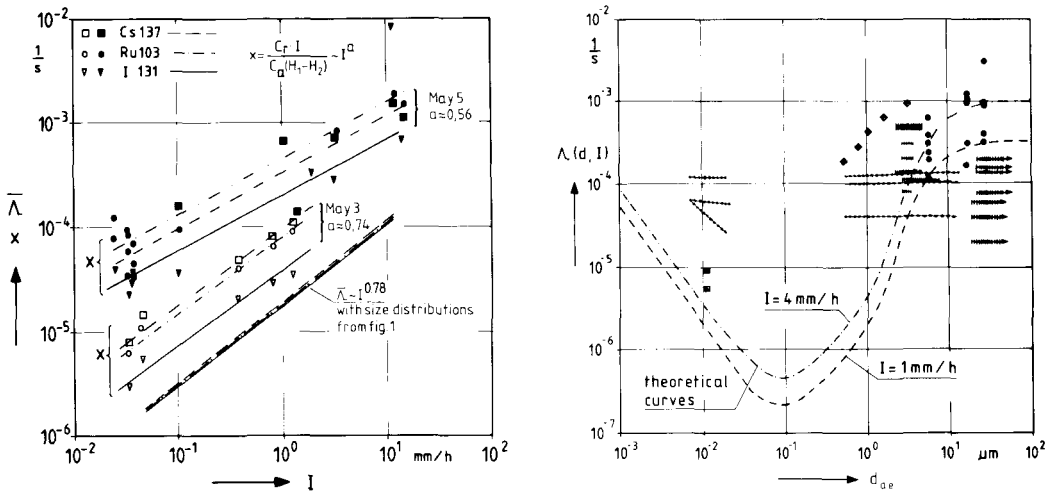

Fig. 4: Concentration of Cs 137 and I 131 in rainwater at Aachen, compared with results of model calculations and published experimental data /4/.

The measurements were sponsored by the Bundesminister für Umwelt, Naturschutz und Reaktorsicherheit under the number St. Sch. 920.

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RELEVANT NUCLIDES AND EXPOSURE PATHWAYS FOR THE RADIATION EXPOSURE IN THE VICINITY OF THE REPROCESSING PLANT AT WACKERSDORF, FEDERAL REPUBLIC OF GERMANY

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1. The Reprocessing Plant

The Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen mbH (DWK) [German Nuclear Fuel Reprocessing Company] plans the construction and operation of a plant for reprocessing spent mixed oxide fuel elements, having a mean daily throughput of 2 t of nuclear fuel. Besides the infrastructure facilities the plant will consist of the following parts:

- fuel element receipt store
- reprocessing and waste treatment plant
- waste storage facility

The main design parameters for the reprocessing plant are shown in Table 1. Table 2 contains the proposed upper limits for the annual radioactive discharges in liquid effluents and exhaust air.

The first part-license for constructing the plant was given by the licensing-authority on 24.09.1985, and the plant has been under construction since 11.12.1985.

Active commissioning of the fuel reprocessing plant is planned for 1997.

The site for the reprocessing plant is located near the village of Wackersdorf in Bavaria, approximately 150 km North-East of Munich and about 80 km East of Nürnberg.

2. The Radiation Exposure in the Vicinity of the Wackersdorf Reprocessing Plant (WAW)

2.1 Relevant Nuclides

In the spent fuel elements is a great number (approximately 150) of radionuclides (fission products, transuranics) with different radiotoxicity and radioactivity.

In view of the radiation exposure in the vicinity of the WAW however, only 14 nuclides are of importance. Considering the construction of the plant and applying the exposure model published in the ICRP-publication Nr. 26, the following nuclides contribute about 99 % of the total radiation exposure:

H-3, C-14, Kr-85, Sr-90, Ru-106, I-129, Cs-134, Cs-137, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-244.

2.2 Calculation of the Reference-Radiation Exposure (discharges in exhaust air)

The calculation of the radiation exposure is based on a calculation procedure, which is obligatory in the Federal Republic of Germany:

General Principles of Calculation of the Radiation Exposure Resulting from Radioactive Effluents in Exhaust Air and in Surface Waters
(Guideline under Art. 45 of the Radiological Protection Ordinance), published by the Minister of the Interior.

Principal assumptions of the assessment of the radiation exposure are:

- operation of the plant for 50 years
- disregard of non radioactive depletion of nuclides on the soil
- residence in the area of maximum exposure during the whole year
- support with foods exclusively originating from the area worst affected
- neglection of decontamination-effects during the preparation of food
- uncommon quantities of food-consumption

The following table shows the calculated maximum radiation exposure in comparison with the limits according to the Radiation Protection Ordinance in the Federal Republic of Germany for the organs Whole Body, Bone and Thyroid:

Exposure / Organ	Calculated [µSv/y]	Limit according to Rad.Prot.Law [µSv/y]
Whole Body	29	300
Bone	201	1 800
Thyroid	160	900

2.3 Relevant Exposure Pathways

Figures 1 and 2 demonstrate the relative percentage of single exposure pathways in the total exposure for bone and the thyroid. 85 % of the total exposure of the bone are caused by the nuclide Sr-90 and 92 % of the total exposure of the thyroid are caused by the nuclide I-129.

3. Discussion of the Calculated Reference Radiation Exposure

The calculation of the reference radiation exposure is based on the "obligatiory procedure" to be certain, that the exposure in reality will always be less than the calculated values.

Is that also true for the site of the reprocessing plant at

Wackersdorf, too?

This question will be answered by the example of the thyroid exposure with I-129.

In reality, each parameter used in the calculation shows a great variation. This is illustrated by the deposition-velocity of Iodine on vegetation (Figure 3) and the milk-consumption of the critical population group (farmers) (Figure 4).

The great variability of the model parameters was taken into consideration by a statistical model. For the following parameters the site-specific variability was taken into account: Meteorological conditions, fall-out, wash-out, transfer factor soil-plant and food consumption. The correlation between single parameters was also taken into account.
The result of this statistical analysis is shown in Figure 5. It demonstrates, that the reference radiation exposure close to the site with great probability (99,9 %) includes the expected "real" exposure . Consequently the application of the "obligatory procedure" leads to an overestimation of the "real" exposure.

Table 1: Main Design Parameters for the Reprocessing Plant

Mean burn up	40.000 MWd/t
Maximum burn-up for all FE for individual FE	55.000 MWd/t
Coding period of FE before reprocessing	7 years
Mean daily throughput of the plant	2 t of nuclear fuel
Maximum yearly through-put of the plant	500 t of nuclear fuel

Table 2: Proposed Upper Limits for the Annual Radioactive Discharges

Exhaust Air Nuclide	Effluents [Bq/a]	Liquid Effluents Nuclide	Effluents [Bq/a]
H-3	1,5 E15	H-3	3,7 E13
C-14	1,3 E13		
Kr-85	1,6 E17		
I-129	1,85 E9	I-129	1,0 E8
Beta-Aerosols	5,4 E10	Beta-Nuclides without H-3	1,33 E10
Sr-90	7,9 E9	Sr-90	1,7 E9
Cs-137	1,3 E10	Cs-137	2,6 E9
Alpha-Aerosols	1,4 E9	Alpha-Nuclides	4,44 E8

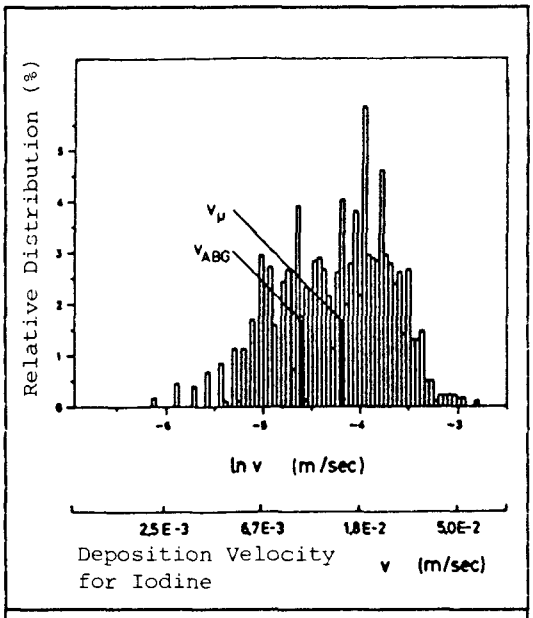

Fig.3: Relative Distribution of the Deposition Velocity for Iodine on Vegetation
($V\mu$= Mean Value,
V_{ABG}= Obligatory model parameter)

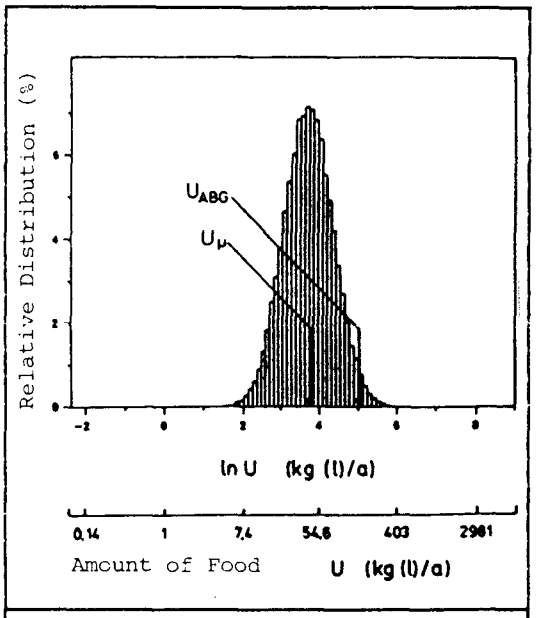

Fig.4: Relative Distribution of the Food Consumption of the critical population group
($U\mu$= Mean Value,
U_{ABG}= Obligatory model parameter)

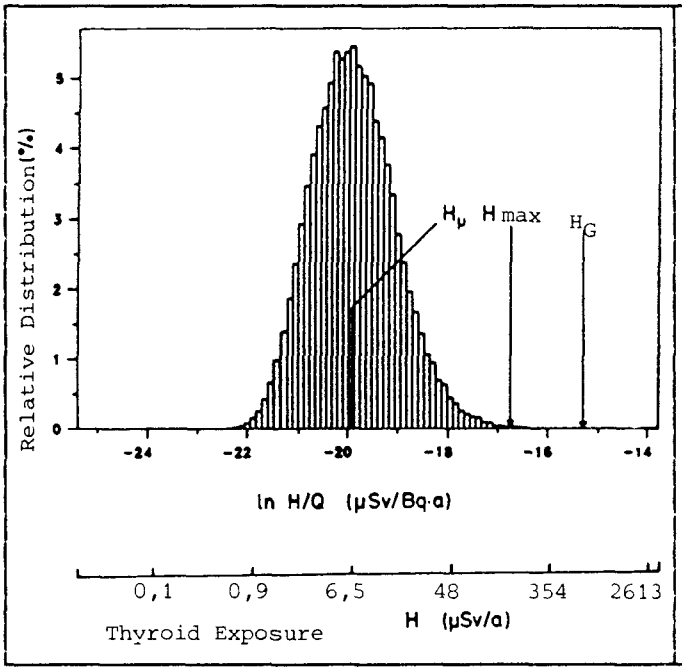

Fig.5: Relative Distribution of the "Real" Thyroid Exposure

($H\mu$= Mean Value,
Hmax= Maximum Exposure, calculated with the obligatory procedure,
H_G= Exposure Limit according to Rad. Prot. Law

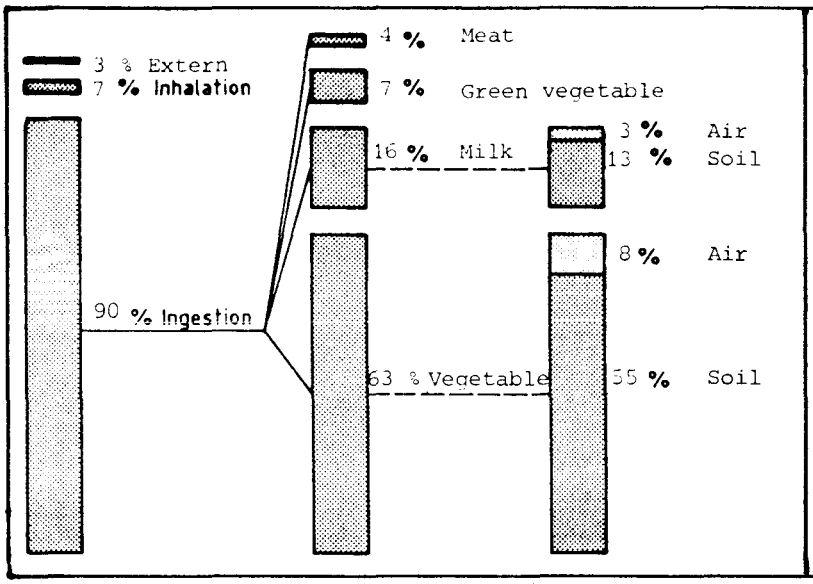

Figure 1:

Relative Percentages of Single Exposure Pathways for Bone

(Air = Activity Intake from Air
Soil= from Soil

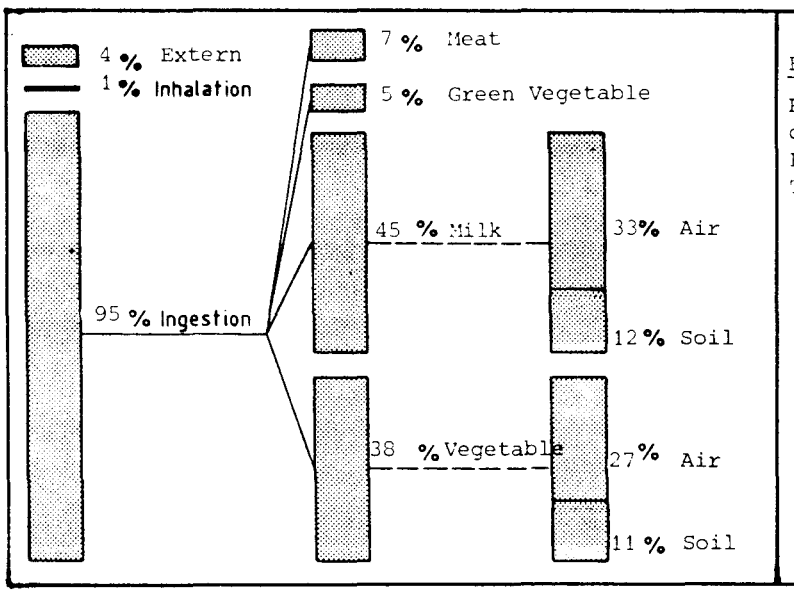

Figure 2:

Relative Percentages of Single Exposure Pathways for the Thyroid

(Air = Activity Intake from Air;
Soil= from Soil)

SIMULATION MODEL PREDICTING CONTAMINANT SPREAD DUE TO RADIONUCLIDES RELEASED FROM NUCLEAR POWER PLANTS

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INTRODUCTION

Presently there are 12 units of nuclear power plants in operation (1-BWR,10-PWRs,1-ATR), 3 units under construction (2-PWRs,1-FBR) in Fukui Prefecture located near the center of the Japanese Archipelago.Environmental radioactivity and radiation surrounding these plants has been studied and it has been confirmed that the released radioactive waste is not detrimental to the environment.

However, we need to know the future contaminant spread in order to judge whether it is necessary to make any changes in regulations. While it is not possible to evaluate the future contaminant spread by means of environmental monitoring alone.

Therefore,a three-dimensional finite model(Fukui Prefectural Model) for evaluating the contaminant spread, has been developed.

PRINCIPLE OF MODEL

The purpose of this research is to investigate the behavior of radionuclides,especially CO-60,which has been mainly released from nuclear power plants and validate the applicability of numerical model.

This model includes:

(1) Transport mechanisms
- ◆ advection and dispersion of radionuclides due to water movement
- ◆ deposition of radionuclides accompaning with settlement of suspended matters
- ◆ vertical diffusion of radionuclides in the interstitial water in sediment

(2) Radionuclide decay term

(3) Adsorption/desorption with sediment

Contaminant movement in coastal area can be expressed by the diffusion equation derived from mass conservation equation.

Assuming the sorption phenomenon consists of the reversible adsorption on sediment grain and the irreversible adsorption sink to sediment,then the sorption model combined with a linear isotherm and a first order kinetics can be used:

$$\frac{\partial Q}{\partial t} = K\frac{\partial C}{\partial t} + k \cdot C$$

where,

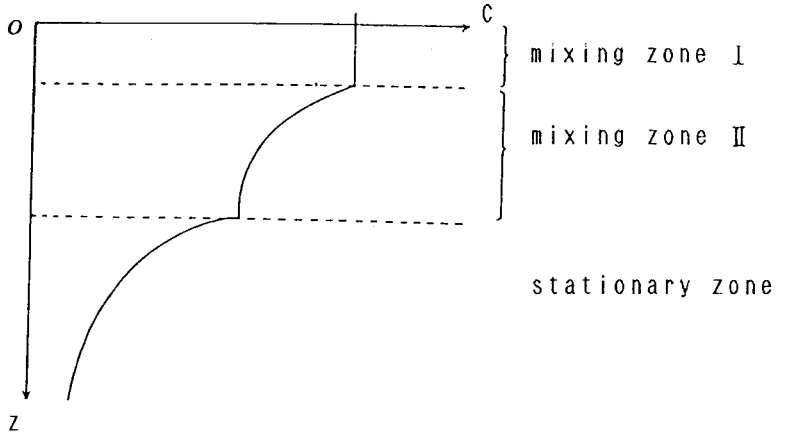
Q:the concentration of the radionuclide in the solid phase[Ci/g]
C:the concentration of the radionuclide in the liquid phase[Ci/g]
K:The distribution coefficent[cm^3/g]
k:The rate coefficent of the first order kinetiic [cm^3/g・s]

We measured the K value by the batch experiment,though it depends on many factors such as clay minerals in sediment and the temperature of sea water.

It is clear through environmental monitoring that coastal

sediment and interstitial water in top layer are continuously stirred with wave movement or the activities of living things. So the sediment layer was divided into a mixing zone and a stationary zone. Moreover, the mixing zone was divided into zone I and II which has more moderate mixing effect. The concentration in the interstitial water in the mixing zone I is assumed to be equivalent to that in sea water.

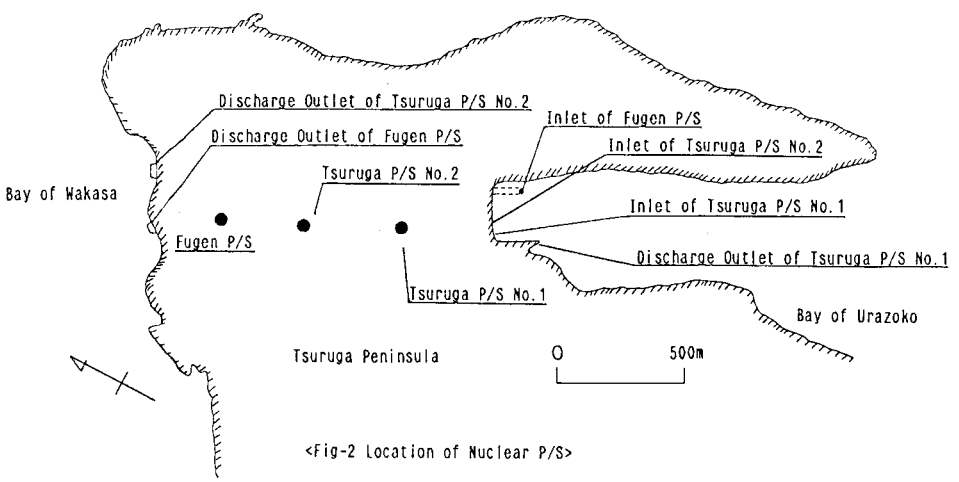
< Fig-1 concept of coastal sediment >



C:concentration in the interstitial water
Z:depth

VALIDITY AND APPLICATION

To evaluate the propriety of this model and predict the behavior of radionuclides till 2000, the contaminant spread of Co-60 resulting from the nucler power plant was calculated in the



<Fig-2 Location of Nuclear P/S>

area of the Bay of Urazoko(1.5km× 700m)in front of Tsuruga Power Station (BWR;357MW) . The Co-60 concentrations of about 0.02～ 1 pCi/g ・dry-sediment have been detected in the bay where the radioactivity was released so enormously at the beginning of the plant operation.

The model constitution and parameter used are as follows,

○ Sea water current

The water current is composed of a tidal current,an ocean current and a wind driven current,etc.These currnts are important for predicting the contaminant spread in sea water.

Material transport in sea water should be sorted out according to the duration considerd.

Fukui Prefectural Model treats long-term prediction of spread, so short-term phenomena such as a periodic current and a turbulent are neglected.Then the flow pattern in a steady state was simulated at first in this model as follows.

(1) The current directions are observed periodically in the surveying sea area and categorized into patterns with considering simultaneity.
(2) Then the frequency of the patterns is measured.
(3) For patterns occurring at a comparatively high frequency,the average flow vectors are calculated.
(4) The flow patterns achieved by the above are treated as the steady flow of the subject sea area and simulated by using a three-dimensional steady flow model.

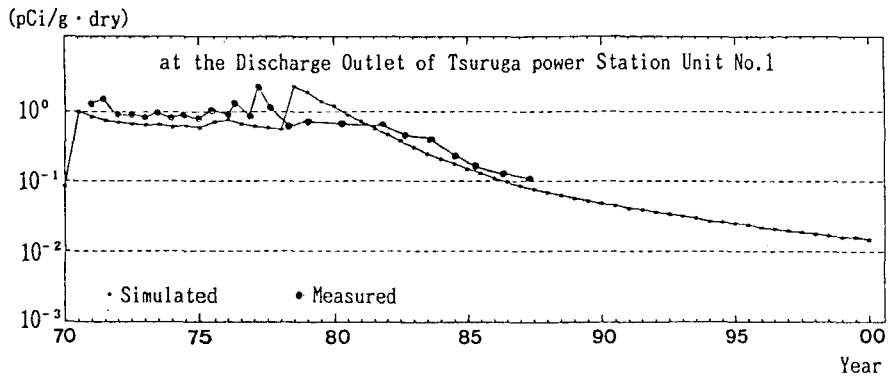
The grid interval is 50m × 50m or 200m× 200m and the differential time is 20 seconds for the finite difference method.

○ Distribution coefficent

The K is measured for several kinds of coastal sediments at 5 ℃, 15℃, 28℃.The degree of adsorption K is 120～ 19000 and desorption K of 2000 ～ 34000 (cm^3/g).

The concentrations of Co-60 both in sea water and in sediment were calculated using the super computer CRAY 1.

＜Fig-3 Concentration of Co-60 in the Sediment＞



CONCLUSION

(1) The results estimated by the finite difference calculation well coincide those of sea bottom measured by a Ge(Li) semi-conductor.

So this model seems to be useful to evaluate and predict the behavior of released liquid radionuclides.

(2) The average concentration of Co-60 in mixing zone has decreased simply since 1979. The half decreasing rate is about five years that is almost the same as the half liffe of Co-60. So it shows apparently that the released Co-60 is not newly adsorbed to coastal sediment. The concentration will be reduced to one-tenth of the present concentration in 2000.

The reasons are mainly,

I the released Co-60 from Tsuruga Power Station unit No.1 has decreased since 1978.

II Tsuruga Power Station unit No.2 and Fugen Power Station have been intaking the circulating water of about 90 m^3/s from the Bay of Urazoko and discharging it to the Bay of Wakasa. The former station has been doing so since 1986 and the latter since 1978. So the desorption from sediments has been promoted in the Bay of Urazoko by replacing sea water.

(3) The contaminated sediment near the discharge outlet of Tsuruga Power Station unit No.1 has been moving to the mouth of the Bay of Urazoko. But the migration of sediment itself is not considerd in this model, so the calculated concentrations are a little bit less than those measured near the discharged point.

(4) It became evident that about 25% of the total released Co-60 was accumulated in the Bay of Urazoko in 1978. But it will reduce to only 5% in 2000.

Fukui Prefectural Committee on Nuclear Safety

Morinobu Kokubu(Japan Atomic Energy Research Institute), Toshinori Iijima(Japan Atomic Energy Research Institute), Masami Fukui(Kyoto University) and Shigeo Uchida(National Institute of Radiological Sciences)

DISPERSION AND RESUSPENSION FACTORS OF RADIOACTIVE DUSTS DERIVED FROM AIR MONITORING DATA IN JAERI

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Jun'ichi ONODERA, Yutaka ANAZAWA, Hozumu YAMAMOTO and
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INTRODUCTION

Dispersion factor of radioactive dust is necessary for designing an exhaustive air cleaning system and for improving procedures of operation. Resuspension factor of radioactive particles deposited is also necessary for estimating the degree of radioactive air contamination due to the resuspension of surface contaminants during the maintenance or the decontamination of contaminated provisions and floors. This paper describes the dispersion and the resuspension factors derived from radiation monitoring data in radiation works which have been made in JAERI.

DISPERSION FACTOR

A dispersion factor, E, of dusts is determined by the following equation (1), and related to the radioactivity released to the atomsphere, Q, from an unsealed radioactivity handling facility.

$$E = D/Q_o \quad (1)$$

$$Q = Q_o E(1-F_d)(1-\eta) \quad (2)$$

where D is the radioactivity dispersed in air, Q_o the total radioactivity contained in the treated part, F_d the deposition fraction of particles to the duct and other components, and η the filtration efficiency of an air cleaning system.

a) Dispersion factor of fission product dust

In the cells of Hot Laboratory irradiated fuels of nuclear reactors or irradiated experimental materials are cut and polished for their post-irradiation test or metallurgical examination. The dispersion factors of each nuclide contained in fuels were measured by Yamamoto et al. [1] and Izumi et al. [2]. They were measured in the cutting of irradiated metal uranium fuel rods with burn-up from 390 to 600 MWD/T and cooling time from a few month to a year [1], and with another conditions of burn-up from 2800 to 5200 MWD/T and cooling time from 300 to 500 days [2]. The cutting was conducted pouring a cooling oil into the cutting part (wet method), and the ventilation rate in the cells was 40 times/h.

Table 1 shows the geometric mean, E_g, and the geometric standard deviation, Σ_g, of the dispersion factors. The recommended values for the design of air cleaning systems are shown in

the table comparing both data and considering the accuracy of measurement.

Table 1 Dispersion factor of fission product dust

Nuclide	Yamamoto et al's[1]		Izumi et al's[2]		Recommended	
	E_g	Σ_g	E_g	Σ_g	E_g	Σ_g
I-131	3.3 E-03	3.3			4 E-02	4
Cs-137	1.1 E-03	2.8	2.0 E-03	3.6	2 E-03	4
Ru-103	5.5 E-04	2.4			6 E-04	3
Ba-La-140	5.5 E-05	9.1			2 E-04	4
Ce-141,144	3.9 E-05	2.9	2.5 E-04	2.1	4 E-04	3
Zn-Nb-95	5.7 E-05	11			2 E-04	4
Sb-125	1.0 E-03	4.7	3.5 E-02	2.1	4 E-02	3
Pu-239	9.9 E-05	5.2			2 E-04	4

b) Dispersion factor of the radioactive dust produced in cutting of pipes

Some stainless steel pipes in the primary cooling systems of Japan Power Demonstration Reactor were cut in the reactor dismantlement. The inner surface of pipes was uniformly contaminated at a density of 1.9×10^5 Bq/cm^2. The dispersion factors in the cutting and the particle size distributions were measured [3]. The cutting was carried out by using a band saw in the simple containment house made of PVC sheet, where the air was ventilated at a rate of 8 times/min. The main nuclide was Co-60 which was a composition of crud deposited inside the pipes.

The geometric mean, E_g, and the geometric standard deviation Σ_g, of the dispersion factors measured are

$$E_g = 1.9 \times 10^{-1}, \qquad \Sigma_g = 1.7$$

and the size distributions of dispersed particles are

$$d_g = 6.8 \sim 8.0\ \mu m, \qquad \Sigma_g = 1.7 \sim 1.9$$

The dispersion factor for the pipe cutting [3] is from one to two order of magnitude higher than those for the fuel cutting [1], [2]. This seems to be attributed primarily to the difference between dry method and wet method.

c) Dispersion factor of the radioactive dust from a compactor for radioactive solid waste

Low level incombustible radioactive solid waste which was generated in reactor buildings, nuclear material handling facilities and chemical laboratories were reduced in volume by a compactor, canned into 200 liter drums and stored in a storage building. Radioactive dusts in the pressing were dispersed into treatment room from the wastes. The dispersion factors in these operations were measured [4]. The ventilation rate was 14 times/min.

The geometric mean, E_g, and the geometric standard deviation, Σ_g, of the dispersion factors measured are

$E_g = 5.3 \times 10^{-5}$, $\Sigma_g = 4.1$

This dispersion factor is different in nature from those for the cutting of fuels or pipes, because of the completely different generation mechanisms.

RESUSPENSION FACTOR

A resuspension factor, K, of radioactive particles deposited is obtained by the following equation:

$$K = C/S$$

where C is the air concentration of radioactive dust particles and S the surface contamination density.

a) Resuspension factor related to the decontamination of hot cells

One of measurements of resuspension factor was made at the No.6 cell of Reactor Fuel Examination Facility [5]. In the cell various irradiated fuels from nuclear power reactors are cut and polished for the metallurgical examination, and thus the floor and the wall are extremely contaminated with the dispersed radioactive dusts. For the periodical inspection of the facility and the interior provisions the cell is decontaminated. In the decontamination the radioactive particles deposited on the floor and the wall are blown up into the air to cause the air contamination. The ventilation rate in the cell is 50 times/h.

The resuspension factors were measured in relation to the type of work: decontamination, transfer of provisions, radiation survey, and no operation. Figure 1 shows the plot of measured values of the factors onto the logarithmic probability paper. As seen in the figure, the geometric mean resuspension factor without work is $K_g = 2.3 \times 10^{-8}$ cm^{-1} ($\Sigma_g = 4$) and is fairly close to Chamberlain et al.'s measurements [6]. The mean factor in decontamination of the floor and the wall is $K_g = 1.8 \times 10^{-6}$ cm^{-1} ($\Sigma_g = 3$), and is about two order of magnitude higher than that without operation. In transfer of provisions or radiation survey the factor is between those in decontamination and without operation. This is because the resuspension of deposited particles is based on workers' movement and the contact surface area between the workers and the floor is fairly smaller than that in decontamination.

b) Resuspension factor in repair of plutonium handling facility

The break of negative pressure in a glove box of plutonium handling facility happened to result in the serious contamination of the floor of room with plutonium powder (PuO_2). The resuspension factors of the plutonium powder in the temporary mending work were measured for estimation of the air contamination produced in the further repair work and the selection of respirators to be used [7].

Two workers wearing self-contained air ventilated blouses entered the simple containment house made of PVC sheet to seal

leakage of the glove box. The volume and the contaminated surface in the house were 6 m^3 and 2 m^2, respectively, and the ventilation rate is 10 times/h. The floor was almost uniformly contaminated at a density of 1.7×10^3 Bq/cm^2. The results are

$$K = 4 \times 10^{-8} \sim 2 \times 10^{-7}$$

and the size distribution of resuspended particles is

$$d_g = 6.4 \sim 26\ \mu m, \qquad \sigma_g = 2.3 \sim 2.7$$

This operation is similar to transfer of provisions or radiation survey in the hot cell stated in a) of this chapter and therefore the resuspension factors are close to each other.

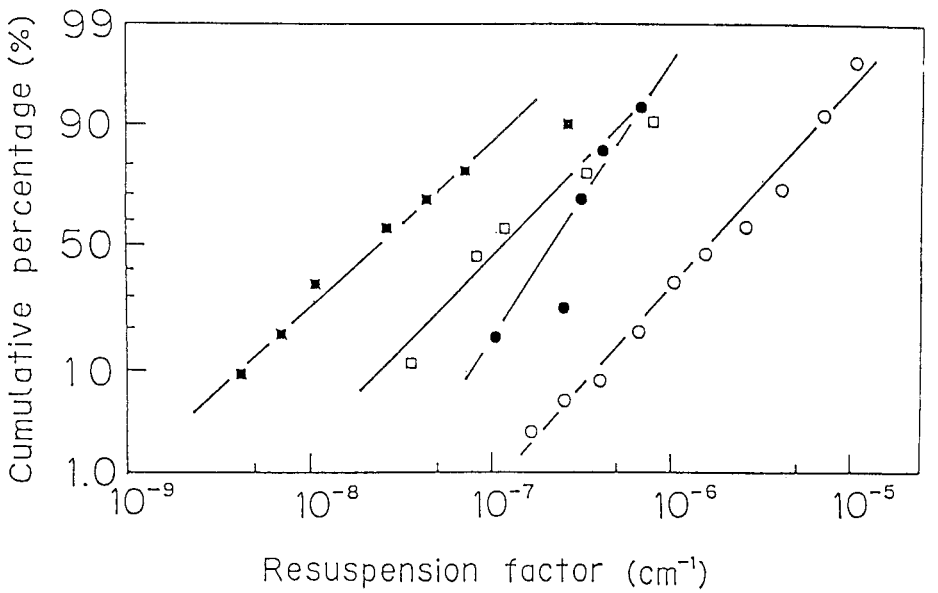

Fig. 1 Distribution of resuspension factors for different conditions. ○ decontamination of floor and wall, ● transfer of provisions, □ radiation survey, ■ no operation.

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TEST OF EXISTING MODELS ON THE LONG-TERM RADIOACTIVE CONTAMINATION OF FOODSTUFFS THROUGH FIELD MEASUREMENTS ON WHEAT CROPS

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The processes of uptake through plant roots and deposition of resuspended material onto vegetation have longly been recognised as the main routes of the transfer of radioactive contamination from soil to vegetables. It is well known/1-2/ that the predictions of existing models of such processes are strongly affected by uncertainties related to the complexity of several chemical and biological processes which rule the transfer. The radioactive contamination caused by the Chernobyl plume is a chance of testing the existing models on the dynamics of radionuclides in the environment in order to improve the assessment of the environmental consequences of nuclear accidents.

The Italian National Institute of Health (ISS), with the collaboration of the Istituto Sperimentale per la Patologia Vegetale (ISPV), has planned a research devoted to the study of the transfer of caesium from agricultural soils to wheat crops. The choice of wheat has been motivated by its relevance in the Italian diet (about 110 kg/y for adults/3/).

The main features of the experimental methodology adopted are: i) selection of soil plots for which it is possibile to get information about meteorological conditions and ordinary culture practices; ii) sampling of soil in order to determine the distributions of Cs-134 and Cs-137 at various depths; iii) chemical analysis of soil in order to measure the acidity, the content of clay and organic matter, the concentrations of exchangeable calcium and potassium; iv) sampling of wheat plants during their growth up to the harvest; v) determination of the specific activities of Cs-134, Cs-137 and K-40 in various components of the wheat (namely, straw, chaff, bran, shorts, red-dog and flour) in order to study the translocation of caesium and potassium inside the plant, in particular inside the grain.

The gamma spectrometry of all the samples has been performed with an intrinsic Ge (resolution 1.95 keV, efficiency 38%).

Two soil plots, belonging to the ISPV, have been selected near Rome (labelled hereafter as soil I and II). In Table 1 the values of the specific activities of Cs-134 and Cs-137, measured in samples taken in July 1986 after the harvest, are reported.

In ref./4/ it has been shown that the observed distribution of Cs-137 in the selected soil plots arises from two different contributions, namely: i) a uniform component due to the fallout relative to the nuclear weapon tests in the sixties; ii) an exponential undisturbed component due to the fallout of the Chernobyl plume. It turned out that/4/ the total deposition in 1986 (vegetation + soil) can be estimated to be equal to ~ 400 Bq/m^2 in both soil plots.

In Fig. 1 the Cs-137 specific activity measured in 5-cm thick samples of soil I taken in July 1987 is shown. It can be seen that

Table 1. Specific activities of Cs-134 and Cs-137,in Bq/kg, of soil samples taken in July 1986 after the harvest (errors with a 95% confidence level).

depth (cm)	soil I		soil II	
	Cs-134	Cs-137	Cs-134	Cs-137
0 - 2	4.8 ± .4	18.7 ± .5	3.4 ± .5	16.4 ± .7
2 - 4	1.1 ± .5	11.4 ± .5	1.4 ± .3	11.3 ± .5
4 - 6	1.0 ± .3	9.9 ± .5	< .3	9.1 ± .7
6 - 10	< .2	9.2 ± .5	< .2	8.6 ± .5
10 - 14	< .2	9.2 ± .5	< .2	8.0 ± .6
14 - 20	< .2	9.5 ± .5	< .2	8.4 ± .6

after one ploughing the distribution of Cs-137 is not completely uniform, whereas the drop after 45 cm is clearly connected to the depth of the disturbed soil.

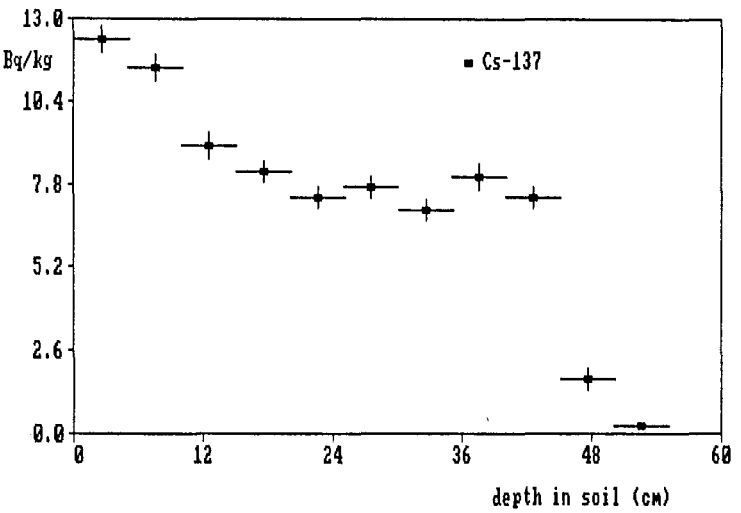

Fig. 1. Distribution of Cs-137 in samples of soil I taken in July 1987 after the harvest (errors with a 95% confidence level).

In Table 2 the results of the chemical analyses of samples of soil I and II are reported. It should be pointed out that both the concentrations of exchangeable cations and the content of organic matter are different in the selected soil plots. This difference is expected to play a role in determining the caesium and potassium content of wheat plants in 1987.

After the harvest in July 1986 the specific activities of Cs-134, Cs-137 and K-40 have been measured in several samples of various components of the wheat plants. The results of the gamma spectrometry for the varieties CRESO (durum wheat) and MEC (tender wheat) are shown in Table 3.

It can be seen that the distributions of Cs-134, Cs-137 and K-40 are not uniform within the wheat plant, as it is known also for other non radioactive elements. In particular, the specific

Table 2. Results of the chemical analyses of the selected soil plots. The acidity, the percentage (in weight) of clay, slime, sand and organic matter, the concentrations of some exchangeable cations are reported.

	pH	% clay	% slime	% sand	% organic matter
Soil I	7.8	33	35	32	2.8
Soil II	6.3	37	17	46	1.5

	Exchangeable cations (ppm)					
	K	Na	Sr	Ca	Mg	Rb
Soil I	537	31	37	21	222	1.4
Soil II	465	116	106	36	690	13

Table 3. Specific activities of Cs-134, Cs-137 and K-40,in Bq/kg (fresh weight) measured in samples of various components of the varieties CRESO (durum wheat) and MEC (tender wheat) harvested in July 1986 (errors with a 95% confidence level).

Component	CRESO (durum)			MEC (tender)		
	Cs-134	Cs-137	K-40	Cs-134	Cs-137	K-40
straw	49±2	97±3	231± 50	38±7	85±8	336±123
chaff	42±5	93±7	204±101	83±6	174±8	< 125
bran	54±2	114±2	335± 37	69±4	135±5	423± 61
shorts	27±2	54±2	186± 26	37±2	76±3	315± 36
red-dog	14±1	30±1	78± 15	14±1	32±2	69± 24
flour	16±1	34±1	112± 15	8±1	16±1	< 35
grain	25±1	50±2	129± 19	20±1	40±2	107± 18

activities are higher in straw and chaff as well as in the external components of the grain (namely, bran and shorts). The ratio between the Cs-137 specific activity observed in the grain and the estimated total deposition on soil turns out to be equal to ~ 0.1 m^2/kg. This value is just twice the value predicted by the compartimental models FOOD-MARC/5/ and ECOSYS/6/. However, in our opinion such a discrepancy should be traced to a non precise knowledge of some parameters appearing in the mentioned models, like the dependence of the initial interception factor on the growth stage.

In 1987 the sampling of four different varieties of wheat has been carried out periodically from April to the harvest in July. The samples have been dried at 130 °C for 24 h before measuring the activities. The ratio between the dry and the fresh weight of the samples varies with the plant growth going from 20% in April up to 80% - 90% at harvest. The specific activity of Cs-134 was below the sensitivity threshold (about 0.4 Bq/kg) in all the samples. The Cs-137 specific activity measured in samples of wheat plants collected in April 1987 was within the range 1 - 2 Bq/kg (d.w.). After the ear emergence (occurred approximately at the end of April) the specific activity of Cs-137 dropped below 1 Bq/kg (d.w.); measurements with better sensitivity are in progress. In

Table 4 the values of the K-40 specific activity measured in the collected samples of wheat plants are shown.

Table 4. Specific activity of K-40 (in Bq/kg d.w.) measured in samples of the varieties CRESO (durum), MEC (tender), LATINO (durum) and ANIENE (tender) collected from April to June 1987 (errors with a 95% confidence level).

sampling date	CRESO	MEC	LATINO	ANIENE
2/4/87	803 ± 19	881 ± 23	960 ± 24	901 ± 19
15/4/87	657 ± 17	---	1173 ± 28	---
15/5/87	673 ± 13	664 ± 14	728 ± 15	785 ± 15
5/6/87	338 ± 7	463 ± 9	486 ± 9	475 ± 15
25/6/87	163 ± 4	286 ± 8	301 ± 7	283 ± 7

It can clearly be seen that after the ear emergence the plant growth drastically reduces the K-40 specific activity. It should be pointed out that the K-40 specific activity found in the last sampling (25/6/87) in the variety CRESO, which growths in soil II, is about one half of the corresponding values relative to the other three varieties, which growth in soil I. In our opinion the final data, including also the time evolution of the Cs-137 specific activity up to the harvest, will allow a meaningful comparison with the corresponding values predicted by the existing dinamical compartimental models on the soil-to-plant transfer. Furthermore, it will be possible to add information on the relationship between soil composition and caesium or potassium levels in wheat crops.

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NATURAL AND ARTIFICIAL LEVELS OF RADIOACTIVITY IN SOIL OF CAMPANIA REGION

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The ground radioactivity is generally ascribed to: i) the natural radiation present in soil and, ii) the artificial radionuclides originated by nuclear weapons tests, accidental releases from nuclear power and industrial plants. The redistribution of natural radioactivity related to agricultural management and the use of fertilizers presenting a high concentration of radioelements (ref.1) must be also considered. The aim of the present work was to investigate the various components of the ground radioactivity in Campania, a region of Italy, including the recent Chernobyl fall out.

Campania (13,600 Km^2) is a south Italy region lying between the Appennines and the Tirreno sea. It consists principally of three areas: the calcareus area of the Appennines, a flat country by the sea containing three volcanic groups (Roccamonfina, Campi Flegrei and Vesuvio) and some alluvial soils, and finally the Cilento high land, rich in karst-formations. Samples of soil (0-5 cm) collected from 19 sites around the region (FIG. 1), in the period May 20, June 5, 1986, were analyzed for gamma-ray activity by using a 20% efficiency and 2,1 KeV resolution (at 1.33 MeV) Ge(Li) detector and a multi channel analyzer. The spectra were analyzed off-line by the automatic code CERNUC (ref.2).

Three major natural gamma radiation sources were identified: ^{40}K, the Uranium and the Thorium decay series. The most intense members of the radioactive families are shown in table I. ^{40}K accounted for more than 50% of the total activity. The sites nearest to the volcanic areas (1,2,3,14 and 15) presented a higher level of radioactivity. In rows 20 and 21 the results of the measurements performed on samples of two fertilizers, mineral superphosphate and potassic sulphate, are reported. The first fertilizer contained a very high concentration of the Uranium family members and some of the elements of the Actinium serie, not detected in the soil samples; the second one presented a ^{40}K activity of 23 kBq/Kg. Nevertheless the use of both fertilizers has

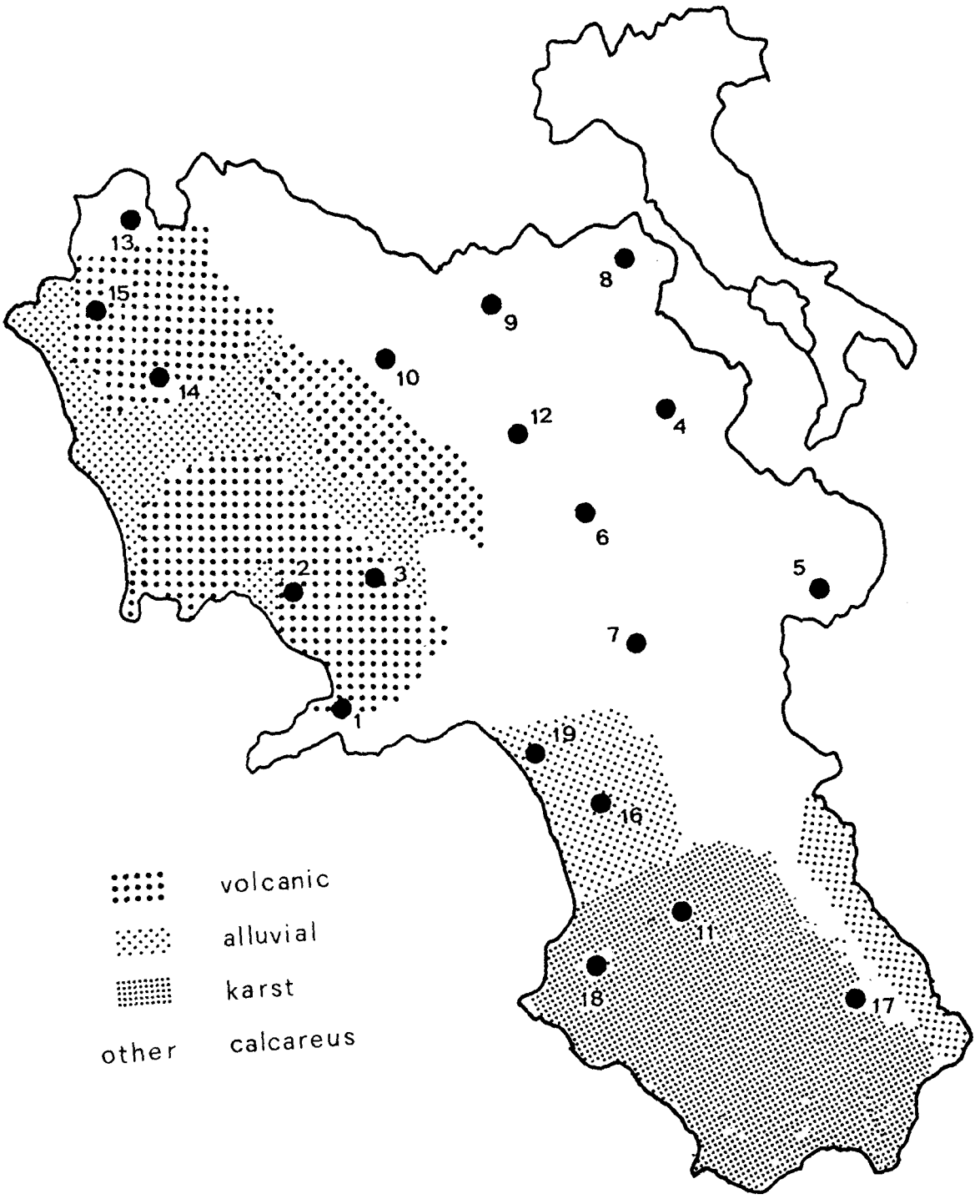

FIG.1 - Campania region and sampling locations

no influence on the soil radioisotopic concentration. In fact, no difference was found between cultivated and uncultivated soils. Therefore local variations of the radionuclide concentrations are to be attributed to geological and geochemical differences.

The measured high values of the Cesium radioisotopes are to be attributed to the Chernobyl accident. Measured radionuclides (ref 3) with short half lives are not reported here since they do not present any long term consequence. The observed dispersion of the values is probably due to the very irregular orographic profile of the Campania region, the local meteorological conditions during the fall out period and the different physical and chemical properties of the soils. The actual mean concentration of ^{137}Cs was found to be 107 Bq/Kg. The average ratio $^{137}Cs/^{134}Cs$ was greater than the estimated ratio (equal to 2) for the activity released from the reactor. This indicates that some amount of ^{137}Cs was already present in the soil, produced by previous atmospheric nuclear tests. As the half life of the ^{134}Cs is 750 d, the presence of this isotope is only due to the Chernobyl release. Assuming $^{137}Cs/^{134}Cs$ = 2 at May 1, 1986 we obtained that 94 Bq/Kg were a consequence of the Chernobyl accident, while the remaining 13 Bq/Kg were pre-existent. Taking into account the depth and the average density of the soil samples (1.387 g/cm^3), we assessed the total ^{137}Cs fall out to be 89 TBq (6.5 KBq/m^2). The uncertainty due to the differences among sample densities and sampling procedures was evaluated around 30%. It is of some interest to calculate the amount of this element in the environment immediately after the experiments of the sixties. An indirect estimation of this concentration may be derived by analysing the content in matrices which have a high efficiency for retention of ^{137}Cs. It is well known that the lichens have this peculiarity (ref.4). For this reason, samples of lichens (Stereocoulon Vesuvianum) were collected at the Vesuvio (altitude 370,490,580,780,960 m) in three different times (October 28 and December 5, 1986 and October 5,1987). Their ^{137}Cs mean contents were 2016±203,1868±240 and 1776±166 Bq/Kg, respectively. As done for the soil samples, we calculated the contribution of the fall out before the May 86 (403 Bq/Kg). Performing a linear least square fit of the logarithm of the ^{137}Cs values versus time, we obtained an effective half life of the isotope in the lichens of 6.1±0.9 y, corresponding to a biological half life of 7.6±1.2 y. Using the measured effective half-life, the ^{137}Cs level in 1970 can be extrapolated obtaining a value of 2600 Bq/Kg.

The ground doses related to the mean concentration of ^{137}Cs and the mean activity of natural radionuclides contained in a layer of 5 cm of top soil were 0.12 mSv and 2.38 mSv, respectively. Although in these estimations we took into account only one of the possible pathways of the dose to the man, the comparison is helpful

for a better understanding of the real impact of the Chernobyl accident.

The authors are grateful to Dr P.Adamo for the sampling of the lichens. This research was partially supported by the regional Government of Campania.

Table I. Radioisotope concentrations (Bq/Kg) at different sites in the Campania region.

#	^{137}Cs	^{134}Cs	^{226}Ra	^{214}Pb	^{214}Bi	^{228}Ac	^{212}Pb	^{212}Bi	^{235}U	^{211}Pb	^{40}K
1	111	40	503	346	294	144	141	118	-	-	2590
2	156	64	488	394	347	148	179	245	-	-	3135
3	234	106	629	588	484	118	84	114	-	-	2801
4	227	110	128	75	78	73	54	93	-	-	1622
5	126	68	61	45	54	68	65	69	-	-	1340
6	69	30	269	202	188	142	159	115	-	-	2498
7	168	76	128	167	159	153	141	95	-	-	2279
8	61	20	54	35	36	51	43	37	-	-	992
9	123	54	144	48	47	78	93	62	-	-	1650
10	53	14	52	51	51	91	88	57	-	-	1289
11	44	19	91	73	41	88	74	41	-	-	1084
12	179	88	273	115	85	145	174	158	-	-	2384
13	51	23	211	74	59	174	164	105	-	-	2198
14	160	78	381	151	116	231	240	150	-	-	2888
15	17	3	211	153	151	206	141	145	-	-	2335
16	168	78	332	106	104	159	100	171	-	-	2042
17	39	9	142	79	76	89	84	60	-	-	1304
18	23	7	81	39	46	38	28	15	-	-	1014
19	33	10	307	161	167	152	134	115	-	-	2291
20 *	-	-	1628	1017	925	14	18	8	102	150	-
21 *	-	-	-	4	7	5	4	5	-	-	23051

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EVALUATION OF THE RADIOLOGICAL IMPACT DUE TO THE OPERATION OF NUCLEAR POWER STATIONS IN ARGENTINA

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INTRODUCTION

There are at present in Argentina two commercial nuclear power stations in operation, Atucha I and Embalse, generating about 10% of the total electrical energy output. Atucha I NPP, equipped with a pressure vessel reactor, has an output capacity of 345 MW(e), while Embalse NPP is equipped with a Candu-type reactor and its output capacity is 670 MW(e). Both plants operate with natural uranium as a fuel, and heavy water as a coolant and as a moderator.

Atucha I is located at the right side of the Parana de las Palmas river, 100 km from Buenos Aires city, and Embalse is situated at the homonomous town in the province of Cordoba, 620 km from Buenos Aires, next to the Embalse del rio Tercero lake.

GASEOUS AND LIQUID EFFLUENTS

During the operation of both nuclear installations, radioactive fission and activation products are produced. These radioactive materials are for the most part retained within the fuel elements. Most of the radionuclides which diffuse into or are formed within the coolant are removed by the gaseous and liquid waste processing systems. Low-level releases which occur during normal operation are controlled and monitored. Radionuclides may reach the environment through either the gaseous or liquid effluents streams.

GENERAL PROCEDURE FOR DOSE ASSESSMENT

In this report, the terms individual dose and collective dose are used to mean individual and collective effective dose equivalent commitment respectively.

The general principles followed in assessing individual and collective doses for both nuclear plants are similar to those used by UNSCEAR (1). The type of models used are those called concentration factor or equilibrium models, where steady state among nuclide concentrations in different environmental compartments was assumed. In this approach, simple multiplicative coefficients were used to obtain the concentration of radionuclides at the point of intake by man (2). Dosimetric factors were taken from published reports (3,4).

Local data, either site specific or regional, was used when possible, such as transfer factors, ingestion rates and other habit data. Default values presented in specialized

reports were used in case of lack of site specific data (2).

As the actual dose equivalents received by members of the public vary widely depending on such factors as age, sex, dietary and other habits, as well as on variations in their environment, appropiate critical groups were identified. These groups are representative of those individuals in the population expected to receive the highest dose equivalents from the installations under consideration. The individual doses in the critical groups estimated in this report, represent the mean effective dose equivalents , assuming the most unfavourable conditions. These values are used to compare with the corresponding individual dose equivalent limits fixed by the Regulatory Authority.

The collective dose commitment represents a measure of total exposure of the population over time from a given release and it is usually considered as an indicator of the total detriment to health from the consequent irradiation. Modelling procedures were similar to those used for estimates of individual doses, and the concentrations of radionuclides in environmental compartments extending over large regions were estimated (5). Global assesments include only H-3 and C-14, because of the small contribution to these evaluations due to Kr-85 and I-129 releases from these plants (1).

Releases activity and composition were informed by the nuclear power plants operators, accordingly to the licensing requirements established by the Regulatory Authority.

RESULTS

Individual and collective doses were calculated for both nuclear installations, by using the previously described methodology. The corresponding values are presented in tables 1 and 2. The main contributors to individual doses in critical groups and regional collective doses are H-3, Kr-88, Xe-133 and Xe-135 for gaseous releases, while H-3, Co-60 and Cs-137 are the most relevant nuclides in assessing doses due to liquid discharges.

Global collective doses for H-3 and C-14 were calculated by using measured and estimated releases (6,7). These collective dose commitments, per unit electrical energy generated, are 1 man Sv/GW(e).a and 7E-02 man Sv/GW(e).a,for H-3 releases from Atucha I NPP and Embalse NPP respectively. The corresponding values for C-14 gaseous discharges were estimated to be 46 man Sv/GW(e).a and 37 man Sv/GW(e).a

CONCLUSIONS

The regulatory authority in Argentina has fixed the pair of values 0.3 mSv/a and 15 man Sv/GW(e)a as upper bounds for limiting the annual exposure of individual members of critical groups and for the collective dose commitment per unit electrical energy generated during the operation of nuclear power plants(8).

The above collective dose commitment is the incomplete quantity, integrated over the period of the practice, assumed to be several centuries (500 years). The results presented in this report are well below the limiting values for individual doses. The normalized global collective doses estimated for C-14 are lower than expected values,(1), but higher than the present upper-bound, which was not standing during Atucha I NPP and Embalse NPP designing period. Besides, much more data, particularly from continous monitoring, are needed before a reliable assessment of the C-14 release rate and its environmental impact can be made.

On the other hand, as an application of the optimization requirement to C-14 releases, a retention system will be installed at the third Argentine nuclear power plant, Atucha II, at present under construction (9).

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TABLE 1
INDIVIDUAL DOSES IN CRITICAL GROUPS (Sv/a)

YEAR	ATUCHA I NPP ATMOSPHERIC RELEASES	ATUCHA I NPP LIQUID RELEASES	ATUCHA I NPP TOTAL RELEASES	EMBALSE NPP ATMOSPHERIC RELEASES	EMBALSE NPP LIQUID RELEASES	EMBALSE NPP TOTAL RELEASES
1974	6.0E-07	5.8E-07	1.2E-06			
1975	2.2E-07	1.6E-06	1.8E-06			
1976	2.1E-06	2.5E-06	4.6E-06			
1977	1.4E-06	1.5E-06	2.9E-06			
1978	3.4E-06	1.1E-06	4.5E-06			
1979	3.1E-06	1.6E-06	4.7E-06			
1980	2.8E-06	1.2E-06	4.0E-06			
1981	1.3E-06	1.3E-06	2.6E-06			
1982	1.4E-06	8.7E-07	2.3E-06			
1983	3.1E-06	6.4E-07	3.7E-06			
1984	9.0E-07	9.6E-07	1.9E-06	1.6E-08	1.6E-07	1.8E-07
1985	1.2E-06	8.9E-07	2.1E-06	4.8E-07	1.3E-06	1.7E-06
1986	1.5E-06	7.0E-07	2.2E-06	2.4E-07	5.6E-06	5.8E-06

TABLE 2
COLLECTIVE DOSES (REGIONAL) (man.Sv)

YEAR	ATUCHA I NPP ATMOSPHERIC RELEASES	ATUCHA I NPP LIQUID RELEASES	ATUCHA I NPP TOTAL RELEASES	EMBALSE NPP ATMOSPHERIC RELEASES	EMBALSE NPP LIQUID RELEASES	EMBALSE NPP TOTAL RELEASES
1974	1.7E-02	1.8E-02	3.5E-02			
1975	2.5E-04	7.3E-02	7.3E-02			
1976	6.8E-02	1.4E-01	2.1E-01			
1977	4.7E-02	2.4E-01	2.9E-01			
1978	1.0E-01	2.4E-01	3.4E-01			
1979	9.2E-02	2.8E-01	3.7E-01			
1980	9.2E-02	3.0E-01	3.9E-01			
1981	4.0E-02	4.1E-01	4.5E-01			
1982	4.8E-02	3.0E-01	3.5E-01			
1983	1.0E-01	2.4E-01	3.4E-01			
1984	3.0E-02	4.0E-01	4.3E-01	1.2E-04	5.6E-03	5.7E-03
1985	3.7E-02	3.7E-01	4.1E-01	4.7E-03	1.5E-02	2.0E-02
1986	4.2E-02	2.6E-01	3.0E-01	1.1E-02	8.9E-02	1.0E-01

SOME EXPERIENCE WITH MEASUREMENTS OF STACK RELEASES AND THEIR CORRELATION WITH ENVIRONMENTAL MEASUREMENTS.

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The first nuclear power plants (NPP) in Switzerland were built during the late 60-ties and the early 70-ties (PWR-Beznau and BWR-Mühleberg). A new generation of NPPs were built ten years later (PWR-Goesgen and BWR-Leibstadt). In all these NPP special attention was given to the sampling and measurements of the releases of radioactive materials, in particular long lived aerosols and J-131. The sampling systems at the Swiss NPP were originally installed on the basis of the "American National Standard Guide to sampling Airborne Radioactive Materials in Nuclear Facilities", (ANSI NI3,1 - 1969) or equivalent standards such as ISO 2889 and DIN 25423. Unfortunately, the sampling systems described in these guides cover only that fraction of radioactive aerosols, which is preferentially retained in various portions of the respiratory tract (0.3 to 10 µm). In a NPP one can expect during an accident a very wide range of particles with diameters as large as 100 and more µm, which can be transported away by the stack effluent and may not be properly sampled and measured. It should be taken in account that there are some non filtered rooms and that filters can fail or even break. Such particles when released cause a ground contamination in the vicinity of the plant. Such an event occurred in September 1986 in the NPP-Mühleberg.

1. THE MUEHLEBERG EVENT

Periodic measurements made in the years 1985 and 1986 in the vicinity of the plant, indicated traces of Co-60 (30 - 70 Bq/m^2). It was assumed that they were due to the routine plant releases. This nuclide was practically never found in the fallout of the atomic bomb tests.

In September 1986 [1], a further γ-spectrometric measurement in the vicinity of the plant indicated a contamination of Co-60, Cs-137 and Cs-134. This relatively high Co-60 (550 Bq/m^2) contamination indicated an unusual release of radioactivity from the plant. It was difficult to interpret the Cs-137 and Cs-134 contamination as a plant release, since the Cs-137 and Cs-134 levels were generally increased due to the Chernobyl fallout. The aerosol monitor in the stack exhaust did not indicate abnormal radioactivity releases. Spectrometric analyses of the aerosol grab sample filters of the stack have shown traces of Co-60 too, but did not indicate any abnormal aerosol releases.

The plant exhaust ventilation system was inspected downstream of the aerosol High Efficiency Particulate Filters. Massive contamination of ducts was discovered, including the exhaust stack. The contamination consisted of radioactive resin particles. The source of the contamination was identified as a centrifugal hydroextractor for resins, which was connected to the exhaust ventilation system through a prefilter. The prefilter was broken.Within the exhaust HEPA-filter system, 15 of 136 filter cells were found failed. However, the weekly records of Δp-measurements across the filters did not clearly indicate whether and when a filter had

failed. If one or two cells are leaking, the local flow increases and the Δp of the filter assembly does not change significantly.

2. MISINTERPRETATION OF THE MEASUREMENT RESULTS

The γ-spectrometric measurements of the stack aerosol filters are made weekly. The highest measured weekly peaks in April and September were about $3 \cdot 10^7$ Bq, thus they did not indicate any abnormal release. The weekly limit is $3.7 \cdot 10^9$ Bq. Since the highest peaks were still 100 times smaller then the weekly limit, no attention was paid to these peaks. Particles with diameters of 50 µm and more did not reach the sample filter and were found deposited in the horizontal parts of the sample line at 120 m. A careful interpretation made after the incident has shown a direct correlation of the measured γ-peaks with the use of the centrifugal hydroextractor to dry spent resin powders.

It is very probable that if a careful analysis of the measured peaks had been made before the incident, it could have been prevented. The environmental measurements, mentioned above, were the only indicators of the incident.

3. ENVIRONMENTAL IMPACT

The total estimated amount of radioactivity released is $1.1 \cdot 10^{10}$ Bq. The corresponding release limit for aerosols is $1.8 \cdot 10^{10}$ Bq per year. The release consisted of: 40 % Cs-137, 50 % Cs-134, 7 % Co-60, 3 % Zn-65. The contamination of the ground was limited to a distance of 1 km. Milk production of nearby farmers was controlled, but no restrictions were necessary.

The dose rate measurements in the two prevailing wind directions (east-west) have shown an increase, especially in the west direction of about 45 nSv/h. Table 1 gives the quarterly net dose data for the years 1985 and 1986 measured with the existing network of the thermoluminescence dosimeters (TLD) in the vicinity of the NPP. These data were obtained with use of the method of the site specific parameters (SSP) [2]. This method gives a possibility to obtain the net doses within an error of ± 20 µSv/quarter and about ± 40 µSv/year (3 S.D.).The estimated average of the Chernobyl fallout dose for 1986 in the vicinity of the NPP-Mühleberg is about 70 µSv ± 40 µSv. The higher doses on some measurement points (*) are probably due to the Mühleberg event (Fig. 1).

4. LESSONS LEARNED

4.1 The Mühleberg event has shown that sampling systems must be designed to identify a wide range of airborne particulates. Not only particles which will be preferentially retained in various portions of the respiratory tract, but also those which can be transported away by the stack effluent flow have to be taken into account.

4.2 Any changes of the measured results from the steady state level must be interpreted and explained.

Table 1. Net dose data of the TLD-network obtained in the years 1985 and 1986 in the vicinity of NPP-Mühleberg (μSv).

Station	1985 (± 20 μSv)				Total (±40 μSv)	1986 (± 20 μSv)				Total (± 40 μSv)
	I.	II.	III.	IV.	1985	I.	II.	III.	IV.	1986
1. Niederruntigen	-7	-12	2	6	-12	-8	45	51	56	152*
2. Siedlung WKW	4	2	7	-1	12	-6	31	35	51	117*
3. Fuchsenried	-23	-17	-7	9	-37	6	24	25	22	71
4. Eiau	5	-15	-9	7	-12	5	24	18	29	11
5. Leimeren	-16	44	-18	-1	9	-15	-9	4	5	0
6. Mühleberg	-3	3	-13	-4	-17	4	23	22	23	68
7. Wileroltigen	-6	0	8	0	2	-2	18	20	18	56
8. hin. Rewag	15	11	-1	3	28	-15	15	16	18	49
9. Ufem Horn	-1	-17	0	2	-16	8	41	38	100	179*
10. Salvisberg	-4	-8	18	2	9	12	39	38	26	103*
11. Aebnitacher	10	-23	2	2	-9	8	37	22	33	92
12. Buttenried	12	3	13	7	34	-4	31	35	24	90
13. Marfeldingen	14	-14	11	9	20	4	54	32	38	124*
14. Oberruntigen	-3	40	-5	-9	24	-7	31	33	25	89
15. Talmatt	1	-17	-4	-7	-27	12	39	20	22	81
16. Frieswil	2	31	5	-12	26	3	17	4	9	30
17. Murzelen	1	-10	-9	-14	-32	-4	17	13	8	38
average net dose	--	--	--	--	--	--	28	25	30	83[1]

1) This value includes an average calculated Chernobyl contribution of about 70 μSv ±40 μSv for the year 1986. The higher doses at some measurement stations (*) are due to the NPP-Mühleberg event in September 1986.

4.3 A redundant, preferably diverse sampling and monitoring system should be installed. An example of such a system, which will be backfitted in the NPP-Mühleberg, is shown in Fig. 2. This has two isokinetic nozzle systems each with their own independent monitoring apparatus. One of the systems is installed very near to the sampling nozzles, thus having very short sample lines. Both systems have particle separators (PSep) combined with on-line monitors for particles with diameters greater then 10 μm.

In addition, on-line monitors for small particles are available in each sampling system. The filters of the particle separators and the in-series installed grab sample filters provide representative particulate samples for laboratory analysis. A special arrangement is forseen for efficiency tests of the whole sampling and monitoring systems.

4.4 The environmental impact of the Mühleberg incident is low. Nevertheless, this event has shown that sensitive environmental measurements provide a useful detection capability for such incidents.

4.5 It should be stated that NPPs with intact HEPA-filters in almost all effluent paths, can be practically free of emission of radioactive particulates. An intensified surveillance programme has been adopted for the HEPA-Filters in the Swiss NPP to guarantee this emission free situation.

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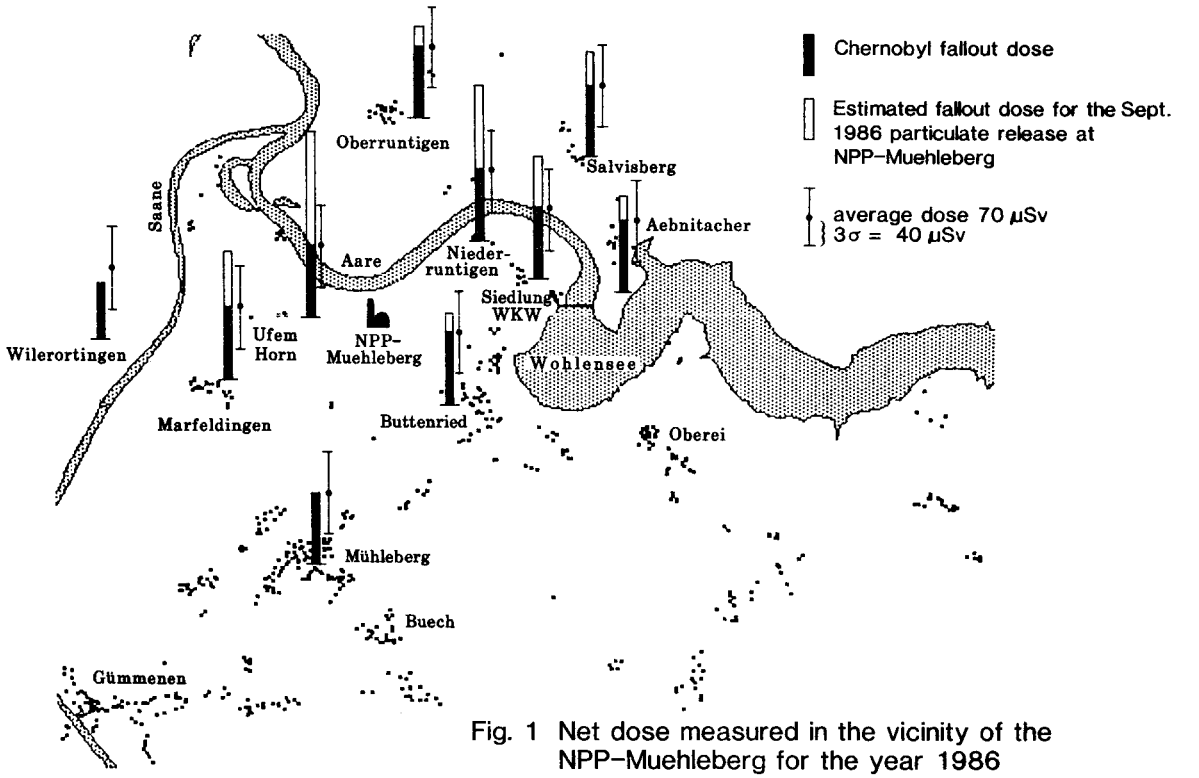

Fig. 1 Net dose measured in the vicinity of the NPP-Muehleberg for the year 1986

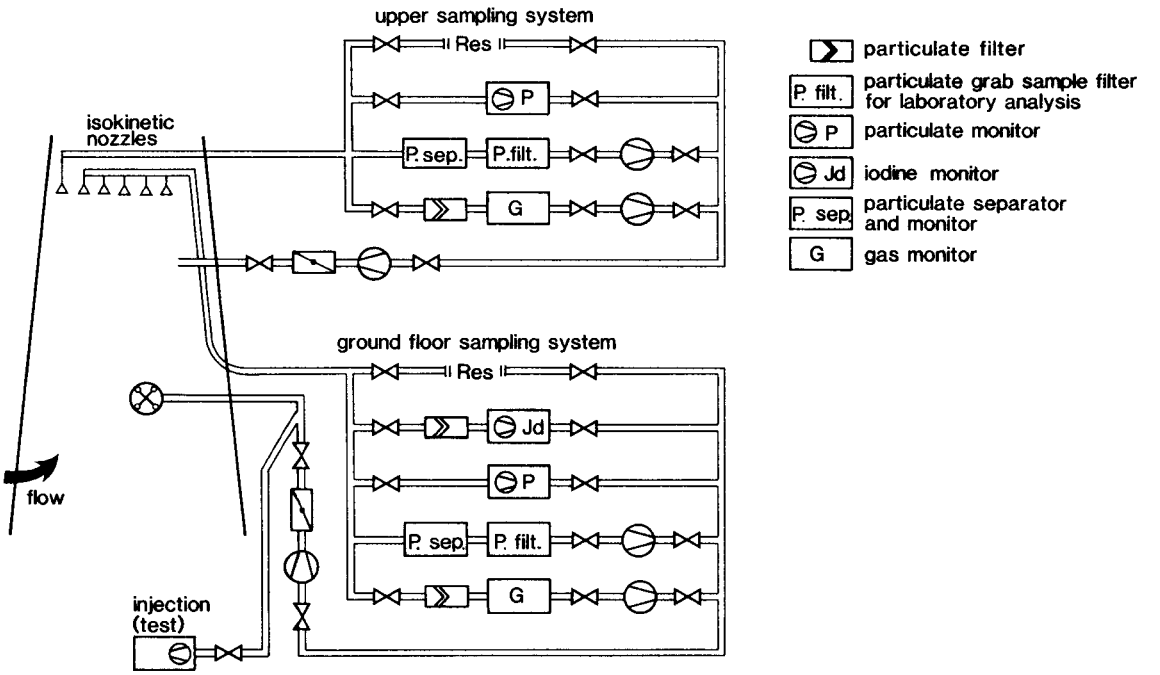

Fig. 2 NPP-Muehleberg backfited redundant stack effluent sampling and measurement system

RADIONUCLIDES SORPTION STUDIES IN EZEIZA SOILS *

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1. INTRODUCTION

The safety evaluation of a radioactive waste disposal system requires the estimation of its radiological impact. For that purpose, the study of the isolation capacity of the different barriers -either natural or man-made ones- which delay the delivery of the radionuclides into the environment is necessary. This report is related with the study of the delay capacity for Caesium and Strontium of the natural barrier where the low and intermediate radioactive waste disposal trenches are placed at the Ezeiza Atomic Center.

2. SORPTION MEASUREMENTS

The use of the distribution coefficient (K_d) was considered the best aproximation to the estimation of the degree of radionuclides sorption in geologic material, taking into account the great number of its limitations, associated with it character of empirical parameter of a natural system and, therefore, with several factors that affect it.

The distribution coefficient , defined as follows:

$$K_d = \frac{\text{Concentration of the radionuclides in solid phase}}{\text{Concentration of the radionuclides in liquid phase}}$$

represents a constant of equilibrium and includes a complex number of reactions or geochemical and biochemical processes, generally named sorption (4). Besides, there are other factors that influence the delay, such as diffusive processes, colloids filtrations or hydrated complex species and migration of particles.

Due to the large number of variables that affect the determination of the sorption of nuclides in geologic materials, it is impossible to controle all of them in an unique experience. Owing to this circumstance, it appears as an useful parameter the K_d coefficient, meassured in similar conditions to the media in study, varying successively the essay conditions. Besides, the K_d variations due to the factors related to the essay methodology have to be estimated.

A batch technic has been used in the experimental K_d determination, according to the following standard conditions: geologic material: extracted from the layer in contact with the free aquifer (1),(2); liquid/solid ratio: 25 cm^3 of traced groundwater / 1 g of dry geologic material; concentration of tracers (Cs-137 and Sr-85) and inactive carriers: see table 3.1; contact time: 1 hour; separation of phases: centrifugation, at 20000 g during one hour; pH: 8; measurement of the supernatant solution: τ spectometry (3).

* This work was done as part of the Research Contract Nº 4189/RB with The International Atomic Energy Agency.

Besides essays were carried out in order to evaluate the probable experimental errors intoduced in the K_d determination, and the variation of parameters of the experience.

3. CALCULATIONS

The experimental estimation of the K_d value is calculated as:

$$K_d = \frac{(A_i - A_f) \cdot V}{A_f \cdot P}$$

Where: A_i Initial activity in the contact solution
A_f Final activity in the contact solution
V Volume of the contact solution
P Weight of the geologic material

By applying the error theory to this it was deduced the formula for the standard deviation of the K_d value, as a result of the propagation of the introduced errors. These deviations were used to evaluate the repetibility of the results.

The expression for the desorption factor estimation is :

$$F_{des} = \frac{A_{f,\ f.l.des.}}{A_{i,\ f.s.}} \cdot 100$$

Where: $A_{f,\ f.l.des.}$ Final activity of the liquid phase after desorption

$A_{i,\ f.s.}$ Activity of the solid phase at the begining of the desorption experience (final activity of sorption essay)

4. RESULTS

The results of K_d measurements taken in different conditions are shown in figures 4.1 to 4.4 . For the standard conditions the K_d values are 850 ± 90 cm^3 /g for Cs and 80 ± 6 cm^3 /g for Sr. The results obtained with the variation of different parameters are the following (5),(6):

4.1. Contact time: In figure 4.1 it can be observed that the Cs and Sr sorption increased rapidly during the first contact minutes, giving constant results of the K_d value after 60 min.

4.2. Concentration of the carrier in solution: In the standard conditions we have worked with solutions about 10^{-4} M of carrier for both radionuclides. This order of magnitude of concentration was used to variate experimental parameters. Other determinations were carried out with concentration of 2 and 4 orders of magnitude lower than the standard one, in order to observe the incidence of the radionuclide concentration in the K_d value. In both cases we obtained an increase of the K_d value, which are shown in figure 4.2.

4.3. Influence of the pH: Determinations of the distribution coefficient in a large range of pH values (2 to 10) were carried out. In figure 4.3 it could be observed the evident influence of the pH in Cs and Sr sorption for this media.

4.4. Competitive ions: K^+ and Ca^{2+} were used as competitive ions for Cs^+ and Sr^{2+}. We observed that the presence of K^+ in high concentrations affect considerably the K_d of Cs (See fig. 4.4). Otherwise, the presence of Ca^{2+}, also at high concentrations, seems not to affect the Sr^{2+} sorption (See table 4.1).

4.5. Liquid/solid ratio: Its influence in the sorption was determined varying the ratio used in the standard tests ($25 cm^3/g$) , either for Cs or for Sr. Essays with the following relations were carried out: 10/1 , 10/2 and 10/5. In the case of Cs, it appears a remarked effect of the increase of the contact surface, because the K_d values obtained were considerably higher when the liquid/solid ratio was diminished (see table 4.2). On the other hand, in the case of Sr, it appeared a different tendency from expected, because the K_d value diminishes with the decrease of the liquid/solid ratio.

4.6. Desorption: In order to study the reversible or irreversible character of the sorption processes it was carried out some desorption measurements of radionuclides sorpted in geologic material into groundwater free of additives. The results, expressed as desorption factor, are shown in table 4.3.

5. CONCLUSIONS

Due to these considerations results that is not representative to inform an unique K_d value for a given media, and consequently, it has to be expresed as a range of values. This range is related with the uncertainty on the variation of phisical, chemical an geological parameters that modify the results.

From the wide spectre of experimental conditions that affects the delay it results that the range of variation of K_d for soils of the Ezeiza Atomic Center are 400 - 3500 cm^3/g for Cs, and 30 - 120 cm^3/g for Sr. Both ends of this range of values could be used in a sensibility analysis of mathemathic models of migration and dosis calculations. It is important too, to remark that these differences in the results (nearly an order of magnitude in the case of Cs) are not significant, considering their utilization in estimation models of mass transport and individual or collective doses. In these models there appear uncertainties due to the long term projections required as a consequence of long periods that the radionuclides migration last in the geologic media before reach the biosphere (thousands of years), and its influence is significative when the nuclide has a long half life. However , the lowest K_d values could be used in radiological impact estimations producing therefore the most conservative dose values.

Table 3.1 - Concentration of carriers and tracers in the contact solutions

	Tracer conc.	Carrier conc.
	Bq cm^{-3}	M
Cs	10.6	3.4 E-5
Sr	35.3	8.4 E-4

Table 4.1 - Influence of the presence of Ca^{2+} in the sorption of Sr^{2+}.

$[Ca^{2+}]$ M	K_d^{Sr} ($cm^3.g^{-1}$)
10^{-2}	80 ± 5,3
10^{-4}	75 ± 5,0
10^{-6}	82 ± 5,4

Table 4.2 - Influence of the liquid/solid rate

Relación líq/solido	K_d^{Cs}	K_d^{Sr}
($cm^3.g^{-1}$)	($cm^3.g^{-1}$)	($cm^3.g^{-1}$)
10/1	740 ± 70	36 ± 3
10/2	2260 ± 510	33 ± 3
10/5	3450 ± 1120	54 ± 3
25/1	850 ± 90	80 ± 6

Table 4.3 - Desorption in groundwater

Contact time	$F_{des.}^{Cs}$	$F_{des.}^{Sr}$
h	%	%
2	7,3	0,75
4	7,4	0,74
6	8,4	--
8	9,7	0,71

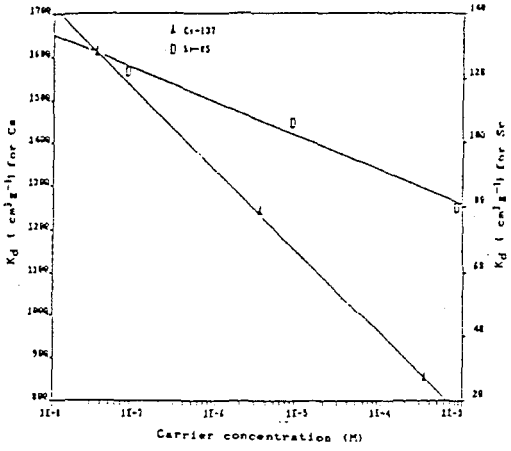

Fig. 4.2 - K_d for Cs and Sr as a function of carrier concentration

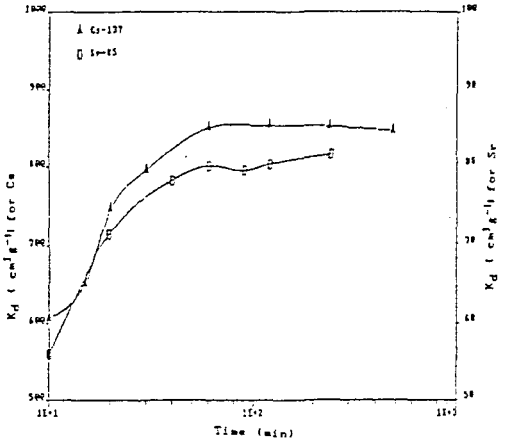

Fig. 4.1 - K_d for Cs and Sr as a function of the contact time.

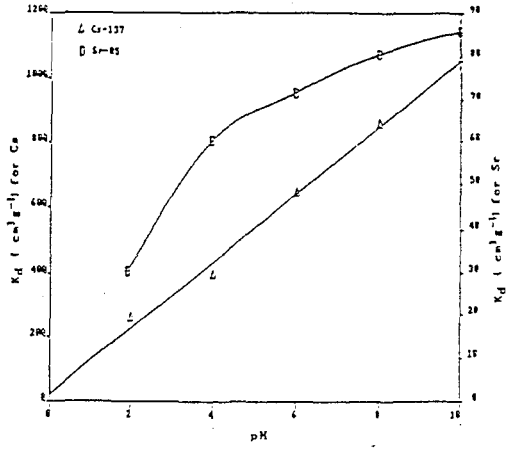

Fig. 4.3 - K_d for Cs and Sr as a funtion of pH

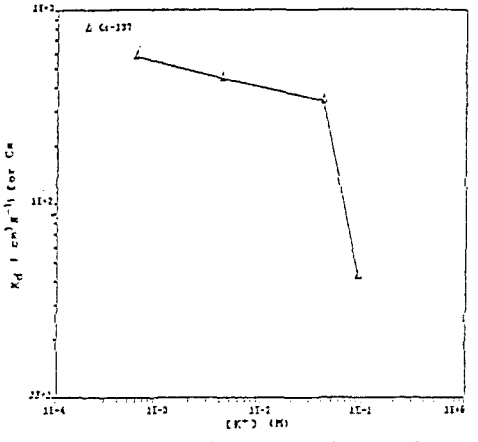

Fig. 4.4 - K_d for Cs as a function of K^+ concentration

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DETERMINATION EXPERIMENTALE DE LA VITESSE D'OXYDATION DU TRITIUM DANS L'ATMOSPHERE

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INTRODUCTION

A activité identique, le risque induit par le tritium sous forme eau tritiée est environ 10^4 fois plus grand que le risque induit par le tritium sous forme gazeuse (ref.1).

Dans le cadre des études de sûreté liées au fonctionnement des futurs réacteurs de fusion, il est primordial de connaître la cinétique de transformation du tritium gazeux (T_2 ou HT) en eau tritiée (HTO) dans l'atmosphère. Des divergences importantes existant dans la littérature (ref.2), les Communautés Européennes ont agréé en 1983 un programme destiné à étudier expérimentalement le comportement du tritium dans l'environnement à la suite de lâchers ; deux cas étaient à examiner : rejet accidentel rapide et rejet de routine lent.

La responsabilité de cette étude a été confiée à l'Institut de Protection et de Sûreté Nucléaire (IPSN) du Commissariat à l'Energie Atomique (CEA). Compte tenu de l'expérience acquise au Centre d'Etudes de Bruyères-le-Châtel dans la manipulation du tritium et de la possibilité d'utiliser une installation représentative (bâtiment avec ventilation et rejet par une cheminée de 40 m), le Service de Protection contre les Rayonnements (SPR) et le Service Chimie (DETN/C) de ce Centre ont été chargés d'organiser et de réaliser cette expérimentation.

PREPARATION DE L'EXPERIMENTATION

Le principe des mesures consistait, après le rejet d'un tritium "parfaitement sec", à effectuer des prélèvements atmosphériques à différentes distances pendant des intervalles de temps déterminés et à mesurer l'évolution du rapport HTO/HT+HTO.

L'appareillage retenu (barboteur), mis au point par le SPR, permet de différencier HTO et HT en utilisant le piégeage dans l'eau et l'oxydation catalytique sur platine ; l'activité tritium est mesurée par la technique de scintillation en phase liquide.

La préparation proprement dite a comporté les phases principales suivantes : examen des possibilités pratiques, vérification expérimentale sur le terrain, calculs théoriques, définition des emplacements des prélèvements, analyse de risque, autorisations.

Les principaux points pratiques examinés ont été : vérification des performances des détecteurs, limites d'analyse, emplacements possibles sur le terrain (topographie, densité de la population, facilité d'accès ...), conditions météorologiques optima (en fonction des données statistiques du site), définition et délais d'approvisionnement des différents matériels.

Une expérience de traçage a été effectuée par le Service de Protection des Installations Nucléaires (SPIN) de l'IPSN ; cette expérience consistait à relâcher du SF_6 par la cheminée prévue pour le rejet et à effectuer des essais de détection dans la zone retenue pour la mise en place des barboteurs.

Les estimations théoriques ont été effectuées à partir du code de calcul ORION, développé par le SPR (ref.3). Ce code de calcul est basé sur l'équation de diffusion classique à trois dimensions dépendant du temps (ref.4). Il permet d'évaluer le transfert de gaz ou d'aérosols dispersés dans l'environnement à l'issue d'un rejet contrôlé ou accidentel et d'obtenir, en quelques minutes et à diverses distances du point de rejet, les résultats suivants : quantité inhalée, concentration maximum, dépôts, irradiation, durée de passage du nuage. C'est la prise en compte dans notre code de calcul des différents paramètres définis précédemment qui nous a permis d'estimer la quantité à rejeter à 250 TBq.

L'analyse de risque a été effectuée pour cette quantité en prenant en compte des hypothèses particulièrement pessimistes : conditions de diffusion mauvaises et période de conversion rapide (1 heure) de HT en HTO. Le risque enveloppe pour une personne située au point le plus exposé (à 800 m dans l'axe du rejet) a été estimé à 2.10^{-6} Sv.

Compte tenu de tous ces résultats, on a retenu pour l'expérience une zone constituée d'un secteur angulaire de 30° centré sur l'axe des vents dominants (venant du 240°) avec l'installation dans le domaine public de 4 lignes de prélèvements : A, B, C, D, comportant respectivement : 1, 14, 14, 3 barboteurs et situées à : 800, 1000, 2500 et 4500 mètres de la cheminée de rejet au pied de laquelle un contrôle HT/HT+HTO était prévu.

En parallèle à l'aspect technique, les autorisations administratives étaient demandées : Service Central de Protection contre les Rayonnements Ionisants (SCPRI) dépendant du Ministère de la santé, Haut-Commissaire du CEA, Préfet, Maire.

REALISATION

La réalisation de l'appareillage et la préparation du tritium étaient de la responsabilité du Service Chimie. L'appareillage était installé dans une boîte-à-gants ventilée sous air. A partir d'un réservoir sous pression, un système automatique avec : capteur de pression, électrovanne, débitmètre massique, permettait d'obtenir un débit constant pendant la durée prévue pour le rejet. En cas d'incident, un système manuel permettait d'assurer le passage du gaz. Le tritium (sous forme HT) avait été purifié sur un four à uranium afin d'éviter toute trace de HTO, stocké dans un réservoir sous pression et analysé par spectrométrie de masse

Compte tenu de l'importance primordiale des conditions météorologiques, une attention particulière a été portée à ce point. Les prévisions météorologiques à 48 heures et 24 heures ont été assurées par le Centre de Météorologie Nationale de Brétigny, situé à 10 km à l'est du Centre de Bruyères-le-Châtel. Le suivi des conditions météorologiques a été effectué à l'aide du mât du Centre de Bruyères-le-Châtel (décalé de l'axe du lâcher de 500 m environ). Ce mât permet les mesures suivantes : vitesse et direction du vent, température, hygrométrie, à des hauteurs de 10, 40, 70 et 100 m ; les mesures sont effectuées en permanence et une valeur moyenne est éditée toutes les six minutes. Deux stations mobiles ont été installées sur les lignes B et C, dans l'axe du lâcher, afin de mesurer en continu la vitesse et la direction du vent à une hauteur de 10 m.

On signalera enfin que l'organisation de cette expérimentation par le SPR a nécessité la mise en place de moyens importants, tant en personnel (une quarantaine de techniciens sur le terrain), qu'en matériel (barboteurs, alimentation électrique, moyens de liaison : UHF-VHF, support logistique, analyses, ...) et a comporté des actions de formation des techniciens et deux répétitions générales.

PARAMETRES EXPERIMENTAUX

L'expérience s'est déroulée le 15 octobre 1986 dans les conditions suivantes : quantité de tritium : 256 TBq ; tritium "sec" : HTO/HT+HTO ≈ 2.10^{-4} ; durée du rejet : 2 mn ; vitesse du vent : 3 m/s au mât principal à 40 et 70 m de haut, 2,5 m/s aux mâts secondaires B et C à 10 m de haut ; vent stable en direction ; pas d'inversion de température ; hauteur réelle du rejet : 49 m (surhauteur cheminée) ; durée des prélèvements par rapport à l'heure H du rejet (12 h 12 TU) H à H + 30 mn, H + 30 mn à H + 90 mn, H + 90 mn à H + 270 mn.

RESULTATS

On a reporté sur les figures ci-après les résultats des prélèvements de H à H + 30 mn pour les lignes B et C et la courbe obtenue par le code de calcul ORION.

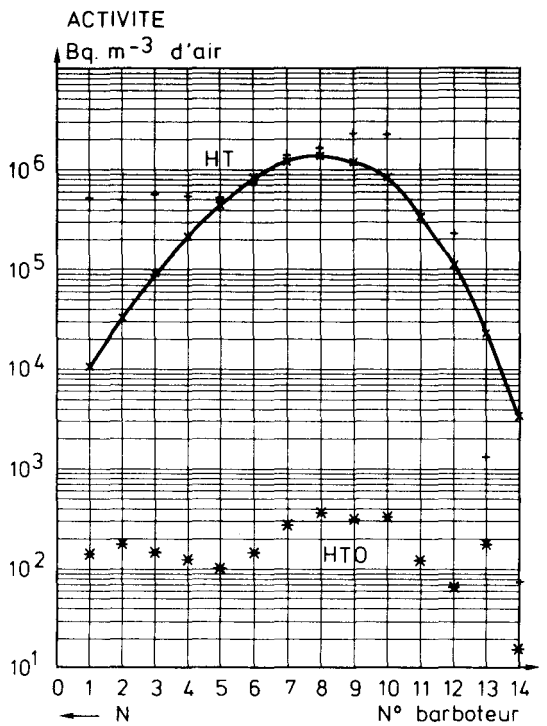

Figure 1 : Ligne B

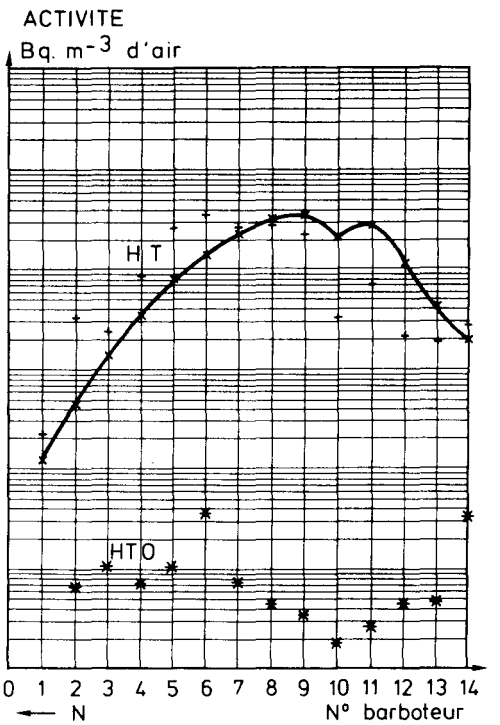

Figure 2 : Ligne C

Comparaison entre les résultats expérimentaux (HT - HTO) et le code de calcul (HT) Prélèvements H à H + 30 mn. Vitesse du vent 3 m/s. Origine 240°.

On notera le bon accord général entre les résultats expérimentaux et les calculs théoriques. La dissymétrie constatée sur la figure 1 est expliquée par la présence proche d'une parcelle boisée et la singularité sur la figure 2 provient du fait que le point numéro 10 était situé en avant de la ligne C moyenne.

On constate que la valeur moyenne du rapport HTO/HT+HTO sur les lignes B et C, vers l'axe du panache, est identique à la valeur mesurée en pied de cheminée soit 2.10^{-4} ; ce résultat indique que l'oxydation du tritium gazeux dans l'atmosphère est très lente. Pour les prélèvements effectués dans les intervalles de temps suivants (H + 30 mn à H + 90 mn et H + 90 mn à H + 270 mn), on a constaté une augmentation du rapport HTO/HT+HTO ; cette augmentation est due au phénomène de conversion du HT au niveau du sol et de réémission sous forme HTO (ref.5).

On notera également que des équipes de différents laboratoires étrangers (Allemagne, Suède, Canada) participaient à l'expérience et se trouvaient au point A (800 m du point de rejet, dans l'axe du panache), leurs résultats sont en bon accord avec les nôtres.

ESTIMATION DU RISQUE

Les résultats ci-dessus permettent de déterminer le risque inhalation : pour une personne se trouvant à l'endroit du risque maximum (point A), l'équivalent de dose engagé par l'expérimentation est de l'ordre de 10^{-7} Sv. Le code de calcul ORION permet de tracer des courbes isodoses centrées sur l'axe du vent, qui montrent un équivalent de dose engagé par le passage du nuage de 10^{-9} Sv sur 0,29 km^2 et de 10^{-10} Sv sur 5,51 km^2. Ces valeurs sont à comparer au risque induit par l'irradiation naturelle qui est de l'ordre de 10^{-3} Sv par an dans la région parisienne.

Ces résultats sont confirmés par l'absence de détection tritium dans les prélèvements urinaires de huit techniciens répartis sur le terrain d'expérience et analysés par les méthodes utilisées pour la surveillance des travailleurs exposés au risque tritium.

CONCLUSIONS

Le lâcher contrôlé de 256 TBq de tritium gazeux dans l'environnement s'est déroulé selon les conditions expérimentales prévues et on peut en tirer les conclusions suivantes :

. L'oxydation du tritium gazeux dans l'atmosphère à l'occasion d'un lâcher accidentel rapide est négligeable,

. Il y a un bon accord entre les résultats expérimentaux et les estimations du code de calcul ORION,

. L'équivalent de dose engagé n'est pas significatif.

REMERCIEMENTS

Il est impossible de citer nominativement ici toutes les personnes et organismes ayant contribué à la préparation et à la réalisation du lâcher, mais nous tenons à remercier l'ensemble des participants à cette expérimentation.

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EXPERIMENTS AND CALCULATIONS ON THE ADSORPTION OF RADIOACTIVE ELEMENTAL IODINE GAS ON AEROSOL

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INTRODUCTION

The deposition velocity of radioactive elemental iodine (I_2) gas from air to vegetation is about five times faster than that of iodine attached to particulate matter (i.e. particulate iodine) [1]. In the environment, a fraction of radioactive I_2 released from nuclear facilities is likely to adsorb on the atmospheric aerosol. The adsorption of I_2 onto various aerosols was studied in order to provide basic data for the realistic and precise assessment of the internal dose to man from radioiodine in the food chain. The influence of I_2 gas concentration and reaction time on the adsorption amount of I_2 onto burnt incense-stick fume and fly-ash aerosol was investigated experimentally and was also analyzed theoretically.

EXPERIMENTAL METHODS

The experimental apparatus is shown in Fig. 1. The I_2 gas was generated by vaporization of the I_2 crystal which was made by a reaction of sodium iodide ($Na^{131}I$) and potassium dichromate ($K_2Cr_2O_7$). A mixture of aerosol and I_2 gas was passed through a glass vessel so that particulate iodine could be formed by adsorption of I_2 on an aerosol. Two types of glass vessels, volumes of which were 8 liters for a short time reaction and 155 liters for a long time reaction, were used.

After the adsorption, particulate iodine and unadsorbed I_2 gas were trapped separately using an iodine species discriminating sampler called a modified Maypack sampler [2]. These iodine activities were measured with an NaI(Tl) scintillation counter. The incense-stick aerosol was generated in a small glass cylinder by burning an incense stick (2 mm dia. × 133 mm). The fly-ash aerosol was generated with a dust feeder (Sibata Scientific Technology Ltd., Model MF-2). Particle concentrations (n) were measured with a condensation nucleus counter (Thermo-Systems Inc., Model 3020). Size distributions of incense-stick and fly-ash aerosols were measured with an electrical aerosol

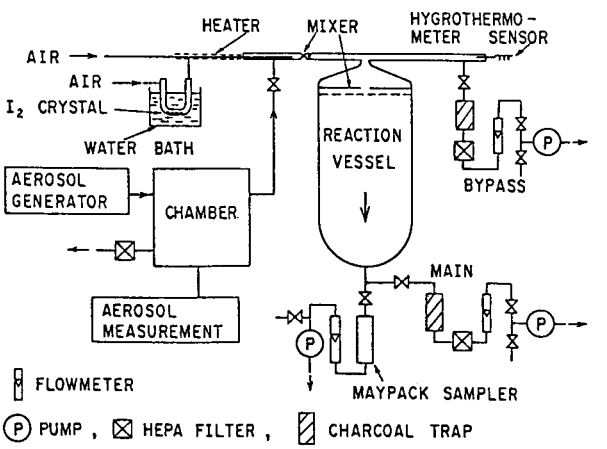

Fig. 1 Experimental apparatus for I_2-aerosol reaction.

analyzer (Thermo-Systems Inc., Model 3030) and an electron microscope (Hitachi Ltd., H-300), respectively. The geometric mean diameter (d_g), the geometric standard deviation (σ_g) and the mean surface diameter ($d_s=d_g\exp(\ln^2\sigma_g)$) measured with these instruments were as follows: d_g=0.14 μm, σ_g=1.75 and d_s=0.19 μm for the incense-stick aerosol, and d_g=0.52 μm, σ_g=1.60 and d_s=0.62 μm for the fly-ash aerosol.

EXPERIMENTAL RESULTS

The adsortion amounts (U) of I_2 as a function of reaction time are shown in Fig. 2 for the incense-stick aerosol and in Fig. 3 for the fly-ash. The curves in these figures represent the results of a theoretical analysis as described later. The adsortpion amount for the incense-stick aerosol increased rapidly with reaction time during the first 1 minute and then reached the equilibrium state at about 2.4×10^{-10} g/cm^3 of initial I_2 concentration (C_0). For the fly-ash aerosol, however, the time required until the adsoption reached the equilibrium state was about 10 minutes at $C_0=1.5\times10^{-10}$ g/cm^3. It was about ten times longer than that for the incense-stick aerosol. The adsorption amounts in the equilibrium state are about 2×10^{-7} g/cm^2 for the incense-stick aerosol and about 5×10^{-8} g/cm^2 for the fly-ash aerosol.

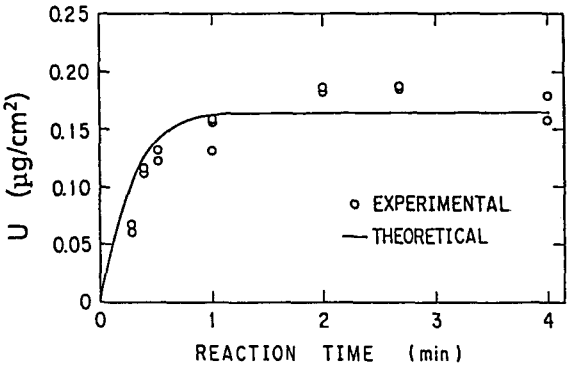

Fig. 2 Adsorption amount (U) of I_2 as a function of reaction time for the incense-stick aerosol at n= 1.7×10^5 cm^{-3} and $C_0=2.4\times10^{-10}$ g/cm^3.

In Fig. 4, the relation between I_2 gas concentration (C) and adsorption amount in the equilibrium state is shown. The curves obtained by the least squres method are equivalent to adsorption isotherms of the aerosols for I_2 at room temparature. The curves were expressed by:

$$\ln U=-11.1-0.407C^{-0.109} \quad (1)$$

for incense-stick aerosol,

and

$$U=0.0357C^{0.620} \quad (2)$$

for fly-ash aerosol. The adsorption isotherm of fly-ash aerosol was similar to Freundlich type one. The isotherm of incense-stick aerosol could not be

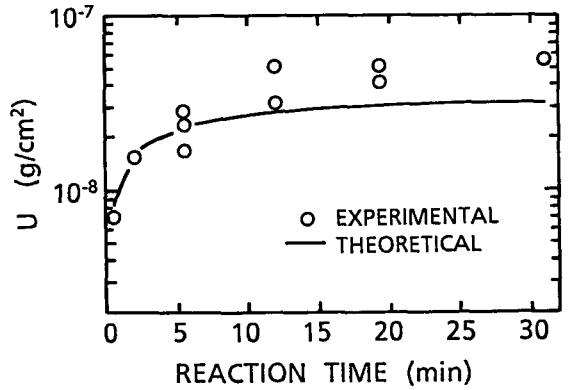

Fig. 3 Adsorption amount (U) of I_2 as a function of reaction time for the fly-ash aerosol at n=1.0×10^4 cm^{-3} and $C_0=1.5\times10^{-10}$ g/cm^3.

represented by an usual type of adsorption isotherm over the whole range of iodine concentration.

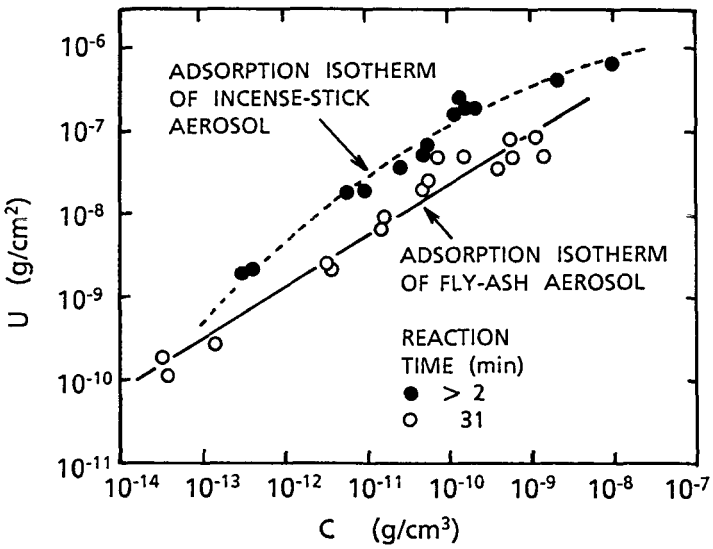

Fig. 4 Adsorption isotherms of incense-stick and fly-ash aerosols for I_2 gas. C is I_2 gas concentration and U is adsorption amount of I_2.

THEORETICAL ANALYSIS

The following equation based on a theory of Fuchs [3] for the evaporation of droplets was applied to iodine adsorption by Chamberlain et al. [4].

$$-dC/dt = 4\pi rnDC/\{D/rN\alpha+r/(r+\Delta)\}, \quad (3)$$

where t=time (s), C=concentration of I_2 in air (g/cm^3), r=radius of particles (cm), n=particle concentration (cm^{-3}), D=diffusion constant of I_2 in air (=0.08 cm^2/s), $N=(RT/2M)^{1/2}$ is a quarter of the mean kinetic velocity of I_2, α=sticking probability (-), and Δ=concentration jump distance ($=1.04\times10^{-5}$ cm [5]). This equation can apply for irreversible adsorption but cannot calculate the iodine loss for reversible adsorption like this experiment. Hence, the following equation [6, 7] was adapted to analyze the present results.

$$-dC/dt = 4\pi rnD(C-C_p)/\{D/rN\alpha+r/(r+\Delta)\} \quad (4)$$

where C_p=concentration of I_2 on particles (g/cm^3). It was assumed that C_p was given by the following equations derived from the adsorption isotherms as a function of adsorption amount:

$$C_p = \{(-11.1-\ln U/0.407)^{-9.174} \quad \text{for incense-stick aerosol,} \quad (5)$$

and $C_p = 215U^{1.61}$ for fly-ash aerosol. (6)

Using the least squares method, the sticking probability for the incense-stick aerosol was determined by substituting into equation (4) the experimental data of relation between the initial I_2 gas concentration and the proportion of iodine adsorbed on particles at 1 minute of reaction time. The sticking probability was 7.6×10^{-3}, which agreed with the data obtained by other investigators: 7.6×10^{-4} to 1.9×10^{-2} for aerosol in a reactor shell [8] and 2×10^{-3} to 2×10^{-2} for the atmospheric aerosol [9]. For the fly-ash aerosol, it was found that the sticking probability depended on the adsorption amount [7]. The equation is described by

$$\ln \alpha = -55.0U^{0.1019}. \qquad (7)$$

An increase of the adsorption amount from 10^{-11} to 10^{-8} g/cm^3 raised the sticking probability from 10^{-4} to 10^{-2}, which also agreed with the previous results.

The adsorption amounts as a function of reaction time were calculated using equations (4) to (7) and were compared with the experimental data. The results are shown in Figs. 2 and 3. It is seen that the calculated results are in good agreement with experimental ones. This indicates that the present calculation method is effective for analyzing the adsorption of I_2 gas onto incense-stick and fly-ash aerosols.

CONCLUSIONS

The adsorption of radioactive elemental iodine (I_2) gas onto burnt incense-stick and fly-ash aerosols was studied. For the both aerosols, the adsorption reached the equilibrium state in about ten minutes. The adsorption isotherm of fly-ash aerosol was similar to Freundlich type one. The isotherm of incense-stick aerosol could not be represented by an usual type of adsorption isotherm over the whole range of iodine concentration. Theoretical equations to explain the adsorption were derived by introducing the term of I_2 concentration on particles, which was estimated from the adsorption isotherms, into Chamberlain's equation. The calculated results were in good agreement with the experimental ones.

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LONG-TERM OBSERVATION OF TRITIUM IN PINE NEEDLES NEAR NUCLEAR FACILITIES

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INTRODUCTION

An interest has been focussed to tissue bound tritium (TBT) in foods, because it may be directly assimilated in the bound compartment of tissues, and increase cumulative total body dose[1]. Transfer from environmental tritiated water to the tissue bound fraction in plants had been investigated through many laboratories and relatively lower percentage fixation was suggested[2]. On the other hand, analysis of various kind of environmental samples which were considered to be affected by fallout tritium only showed high specific activity ratio of TBT to tissue water tritium (TWT)[3,4]. This phenomenon had been attributed to either following cause:a) long retention of tritium previously incorporated in the soil during the period of maximum fallout[3], or b) an incorporation of high specific activity elemental tritium in the atmosphere into the soil microbes and plants [5]. More studies have to be performed in the environment as well as in the laboratories on the behaviour of TBT in different types ecosystem in order to solve this problem, because few literatures have been available on the TBT behaviour in the chronically contaminated environment [6,7]. Therefore, a long-term study was conducted on the behaviour of TBT and TWT in plants in a location which had been affected mainly by tritiated vapour discharged into the atmosphere.

METHODS

Pine trees were selected for the investigation, because of their availability regardless places and seasons in Japan and a capability of simultaneous collection of present and old pine needles back to a few years ago from a branch. Samples had been collected in their growing seasons on the monthly basis from 1981 to 1986 except 1984 at several points in a down-wind area within 2km from two heavy-water moderated research reactors JRR-2 and JRR-3 of JAERI in Tokaimura, Ibaraki Prefecture. Monthly accumulated precipitation, groundwater and surface soil to 1m depth were also collected. Detail discription of sampling points, location of nuclear facilities, meteorological conditions, tritium deposition-source distance relationship in this area and the time-variation in tritium concentration of water form in various samples during 1981 and 1983 were presented previously[8,9].

Freeze-dried pine needles were plasma-ashed[10] and tritium concentrations of both freeze-dried water and oxidation water

were determined by means of liquid scintillation counters (LSCs). The tritium concentration of both waters are expressed in pCi/l The detection limit for 500 minutes of counting lies at approx. 20 pCi/l for 40ml freeze-dried water by a LSC ALOKA-LB1, and 50 pCi/l for 10ml oxidation water by a PACKARD TRICARB 2000CA/LL.

RESULTS AND DISCUSSIONS

Time-variations of tritium concentrations in the tissue water (TWT) and the tissue bound form (TBT) of the pine needles and in monthly accumulated precipitation collected at about 700m south-west from JRR-2 and JRR-3 were shown in Fig.1 with that of sum values of monthly discharge rates of tritiated vapour into the atmosphere from the two reactors. It is shown that several curies of tritiated vapour were discharged every month under their normal operation except June 1982 when abnormal discharge rate 44Ci was recorded due to a small leakage of heavy water from JRR-3. The tritium concentration of monthly precipitation varied from about 50 pCi/l to about 600 pCi/l except an abnormal high value of 2570 pCi/l in June 1982. The tritium concentration of tissue water varied from about 100 pCi/l to about 1000 pCi/l throughout 1981-1985 except a value of 2500 pCi/l in June 1982. On the contrary, the concentration of tissue bound tritium varied from about 250 pCi/l to about 1400 pCi/l without exception. The TBT data in 1984 were estimated by oxidizing pine needles having grown in 1984 which were separated from branches collected in 1985 and were plotted inside a rectangle in Fig.1 in order of their collection date. This memory effect of TBT in the pine needles has been almost verified with agreement in TBT concentrations between some pine needles of a same growth year but different collection years.

The concentrations of TWT and TBT in the pine needles at the point mentioned above proved to be 10 to 20 times higher than natural environment[11]. The tritium discharge rates, the tritium concentrations in both the monthly precipitation and tissue water of the pine needles fluctuate larger than the TBT concentrations, however annual average of these four items seemed to correlate each other. Therefore correlation was examined between annual mean TWT or TBT concentration and each of following three factors:a) annual mean tritium concentration in monthly precipitation, b) annual tritium deposition and c) annual discharge rate. The results were shown in Fig.2 for TWT and in Fig.3 for TBT . From Fig.2, the TWT concentration proved to correlate well with the three factors a, b and c with correlation coefficients from 0.95 to 0.99. From Fig.3, the TBT concentration proved to correlate fairly well with the three factors with correlation coefficients from 0.75 to 0.92. These results suggest that tritium in the pine needles at the point originated from the atmospheric discharges mainly from JRR-2 and JRR-3 and was incorporated mainly through the root after washout deposition as well as through the air. It is worth noting that the ratio of annual mean specific activity of TBT to TWT was about 1.7 from 1981 to 1983 and 1.1 in 1985, and the ratio of

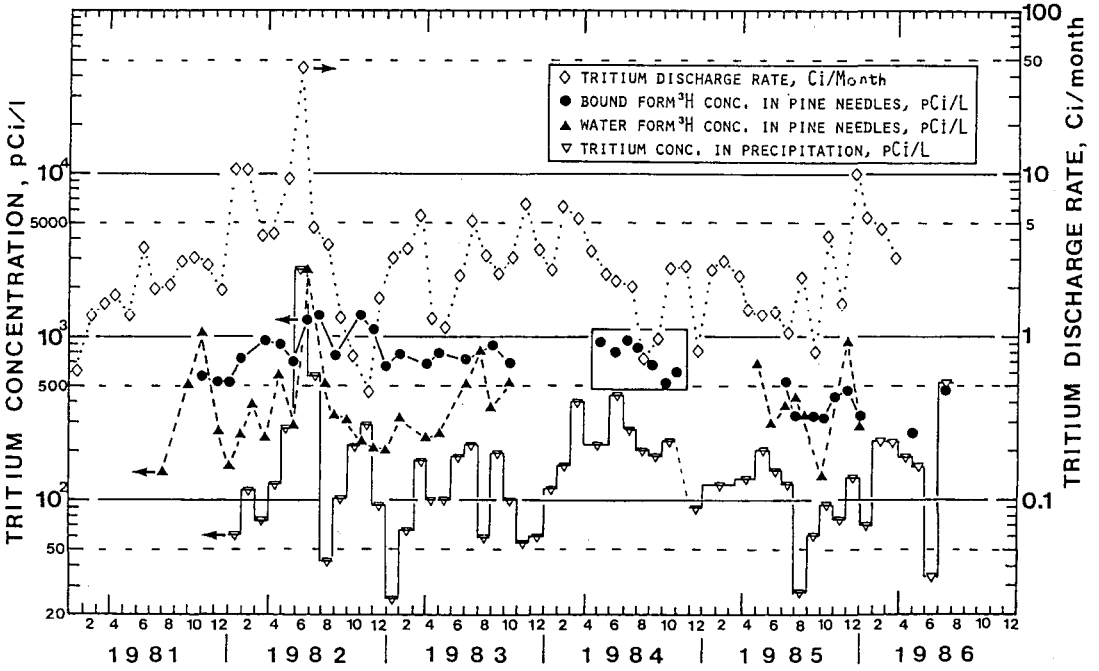
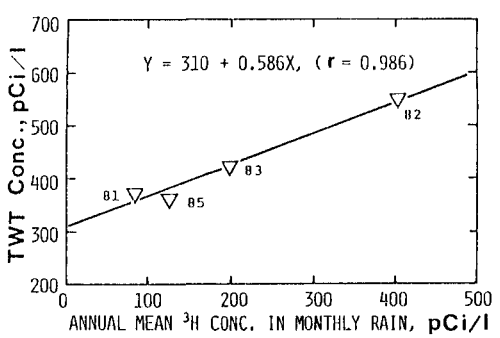


Fig.1 Time variations of TWT and TBT concentrations in the pine needles, ^{3}H concentration in rain and ^{3}H discharge rate.



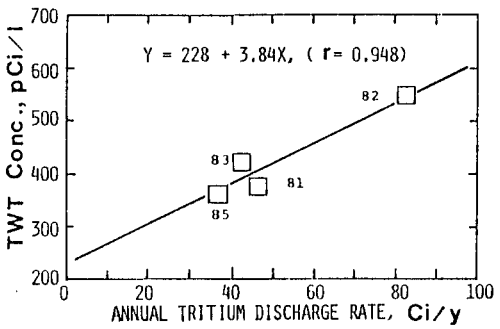
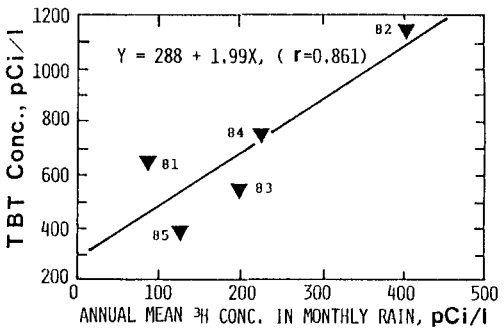


Fig2. Correlation of TWT conc. with ^{3}H conc. in rain (↑) and with annual ^{3}H discharge rate (↓).



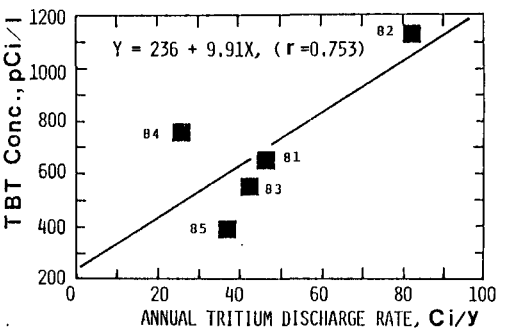


Fig.3 Correlation of TBT conc. with ^{3}H conc. in rain (↑) and with annual ^{3}H discharge rate (↓).

the annual mean specific activity of TWT to the precipitation was ranged from 1.4 to 4.5 with an average 2.7 throughout 1981-1985 except 1984.

Weekly monitoring data of HT in air having been obtained by PNC and JAERI in this area showed its level of 1-2pCi/m^3 through several recent years which was the same level with that in natural environment. The mean specific activity of TBT in pine needles in whole Japan was reported 66 $\pm$18 pCi/l whereas that of TWT was 46 $\pm$15 pCi/l[11]. Therefore, the elevated specific activity of TBT observed at the point can not be attributed to HT in the air. The mean specific activity of tritium in the soil moisture in the area appeared to be medium between those of the precipitation and the tissue water in the pine needles. The monthly mean specific activity of tritium in the air vapour was reported to be about twofold higher than that of the monthly precipitation at the monitoring station No.7 of JAERI which was located close to the point of the study. The elevated specific activity of TBT can not be explained by the tritium level of monthly mean or chronically contamination both in the soil moisture and the air vapour.

It has been demonstrated under non-equiribrium conditions at Karsruhe that after leaves were exposed temporarily high specific activity tritiated vapour, the specific activity of TBT in leaves reached approx. one tenth of vapour and it was retained with a considerably high ratio of TBT to TWT until the end of the vegetation period due to a retention effect of TBT[6]. An exposure to abnormally high level of tritiated vapour had supposedly happened also to the area in Tokaimura in the middle of June 1982 accompaning with the high level of tritium washout deposition. In addition to this event, temporal elevation of tritium level in the air vapour might had happened repeatedly at the area along the main downwind direction from JRR-2 and JRR-3 and resulted in the long-term retention of higher level of TBT in the pine needles.

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BEHAVIOR OF TRITIUM IN ENVIRONMENTAL SAMPLES AROUND NUCLEAR FACILITIES AT TOKAI-MURA

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1.INTRODUCTION

Many kinds of nuclear facilities such as Power Reactors, Research Reactors, Reprocessing Plant etc. are operated in Tokai-mura, located 140 km north of Tokyo. From a point of view of tritium discharge to the environment, Heavy Water Reactors and a Reprocessing Plant are noteworthy.

Tritium concentrations in drinking water, river water, rain water and sea water have been measured in a series of environmental monitoring around the fuel reprocessing plant at Tokai-Works, Power Reactor and Nuclear Fuel Development Corporation (PNC). Besides this monitoring, a measurement method of tritium concentration in the air has been investigated. This report summarizes the results of tritium concentration in the air at Tokai-mura.

2.METHOD

Atmospheric tritium occurs mainly in two chemical forms which are tritiated water vapor (HTO) and tritium gas (HT). We developed an atmospheric tritium sampler which can differenciate between HTO and HT (Fig.1). These samplers have been set at three monitoring points around Tokai-mura and the concentrations of HTO and HT in the air have been measured separately since 1983; while only HTO had been measured by using the other samplers since 1976.

The procedure of differenciating between HTO and HT is as follows:

Water vapor is adsorbed on the first molecular sieve column. Tritium gas passed through this column is oxidized by the palladium (Pd) catalyzer and adsorbed on the last molecular sieve column (HT column in Fig.1). HTO free water is decomposed into hydrogen and oxygen electrolytically in an Electrolysis cell to give a hydrogen carrier for HT. Then, the decomposed hydrogen with the sampled HT in the air is oxidized by the Pd catalyzer. The produced H_2O and HTO are adsorbed on the HT column and the Pd catalyzer column. By knowing the ratio of an amount of water adsorbed by both columns to the electrolyzed water, the efficiency of Pd catalyzer is obtained as follows;

$$\text{(Efficiency of oxidation)} = \frac{W_{Pd} + W_{HT}}{W_E - W_D} \times 100 \quad (\%)$$

W_{Pd} : Gain in weight at Pd Column (gram)
W_{HT} : Gain in weight at HT Column (gram)
W_E : Loss in weight at Electrolysis Cell (gram)
W_D : Gain in weight at Drying Column (gram)

Efficiency of oxidation is, on average, almost 100%

After the sampling, each column is heated about 450°C in a furnace. Water trapped by each column is desorbed and recovered by a cold trap. Forty milliliters of sampled water is mixed with 60 ml of scintillator (AQUASOL-2). After that, the mixture is kept in a cooler (about 10 °C) for about one day. Then, tritium concentration in the recovered water is measured by a low background liquid scintillation counter. That measurement is repeated ten times, counting unit being 50 minutes. The detection limit is about 20-30 pCi/ℓ.

3.RESULTS and DISCUSSION

As an example of seasonal variation of HTO concentration in the air, the data measured at one monitoring station are shown in Fig.2. There are two kind of units to show tritium concentration in the air by pCi/m^3, and in the water vapor by pCi/ℓ. Although tritium concentrations in water vapor are stable, the tritium concentrations in the air show high in the summer and low in the winter because of humidity. The chronological change of annual tritium concentration in water vapor is shown in Fig.3 and that in the air is shown in Fig.4. Each geometrical mean (m_g) plotted on Figs.3 and 4 is obtained based on about fifty individual values measured at each place during a year, respectively. Figures 3 and 4 also show the values of '$m_g \times \sigma^3$' where σ is a geometric standard deviation. The measured data at monitoring station-1 (ST-1) have practically shown the same values around 140 pCi/ℓ in water vapor and 1 pCi/m^3 in the air, respectively . On the other hand, the measured data at monitoring station-2 and 3 (ST-2,3) have shown a decreasing tendency. The decreasing rates calculated by the least square method, based on the measured data at ST-2 and 3 from 1980 to 1986, are about 5 pCi/ (ℓ·year) and about 0.05 pCi/(m^3.year), respectively. These values correspond to the half-life of 10-11 years which are almost the same as the decay half-life of tritium.

The chronological concentration of HT in the air is plotted on Fig.5. The geometric mean based on the measured data at ST-1, 2, and 3 show the same values and tendency. There is no difference among these data. The individual data throughout the year also present no seasonal variation as in the case of HTO.

The measured data at ST-3 in 1984 are plotted on a log-normal distribution paper as shown in Fig.6. This indicates that the data of HTO and HT in the air fit the log-normal distribution. Figure 6 shows that the geometric mean of HTO is 0.45 times lower than that of HT, while the standard deviation of HTO is 1.5 times higher than that of HT.

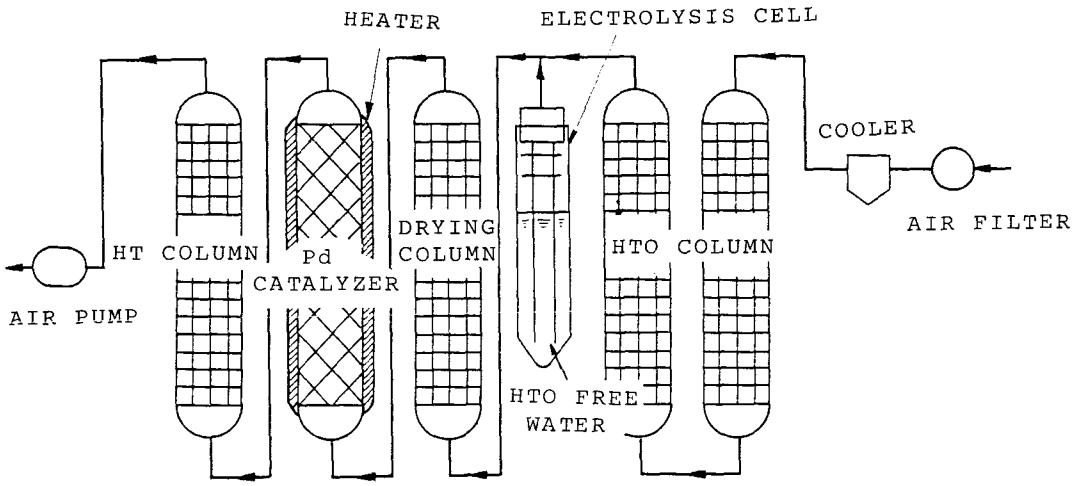

Fig.1 Atmospheric HTO and HT Sampler

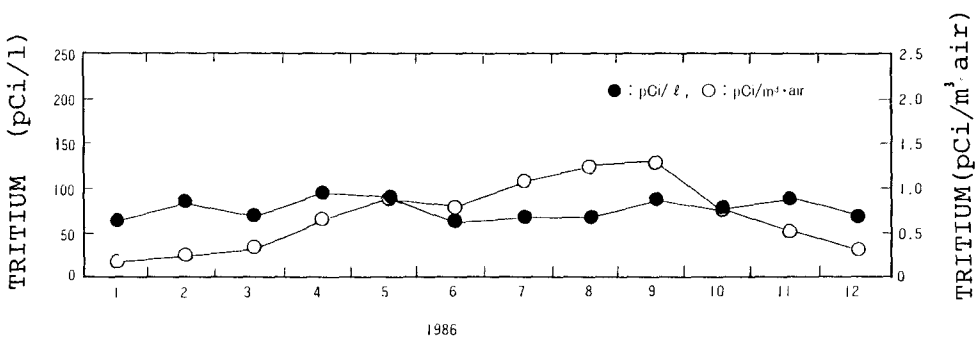

Fig.2 Monthly Tritium Concentration in Air around Tokai-mura

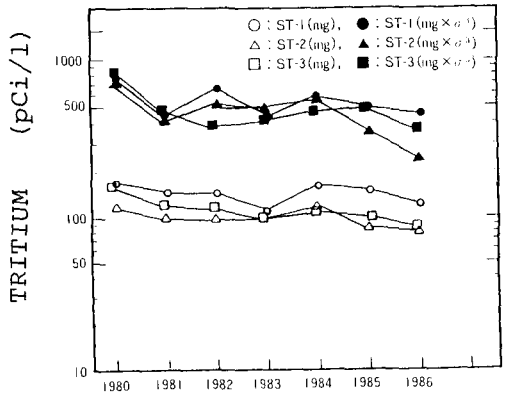

Fig.3 Annual Tritium Concentration in Water Vapor (pCi/l)

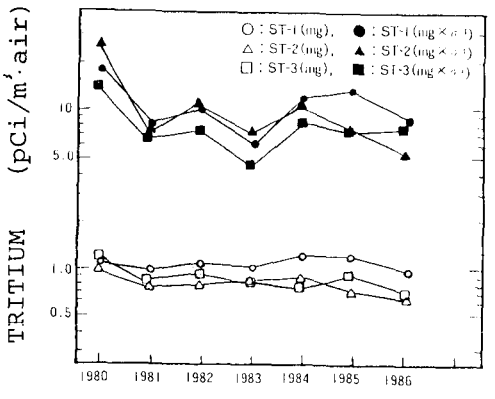

Fig.4 Annual Tritium(HTO) Concentration in Air (pCi/m³·air)

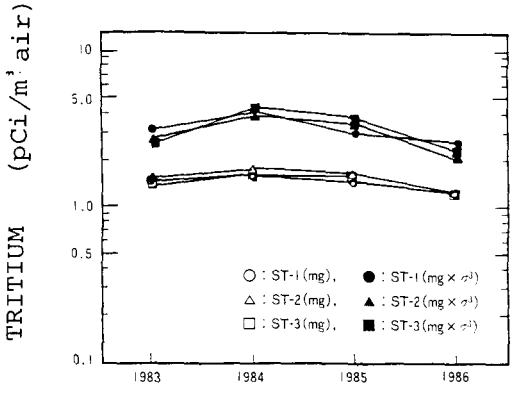

Fig.5 Annual Tritium(HT) oncentration in Air (pCi/m³ air)

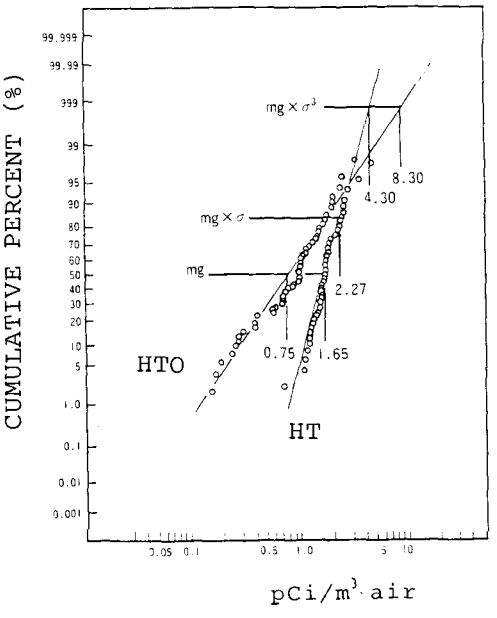

Fig.6 Measured HTO and HT Concentration plotted according to an assumed log-normal distribution

DETERMINATION OF ^{127}I AND ^{129}I IN ENVIRONMENTAL SAMPLES BY NEUTRON ACTIVATION ANALYSIS

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ABSTRACT

The analytical method of ^{127}I and ^{129}I in environmental samples has been studied and the background levels of these nuclides in soils, milk, atmosphere and seaweeds were measured.

The analytical method consists of separation of iodine from environmental samples by combustion, neutron irradiation, radiochemical purification of irradiated iodine by solvent extraction and gamma-ray spectrometry.

The detection limits of ^{129}I by this method were 4×10^{-7} Bq/g for dry soil, 7×10^{-6} Bq/ℓ for fresh milk, 2×10^{-8} Bq/m^3 for air and 7×10^{-8} Bq/g for fresh seaweeds, respectively. The relative standard deviation of ^{129}I Analysis in soil and milk were less than 10%.

^{129}I concentrations and atom ratios of $^{129}I/^{127}I$ in surface soil(0~5 cm depth) in Japan were from 1.1×10^{-7} Bq/g to 4.8×10^{-5} Bq/g and from 1.2×10^{-9} to 2.7×10^{-7} respectively. Other analytical results are discussed.

MONTE CARLO CALCULATION OF GAMMA RADIATION FIELD DUE TO IODINE-131 RELEASED TO THE ENVIRONMENT

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and

Susumu Minato

ABSTRACT

Properties of the radiation field due to iodine-131 distributed on the air-ground interface and in the atmosphere are evaluated by means of a one-dimensional Monte Carlo transport code for the propagation of gamma radiation in two medium geometry.

Figures 1 and 2 show the calculated data on the height distribution and on the energy and angular distibution of flux density and exposure rate from the iodine-131 source.

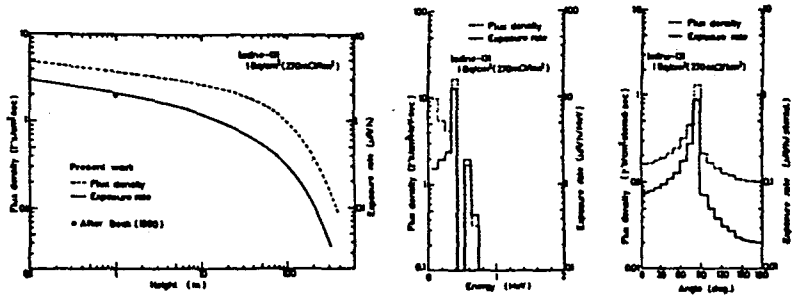

Fig.1 Height distribution and energy and angular distribution of flux density and exposure rate from I-131 infinite plane isotropic source on the air-ground interface. (1 Bq/cm²)

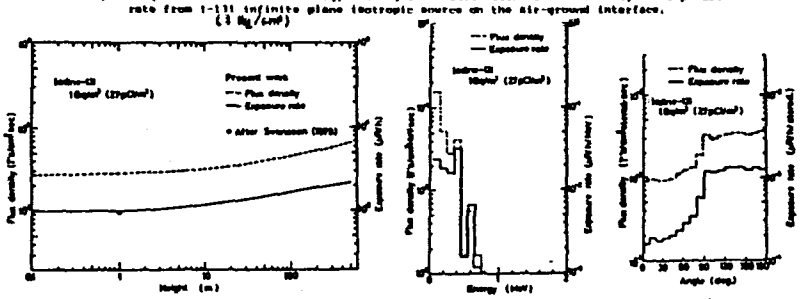

Fig.2 Height distribution and energy and angular distribution of flux density and exposure rate from uniformly distributed I-131 source in the atmosphere. (1 Bq/m³)

EVALUATION OF RISK-BENEFIT IN DIAGNOSTIC IMAGING

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INTRODUCTION

Medical radiation exposure of patients undergoing diagnostic imaging procedures accounts for around 95% of man-made radiation exposure of the population. Application of the ALARA principle to this group is therefore extremely important, and in this context some sort of quantitative evaluation of risk-benefit may prove to be useful in determining whether the benefit of a diagnostic examination (for a given group of patients) outweighs the iatrogenic effects produced by radiation: This, in fact, is a primary requisite for proceeding with the examination, as explicitly stated in I.C.R.P. Reports Nos. 26 and 34.

DECISION TREE ANALYSIS OF LIFE EXPECTANCY

The use of life expectancy in risk-benefit calculations facilitates the comparison of risk-benefit in terms of net years lost or gained. This sort of analysis can also be used to advantage in cost-benefit comparisons. In order to quantitate the net life loss or gain, the diagnostic test being evaluated must be looked at in the context of the diagnostic/therapeutic chain which both precedes and follows it. Figure 1 illustrates the type of decision tree which would typically be associated with a diagnostic imaging procedure. The consequences of each decision (1 to 4) can be computed and compared in terms of years of life expectancy (YLE) lost or gained.

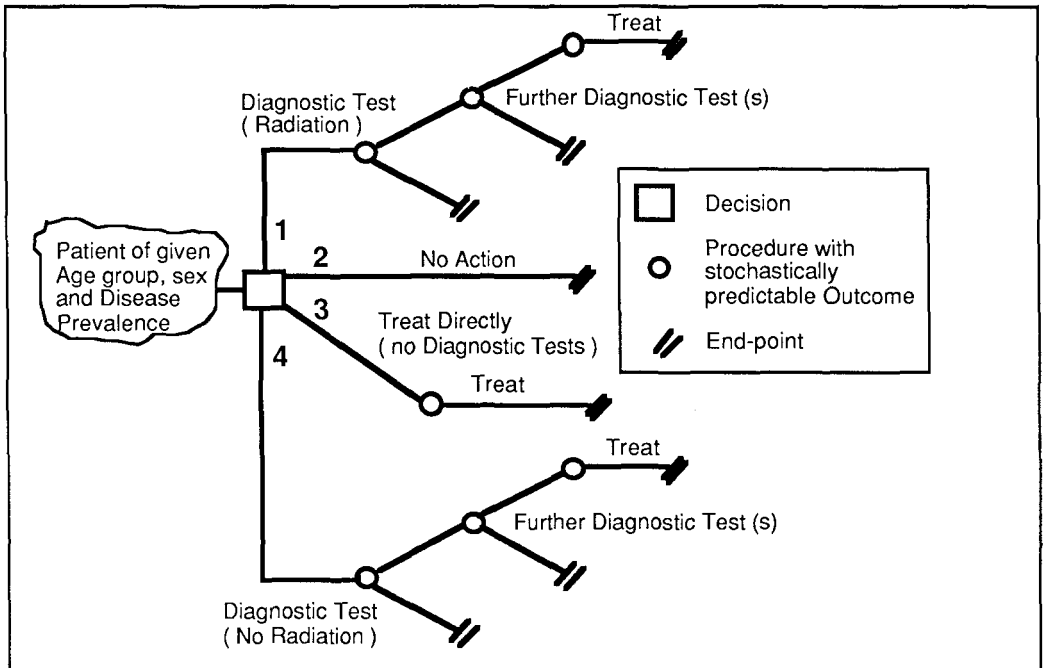

Figure 1. Simple Decision Tree relating to Patients being considered for a Diagnostic Imaging Procedure.

IMPLEMENTATION

The mathematical formulation required to compute the outcomes of the various decision tree branches is relatively straightforward, but is too lengthy to expound in detail in this short paper. Basic input data required includes the following:

* Survival tables for males and females (Australian Bureau of Statistics Life Tables, 1980-82).
* Radiation risk data (BEIR Report 3, 1980).
* Prevalence of disease being investigated.
* Survival data for diseased patients who are untreated.
* Treatment details (risk and survival data)
* Diagnostic test details (sensitivity, specificity and radiation dose).

RESULTS

Obviously, in terms of YLE lost, the effect of radiation risk will be much greater for younger patients than for older patients. Figure 2 shows the radiation risk for males and females as a function of age.

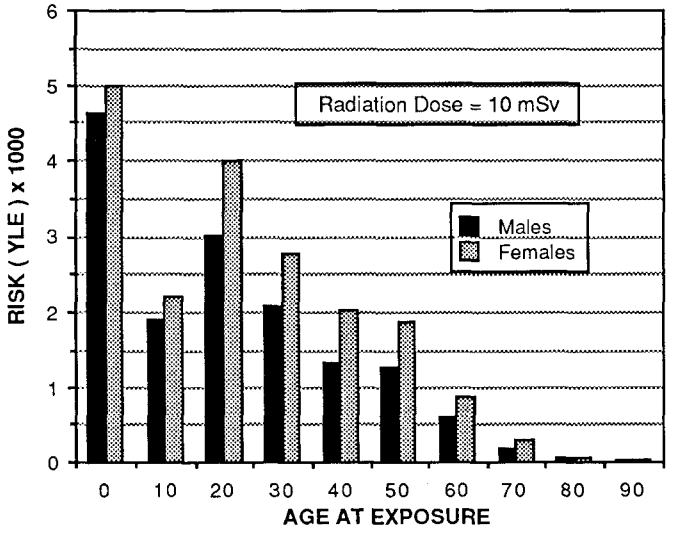

Figure 2. Radiation Risk in Terms of Years of Life Expectancy (YLE) Lost as a Function of Patient Age and Sex

Using this sort of technique, it is also possible to obtain quantitative answers to questions such as the following:

"Diagnostic imaging procedure A has a sensitivity of 89% and a specificity of 84% and for the detection of a given disease, and does NOT involve ionizing radiation. Diagnostic imaging procedure B has a slightly improved sensitivity of 91% and a specificity of 86%, but delivers an effective dose equivalent of 10 millisieverts . Which is the best test to use (ignoring cost considerations)?"

Some further information is required before an answer can be given, this being as follows:-

- Male patient, aged 50
- Disease prevalence: 15%
- Survival for diseased patients (untreated): 50% at 5 years.
- Survival for diseased patients (treated): 100% at 5 years.

Although the above data is for a hypothetical diagnostic examination and treatment, the parameters chosen are not at all unrealistic or atypical.

The answer to this question, perhaps surprisingly, is that imaging procedure B is best (2% improvement in YLE is obtained) despite the relatively high radiation dose. Even if the sensitivity and specificity of procedure A were increased to 90% and 85% respectively, the effective dose equivalent of procedure B would have to exceed 50 millisieverts in order to annul the benefit of procedure B's 1% increase in sensitivity and specificity .

The effect of radiation dose on the net benefit of a diagnostic procedure is illustrated inFigure 3. The radiation detriment is, relatively speaking, much more significant for examinations with poor diagnostic efficiency. At 10 millisieverts, for example, radiation detriment has reduced the potential net gain of the better imaging procedures by around 6%, whereas, for the less efficient procedures this figure is nearer 13%. Since most diagnostic procedures entail effective dose equivalents of a few millisieverts or so, it is evident from Figure 3 that the diagnostic utility of an examination would have to be very poor in order to exclude it because of radiation dose to the patient.

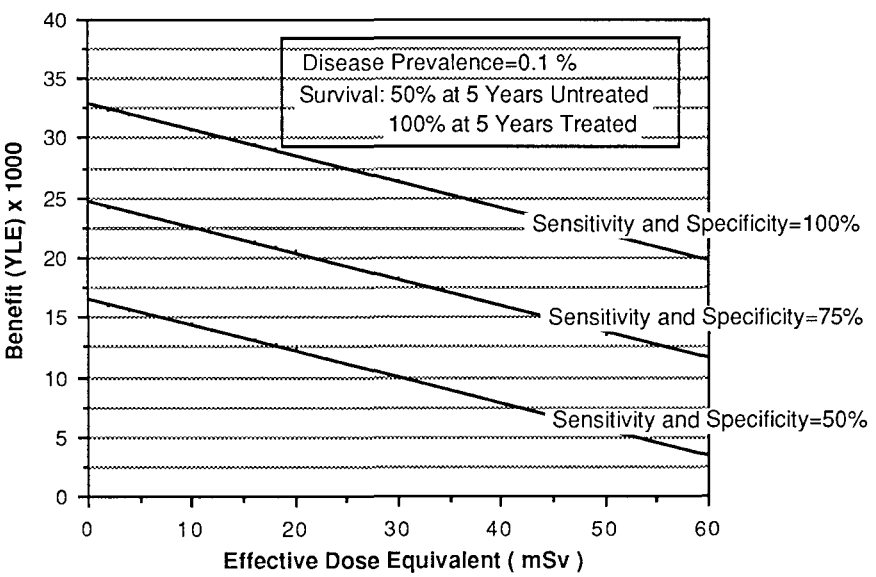

Figure 3. Effect of radiation dose on net benefit

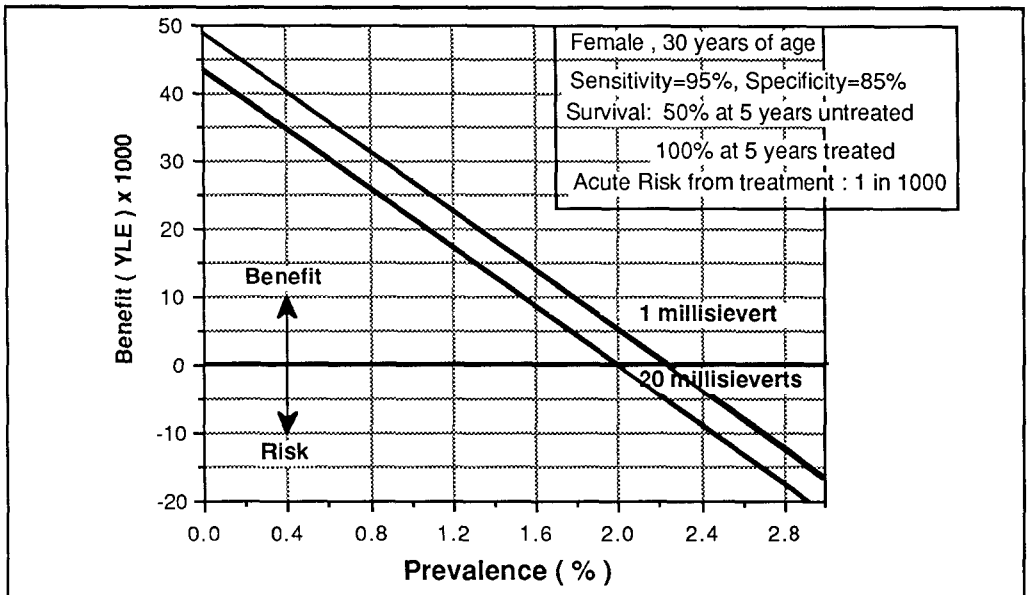

Figure 4. Effect of disease prevalence on risk-benefit

Figure 4 shows the effect of disease prevalence on the risk-benefit of a typical diagnostic procedure. Details of the diagnostic procedure and test procedure are shown on the Figure. It is quite evident that the question as to whether or not there is a net benefit to the patient is highly dependent on disease prevalence. Note also that there may be a relatively narrow "band" of disease prevalence for which the diagnostic examination is useful, and that this band narrows as the radiation detriment increases.

Finally, this type of technique can be used to evaluate screening procedures such as mammography. The input data used for this calculation (drawn from various sources) was as follows:-

Sensitivity of Mammography = 91%

Specificity of Mammography = 100%

Prevalence of Breast Cancer = 0.18%

Survival: 60% at 5 years if untreated

90% at 10 years if treated

Radiation Dose Equivalent = 10 mSv to female breast

Effective Dose Equivalent = 3 mSv.

Using the above data a Benefit : Risk ratio of 85 : 1 and a net benefit of 0.039 years per patient was computed. This sounds small, but it should be remembered this figure will, as with most screening procedures, be comprised of a large benefit to a small number of diseased patients and a very small risk to a large number of non-diseased patients. Put differently, this means that if 10,000 patients are screened, 390 person-years will be saved. Whether or not this is worthwhile may also be a matter of cost-benefit.

NEW TRENDS AFFECTING SWEDISH PATIENT DOSES FROM DIAGNOSTIC PROCEDURES

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SUMMARY

The number of medical X-ray examinations increased each year until around 1970, then decreased, but apparently increases again after 1980. The dose per examination has gone down since the early 1970ies, probably by around 50 %. In nuclear medicine, the number of examinations trebled from 1963 to 1983, when the trend changed to a decrease. The dose per examination declines, mainly due to substitution of Tc-99m for I-131. The number of dental X-ray films increased fivefold from 1963 to 1983, but the dose per examination has dropped perhaps even faster. The net effect of these trends is a reduction of the collective dose to patients from say 9 000 manSv in the early 1970ies, to say 5 000 manSv in the mid-1980ies.

The technical and medical changes behind these observations are bigger than the collective dose change. The results provide a platform for policy decisions: Is further dose reduction warranted? Does the examination serve its purpose? How do we optimize doses and image quality?

MEDICAL X-RAY EXAMINATIONS

Medical X-ray examinations contribute the biggest part by far of the collective dose from diagnostic procedures. Table 1 shows that after a steady increase, their number levelled off from around 1970. Statistics for the entire country (1) are not available after 1980, but data from the county of Stockholm (2) indicate that the number of examinations is now on the rise again. The fall-off during the 1970ies depends on the introduction of alternative modalities. The cause of the recent increase is not clear. In 1986, the National Board of Health and Welfare recommended mammography screening. Nationwide adoption of this will of course lead to many new examinations, but this could not affect statistics before 1986.

Table 1. Number of X-ray examinations, x 10^3

Year:	1950	1960	1965	1971	1974	1977	1980	1983	1985
Entire country	950	2 266	2 908	3 884	3 904	3 684	3 576	?	?
County of Stockholm	?	?	?	914	955	892	896	941	1 022

While the number of examinations increased, doses per examination decreased. This is in part due to a changed selection of examinations. Discontinuance of routine lung photofluoroscopy, e.g., saves some 200 manSv per year. But probably, the most im-

portant change lies in better technique, in terms of more careful work as well as new methods such as faster film-screen combinations.

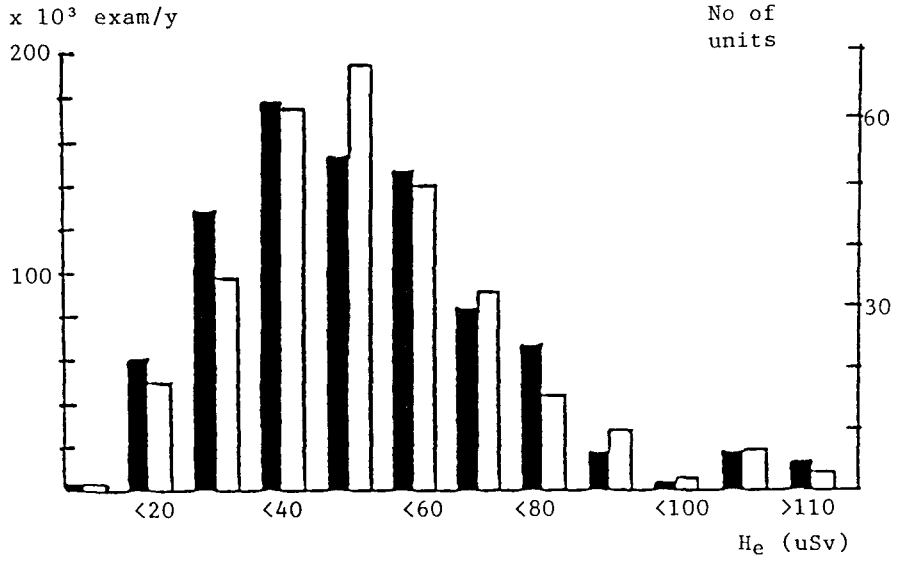
Table 2 compares the energy imparted in some X-ray examinations in Sweden 1974 (3), England 1984 (4) and a Swedish county 1986 (5). For full-size chest examinations, the energy imparted 1984/86 -and hence the dose- are about a quarter of the Swedish 1974 values. The same proportion applies to the other examinations in England, while the average appears to be say 40-50 % in Sweden.

Table 2. Energy imparted to the patient in four routine types of X-ray examination, in mJ

Examination	Sweden, 1974	England, 1984	County of Halland, Sweden 1986
Thoracic spine	210	55	58
Abdomen	200	57	145
Lumbar spine	410	124	167
Chest	21	4.4	7.4

We have also made a survey of all full-size chest units in Sweden. This was designed in collaboration with the US Center for Devices and Radiological Health on basis of their NEXT programme, and CDRH kindly lent us advanced phantoms and other measuring equipment. We do not only look at doses: image quality is tested, individual units are optimized, recurrent problems are attacked in dedicated projects, and the relevance of the use of the unit is considered. Many parameters are registered, among them dose, the distribution of which is shown in Figure 1.

Figure 1. Distribution of effective dose equivalent in full-size chest examinations (black = no. of examinations, white = no. of units), posterior-anterior view



The mean dose is 0.05 mSv, i.e. about a quarter of the 1974 mean of 0.22 mSv. Thus, the Halland data in Table 2 appear representative. If this is true for all four examinations, an average dose reduction by ½ could perhaps be assumed. The 1974 collective dose was c 8 000 manSv (3). With provision for the recently increased number of examinations (Table 1) and discontinued lung photofluoroscopy, the collective dose would now be c 4 500 manSv. Nationwide mammography screening will add perhaps 200 manSv.

EXAMINATIONS IN NUCLEAR MEDICINE

In Sweden, radiopharmaceuticals are only used in public health care. The number of examinations and activities used must be reported to us. Table 3 shows that the number of examinations rose steadily until 1983 and then fell off slightly. The biggest increase concerned Tc-99m in various kinds of scintigraphy. The Tc-99m trends are not monotonical. Bone scintigraphy no longer increases since 1984, liver scintigraphy (where Tc-99m supplanted Au-198 in the early 1970ies) is now replaced by ultrasound, brain examinations are now made with CT scans, and so on. I-131 decreases strongly, which reduces the collective dose. It is still used for function studies, but for thyroid scintigraphy and renography it is replaced by Tc-99m and in recent years I-123. Ultrasound tends to replace I-125 for renal localization. In renal clearance studies Cr-51 finds increasing use. Tl-201 is increasingly used in heart scintigraphy.

Table 3. Number of examinations in nuclear medicine, and proportions used of important nuclides

Year	No of exam:s x 10^3	Per cent of examinations done with Tc-99m	I-131	I-125	I-123	Au-198	Cr-51	Tl-201
1968	48	5	59	10	-	6	0.4	-
1971	67	20	56	11	-	2	1	-
1974	96	38	38	10	-	0.3	3	-
1977	114	50	28	8	-	0.2	5	0.7
1980	125	59	20	8	0.2	-	6	0.8
1983	132	64	14	6	0.5	-	8	1.6
1986	120	64	13	3	0.9	-	10	3.2

The radiological impact of these examinations is still dominated by I-131, which contributed 51 % to the collective dose. Tc-99m contributed 38 %, Tl-201 6 % and all other nuclides together contributed 5 %. The total collective dose was 720 manSv in 1973 (including I-131 profiles), but in spite of the much bigger number of examinations it was only 420 manSv in 1986 (excluding profiles).

DENTAL X-RAY EXAMINATIONS

A Dental Health Insurance Act introduced in 1974 stipulated X-ray examinations before all expensive treatments. This and other causes have dramatically increased the number of exposed intraoral

films, from 3 million per year in the early 1960ies to 15.7 million 1983 (6). But vast technical improvements and more stringent radiation protection have led to even more marked dose reductions. Here, the collective dose in the early 1970ies is rather loosely estimated as 300 manSv. A thorough estimation for 1983, based on our data from 400 randomly chosen dental offices, shows a collective dose of about 80 manSv (7).

TOTAL IMPACT OF DIAGNOSTIC PROCEDURES

Table 4 summarizes these results, and shows that we believe the collective dose to patients to have decreased by 40-50 % in c 15 years. Now, we find it more and more important to take all factors such as image quality, not only dose, into account. Our aspiration is not to minimize doses, but to optimize processes where radiation is involved.

Table 4. Collective effective dose equivalent in manSv to patients and average per caput dose in mSv from diagnostic procedures in Sweden

Source	Early 1970ies	Middle 1980ies
Medical X-ray examinations	8 000	4 500
Nuclear medicine examinations	700	420
Dental X-ray examinations	300	80
Total, manSv	c 9 000	c 5 000
Average dose per caput, mSv	1.1	0.6

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ANALYSIS OF RADIATION EXPOSURE AND CONTROL IN THE CARDIAC CATHETERIZATION LABORATORY

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ABSTRACT

One of the highest groups of occupationally exposed personnel in medicine are those associated with cardiac catheterization. This paper reports on our experience and evaluation of the various radiological health aspects during cardiac catheterization. Dosimetric data for faculty, residents, fellows, nurses, and technicians are reported with a temporal analysis of mR/study and average fluoro time per study. Analysis of time and motion studies evaluates what procedures or circumstances lead to higher personnel exposure. Typical isodose profiles are presented, along with an evaluation of patient thickness versus scatter photon flux. Spectral analysis of the scatter radiation is presented, along with recommendations for effective shielding of personnel.

COMPOSITIONAL QUALIFICATION OF RADIATION PROTECTION IN NEUTRON RADIOTHERAPY ROOM WITH 50 MeV CYCLOTRON

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ABSTRACT

For the independent hospital use of radiotherapy facility, a 50 MeV cyclotron had been installed in the Korea Cancer Center Hospital, Seoul, Korea, for the purpose of neutron therapy of the cancer patients, which had been started on November 1986, as well as production of short half life radioisotopes, It is clear that radiation protection must be required more attention in the hospital than in the non-hospital institute because there are a plenty of people unlimited to enter the facility who are unoriented and untrained in radiation protection, i.e. patients and relatives.

The authors analyzed radiation safety of the neutron treatment room, machine control area and patient waiting section in KCCH-cyclotron as follows;

1) Leakage mapping of the neutron gantry room
2) Measurement of gamma activation of equipments including patient couch, collimator of gantry and additional devices necessary for the treatment
3) Detection of air activation in the gantry room after neutron exposure
4) Measurement of T-1/2 and T-1/10 of clearing time of room radioactivity of gamma and neutron
5) Measurement of total accumulated exposure to the personnel for an appropriate time
6) Detection of patient exposure rate on the body out of treatment field and tissue activation within primary treatment region to identify unuseful irradiation to normal organ of patient
7) Finally, discussion of compositional qualification of radiation safety of the treatment room area and conclusion of adequate arrangement to the personnel and patients.

DETERMINATION OF THE COLLECTIVE RADIATION DOSE TO CERTAIN ORGANS IN PATIENTS UNDERGOING X-RAY EXAMINATIONS

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ABSTRACT

The organ doses from 7 types of diagnostic examination have been measured at 3 hospitals in Metro Manila for assessing the collective radiation dose. A total of 73 patients were investigated and the average individual dose to some organs was calculated. Thermoluminiscent dosimeters, consisting of Lithium Flouride (TLD-100) ribbons contained in plastic sachets, were used for the measurements. They were attached to patients to monitor the organ dose directly or at the entrance and exit skin dose at the level of the organ in question. From the entrance and exit dose measurements, the organ dose was calculated using such data as tabulated absorption data, ICRP reference man for organ parameters and available data on "Filipino Standard Man". Using the values taken from ICRP-26 and UNSCEAR 1977 Report, the risk data for radiation induced fatal cancer cases following the the examination was determined.

The results indicate that for most types of examination, the lateral projection showed a higher dose to the organs compared with PA projection. For chest examination for example, the thyroid dose was measured to be around three times higher for the lateral projection as compared to PA projection. Intravenous pyelography and upper gastro-intestinal series resulted in higher doses to the organs studied as compared with other examinations. Based on the annual number of different types of examination, the number of various types of examination needed to cause one fatal case was calculated.

RADIOLOGICAL DOSE TO THE CHILD DURING CARDIAC CATHETERIZATION (AXIAL RADIOLOGICAL PROJECTIONS)

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The use of axial angiography techniques for hemodynamic diagnosis considerably affects the levels of radiation dose to the pediatric patient under examination. Recently, several assessment procedures of radiation dose were carried in the literature, nevertheless many doubts still remain concerning the techniques proposed and the dose values obtained for the different examined organs. Particularly the results obtained by using thermoluminescent dosimeters applied to the patient's body are not convincing, because in these procedures the detectors may easily move away, without possibility of control, from the radiation beam area, thus supplying incomplete and fragmentary results. The aim of this study is to ascertain the dosage levels for the child undergoing cardiac catheterization with the use of axial radiological techniques, in order to define the relationship between fluoroscopic and angiographic dosages and to critically evaluate the possibility and the levels of irradiation of critical organs, as a function of radiological beam positioning.

MATERIALS, METHODS AND RESULTS

The research was carried out in the Hemodynamics Laboratory of the Cardiology Institute of the University of Bologna with a total of 50 cardiac catheterization performed by using axial techniques on patients whose age ranged from 1 day to 11 years, mean value 4 years.

The laboratory, where examinations were performed, is equipped with biplane CGR Radiological System. When in PA projection, the image intensifier is in the upper position. Each radiation tube is provided with a screening system which allows to confine the radiation field to the minimum useful surface. Rx-I.I. distance amounts to 105 cm, both for PA and LL projections. The radiological system is equipped with an automatic control device which acts by changing the voltage applied to Rx tubes. Current and graphic emission time are manually set.

The research protocol was divided in two phases:

1) characterization of the radiological system from the patient's radioprotection standpoint;
2) recording of radiological data and emission times during examination to calculate the dosages for the individual patients.

The characterization of the radiological system was performed by using IONEX-type dosimeter with 0.6 cm^3 ionization chamber to record the dose in air, 15 cm from I.I. entrance screen, in correlation with the voltage and the current supplied to the radiological tube, both in fluoroscopic and angiographic conditions. The results are illustrated in Fig.1. For each diagnostic examination taken into consideration, recordings were made of radiological data and time in fluoroscopy of various projections with the relevant radiological data and time in angiography, of

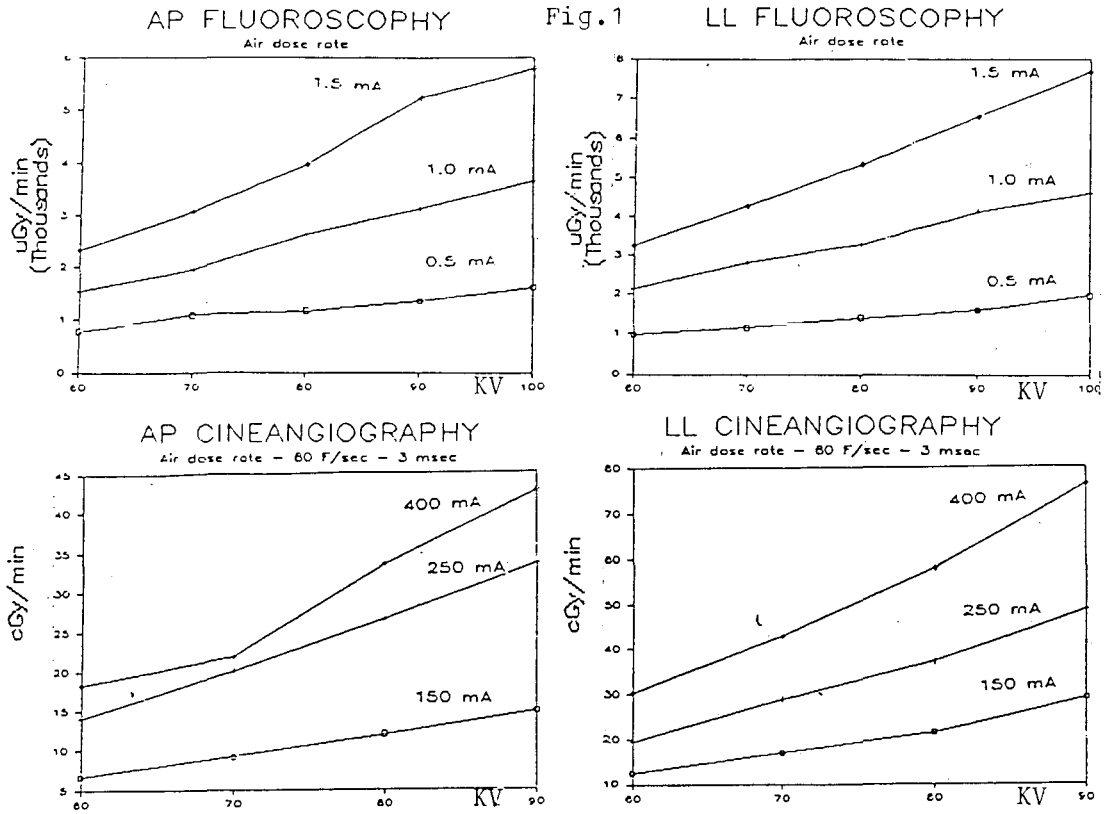


Fig.1

age, weight, and height of the patient. Thus, by using Fig.1 curves,it was possible to assess the fluoroscopic and angiographic dose to the skin of the patient for each single projection. Fluoroscopic emission time ranges from 0,7 to 58,8 minutes (M=22,1 min) and fluoroscopic air dose for PA ranges from 0,4 to 7 cGy (M=3,7 cGy). Angiographic emission time, for PA and LL projections respectively ranges from 14 to 107 sec (M=36,5 sec) and from 5 to 87 sec (M=30,8 sec).Angiographic air dose ranges from 3,5 to 107 cGy (M=55,3 cGy) for PA and from 2,6 to 165 cGY (M=83,8 cGy) for LL projections. The results of the fluoroscopic tests for LL projection are not recorded, as they are negligible if compared to PA results. Fig. 2 illustrates the dose distributions for the individual patients.

DISCUSSION

The curves of Fig.1 confirm the good performance of the radiological equipment. Fluoroscopic information (time and radiological data) may be compared with those recently published in the literature. The angiographic time (number of frames/examination), although considerably changing case by case, is on average similar to that reported in the literature, whereas resulting air dose values (23 cGy (PA), 29 cGy (LL)) are particularly high and, anyhow, at lest 10 times higher than those recently presented. Our results point out that there is at least a 1/10 ratio between fluoroscopic and angiographic air dose and, hence, they suggest that the cineangiographic phase is the most dangerous for the

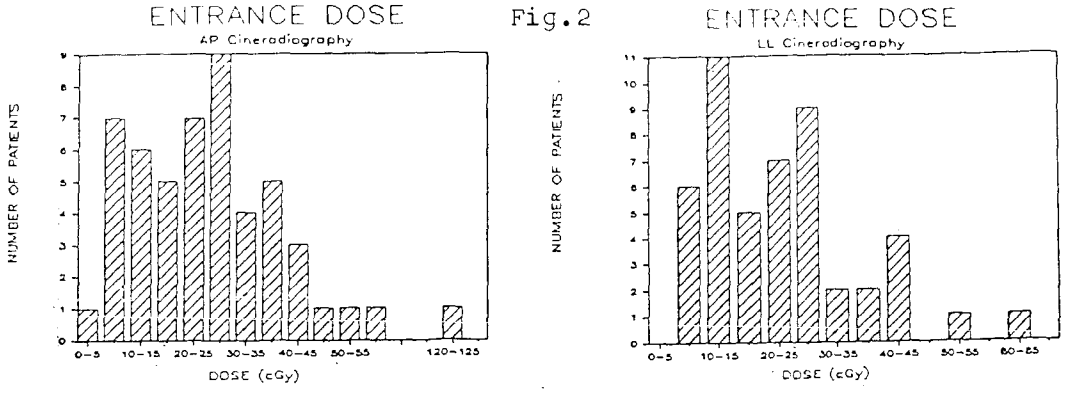


Fig.2

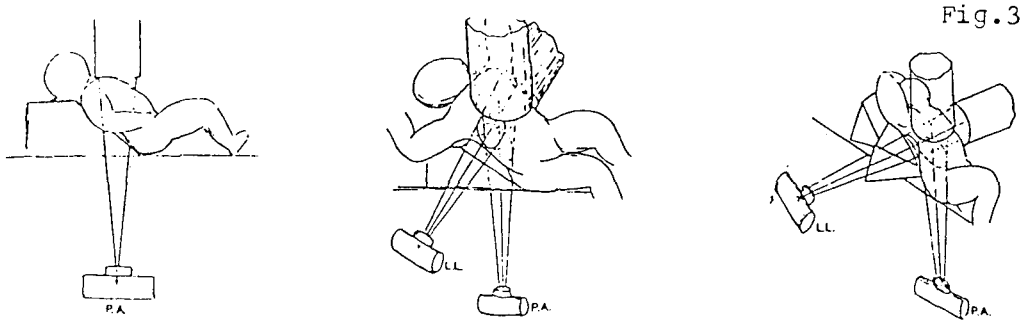
patient from the protection standpoint. In our opinion, the low graphic air dose values quoted in the literature are due to the difficulties met in keeping the thermoluminescent dosimeters within the radiation beams when the patient is moved around or variously positioned for the different projections. This problem arises only during the angiographic phase, as in the fluoroscopic one the patient remains in PA position and the dosimeters are certainly within the radiologic beam. This explains why the fluoroscopic values are in agreement.

Considering our results, we decided rather to give separate air dose figures for PA and LL than to add them up, as is usually done, because a careful analysis of the examination procedures and the different projections (Fig.3) very seldom reports cutaneous overlapping of the irradiation fields of the two tubes. Therefore, we prefer to consider a larger area exposed to a smaller dose than to overestimate the exposure at any possible point. Nevertheless, it is extremely important to take account of the very different results, on the other hand expected in this kind of diagnosis, to pinpoint and discuss the cases of abnormal irradiation of the patient. This analysis is possible thanks to the method we adopted, based on the recording of radiological data and emission time. Furthermore, with reference to the cases under examination, for all air doses over 43,5 cGy,the analysis highlighted the technical causes (misadjustment or breakdown of the radiological system) and the operational causes (projection repetition) which brought about the abnormal value.

Observing the diagnostic medical procedure, the recorded data point out a systematic use (49 out of 51 cases) of the biplane radiological recording. A greater attention in estimating the real usefulness of a biplane observation in each projection could possibly lead to a decrease in the use of LL projection and a remarkable dose reduction.

The use of "axial" cineangiography modifies the risk of irradiation of important organs, both for the different positioning of the beams, which are at times closer to them, and for the greater energies employed to penetrate thicker layers of tissue. The evaluations reported in the literature cannot be compared whatsoever, both because of the different technical performances of the system and because of the different medical procedurte management (seriography, cineangiography, etc.). By observing the images in Fig.3, it is clear that, when the patient is tilt up, the radiation beam gets closer to the thyroid gland and to the gonads but, for the thyroid, it cannot be higher than

Fig.3



the I.I. entrance air dose (i.e. 0,26 uGy/frame). In a standard examination, including about 2000 frames for each projection, the thyroid air dose should therefore be less than 104 uGy. Considerably higher values indicate bad operation of the system, wrong placing of the tubes and inadequate screening.

CONCLUSIONS

For the pediatric patient undergoing cardiac catheterization, the radiological dose is one of the parameters which have to be carefully considered as a risk, expecially with reference to the possibility of repetition of the examination or integration with less dangerous techniques (ultrasound). Comparing fluoroscopic and angiographic dose values, the latter show greater irradiation for the patient. This leads to a critical evaluation of the methods of examination and, expecially, to consider the advisability of systematic biplane radiological recordings. The proposed methods, in the framework of which parameters strictly connected to the technical performances of the system and to the operational examination procedure are regularly collected, permits a critical analysis of the results and highlights the causes of abnormal irradiation of the patient. The radiation of important organs (thyroid, gland, gonads) may be evaluated both by analysing the beam angle and by using the data obtained as input parameters of mathematical models which take into consideration the patient's soma and the beam bearing. In any case, the results show a smaller irradiation than that expounded in the literature.

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RADIATION EXPOSURES TO PATIENTS DURING CARDIAC ANGIOGRAPHY AND CORONARY ANGIOPLASTY

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Of all medical radiological procedures currently undertaken, it has been well established that those which deliver the greatest radiation dose to the patient are associated with imaging of the heart; more specifically during coronary artery angiography and angioplasty. It is important, therefore, to measure the doses to patients associated with such procedures, as a preliminary step towards a programme aimed at dose reduction.

The method used in this study involved the mounting of a large area parallel plate ionization chamber, known as a Diamentor Chamber (PTW Freiburg), on the x-ray collimator housing. Such chambers were mounted on two machines, one being a Siemens Tridoros 5S, which employs an under-table tube, while the other is a Siemens Polydoros, employing a C-arm. The Diamentor system measures the exposure-area product emanating from the x-ray source. The skin dose to the patient was determined from this using a Capintec 192 Exposure Meter with a 28 ml air-equivalent chamber. The range of tube potentials for all the measurements was 60-110 kVp, while the frame rate during cine was 48/second for the under-table tube and 25/second for the C-arm.

The measurements obtained, which are illustrated in the following figures, were classified in accordance with the type of cardiac procedure performed, namely: Left and Right Coronary Angiography, Left Ventriculography and Coronary Angioplasty. For each procedure, the exposures obtained during the fluoroscopy and cine stages were combined. The relatively small dose received during catheter insertion is combined with that of Left Coronary Angiography, given that it is usually the first study performed in nearly all cases. The figures illustrating total dose per patient excludes Angioplasty and pacemaker insertion, but includes Right Heart Catheterization, Electrophysiological and Ergotamine studies, where they were performed in a small number of instances, along with the other studies, A total of 12 Cardiologists participated throughout this study.

It is evident from this study that some patients are receiving relatively high doses from cardiac catheterization procedures, especially in the case of coronary angioplasty. It is, therefore, imperative to implement means of reducing the total dose to patients undergoing such procedures, while maintaining an acceptable image quality.

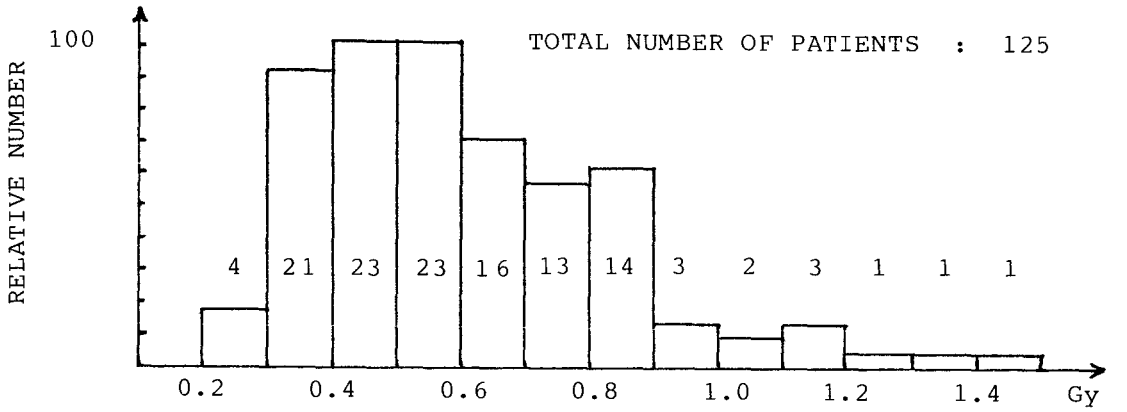

Fig. 1. Skin dose distribution for Left Coronary Angiography procedures, using an under-table tube.

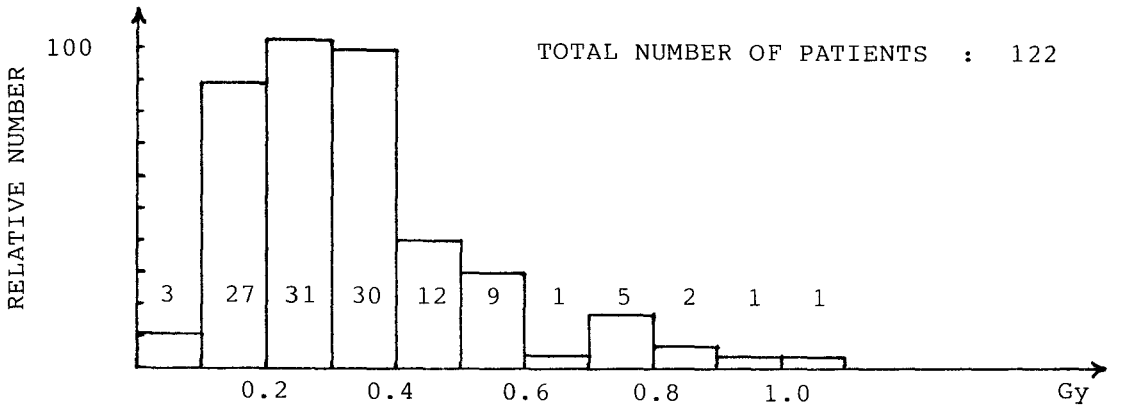

Fig. 2. Skin dose distribution for Right Coronary Angiography procedures, using an under-table tube.

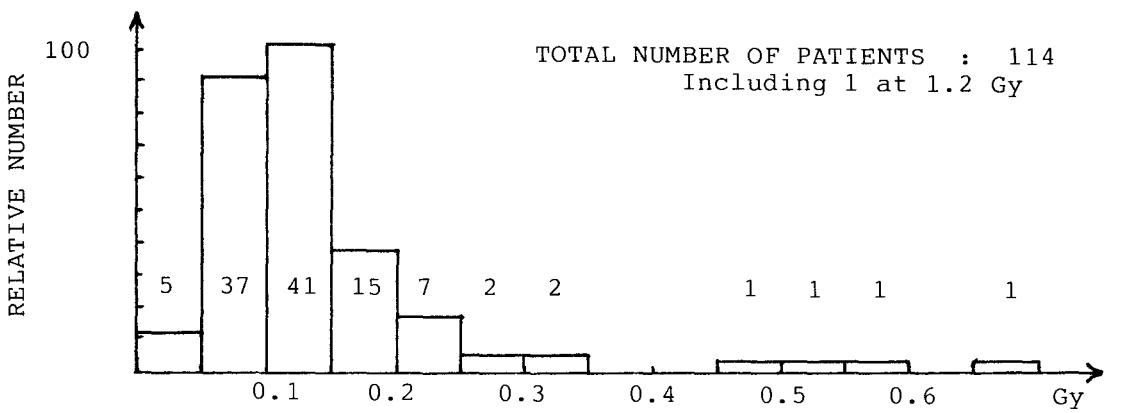

Fig. 3. Skin dose distribution for Left Ventriculography procedures, using an under-table tube.

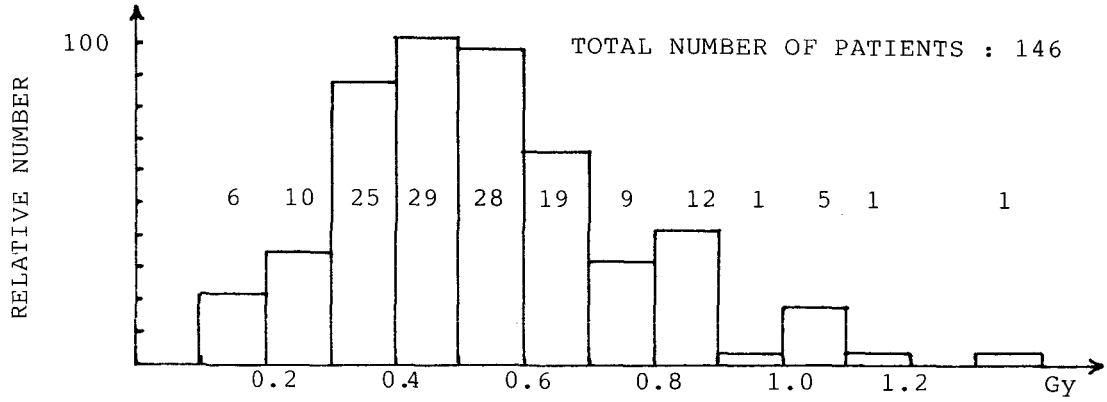

Fig. 4. Skin dose distribution for Left Coronary Angiography procedures, using a C-arm.

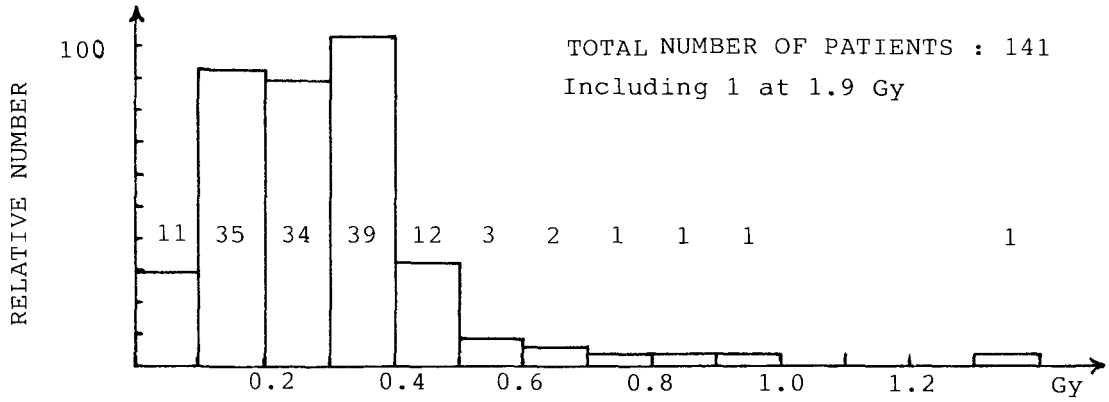

Fig. 5. Skin dose distribution for Right Coronary Angiography procedures, using a C-arm.

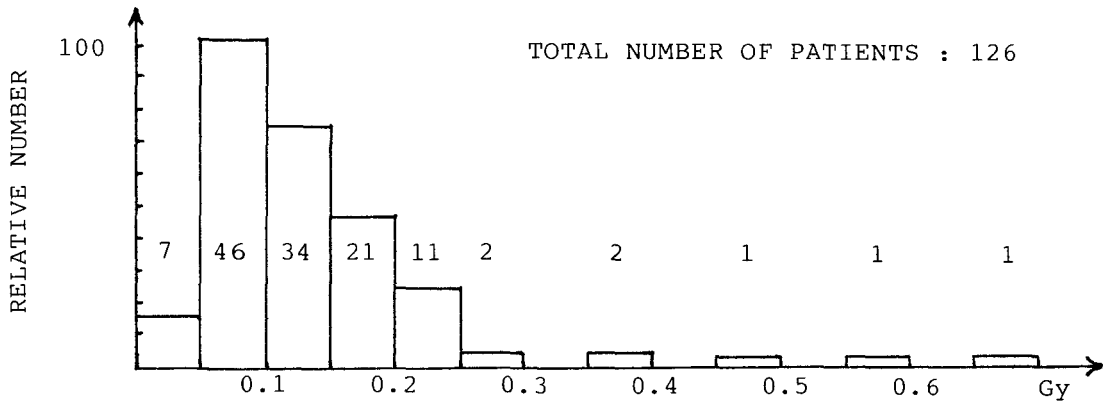

Fig. 6. Skin dose distribution for Left Ventriculography procedures, using a C-arm.

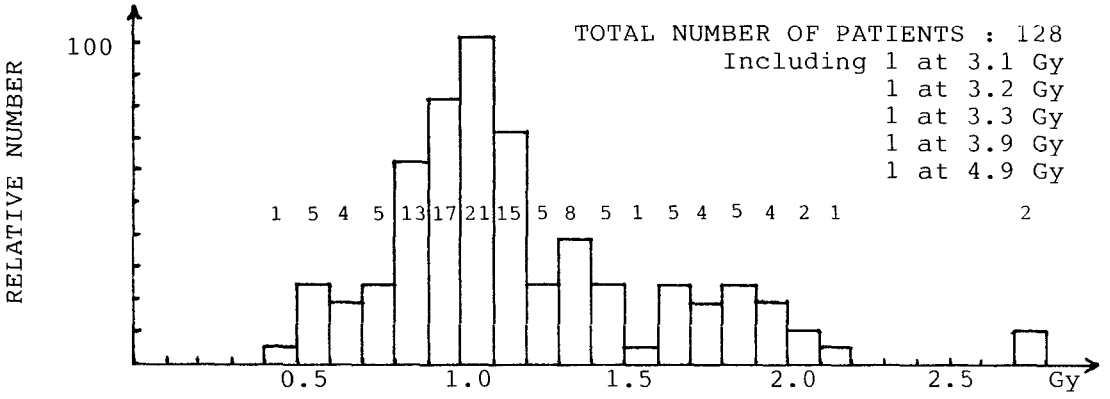

Fig. 7. Total skin dose distribution for all procedures, per patient, using an under-table tube.

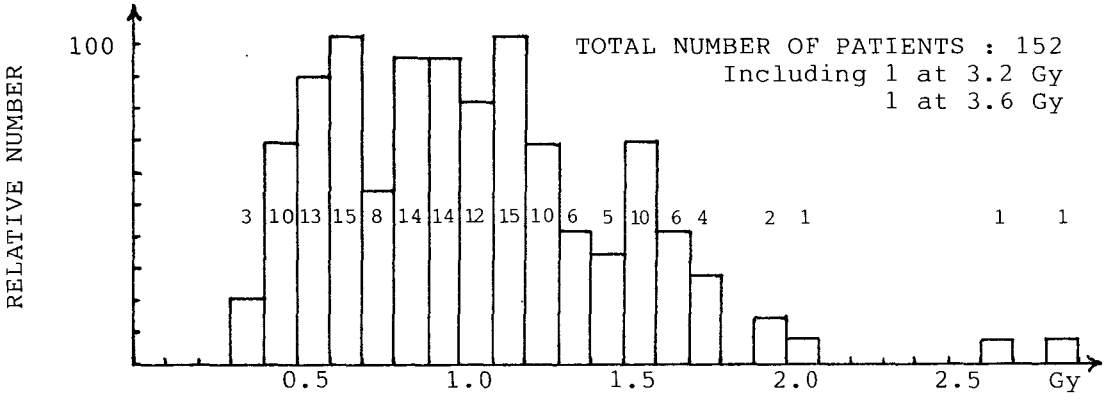

Fig. 8. Total skin dose distribution for all procedures, per patient, using a C-arm.

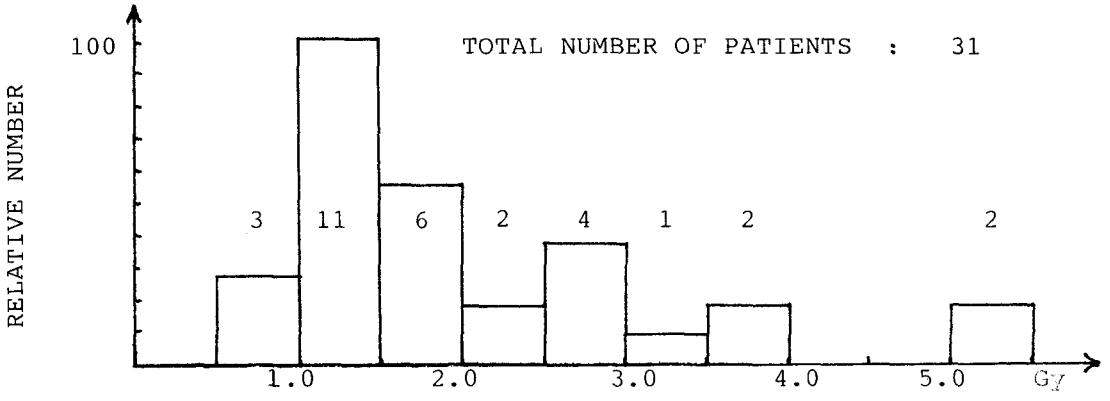

Fig. 9. Skin dose distribution for Coronary Angioplasty procedures, using a C-arm.

ASSESSMENT OF RADIATION EXPOSURES IN BI-PLANE C-ARM PEDIATRIC CARDIAC STUDIES

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ABSTRACT

Radiation exposures to patients and staff during cardiac catheterization has been investigated previously. However, to date, most studies assessed the amount and/or patterns of radiation exposures during examinations of adult patients. Few reports were on the subject of children's radiation exposure while undergoing the same procedure. Due to continuing improvement of modern-day neonatal care capabilities, more and more children as well as infants are being examined using catheterization procedures. In the past few years, bi-plane C-arm device have become the de-facto radiological system for children' catheterization studies. Children can be sedated and maintained in one position on the examination table throughout the entire procedure. By manoeuvring the C-arms, flouroscopic or cine images in various projections can be easily obtained without much, or any, patient movement or cooperation. Objective of our project was to assess the range and average amount of radiation exposures to child patients undergoing various cardiac radiological examinations. Measurements were made on phantoms as well as patients. Radiation exposures to different size phantoms were measured under routine protocols. Results were then compared with that obtained on patients under clinical environments. All measurements were made at entrance and/or exit skin level. Thermoluminecent dosimeters were used as the primary measuring device complemented with a flat chamber x-ray monitor. The results indicated a wild range of values, e.g. for anterior chest wall it ranged from 15 mSv to 94 mSv per examination. Correlation of radiation exposures to patient size, examination type and technique factors were attempted. Scattered radiation patterns which determines the amount of staff exposure were also investigated. Results in this aspect were in agreement with previous reports. Dependent upon the C-arm orientation and other factors, the radiation exposure rate to staff during the radiation-on period ranged from minimum to more than 800 mR per hour at various locations within the examination room.

RADIATION EXPOSURE DURING CARDIAC CATHETERIZATION PROCEDURES

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INTRODUCTION

For some time there has been an increased interest in more information about radiation exposure during cardiac catheterization because of: (i) relatively high doses to workers and patient, (ii) rapid increase of numbers of examinations, (iii) introduction of new procedure-types (e.g. Percutaneous Transluminal Coronary Angiography, PTCA) and (iiii) introduction of new techniques (e.g. Digital Subtraction Angiography, DSA).

This paper reports about a study on the exposure to medical personnel and patient in two major hospitals[*] [**] in the Netherlands. The total number of cardiac catheterization procedures in both hospitals amounts to circa 3000 per year (approximately 10% of all cardiac procedures c.q. 20% of all PTCA procedures in the Netherlands). This study is related to 1300 cardiac examinations.

CARDIAC CATHETERIZATION PROTOCOL

The catheterization team usually consists of one cardiologist performing the catheterization, one sterile assistant, one or two circulating assistants and personnel behind lead glass walls. Regularly, a cardiologist trainee is involved too. Sterile assistance is performed by a member of staff of the catheterization laboratory (hospital A) or by a cardiologist (hospital B). In hospital A, the sterile assistant stays opposite of the investigator on the left side of the patient. In hospital B, investigator and sterile assistant stay both on the right side of the patient. All personnel in the vicinity of the patient wear aprons (0.5 mm lead equivalent). Thyroid collars and lead eyeglasses are not used. In hospital B, most of the time a lead apron, attached to a framework, is positioned aside the patient table between cardiologist and patient. Protective shields are not used in hospital A. Hemiaxial views are obtained with rotation and angulation of the C arm in hospital A and with rotation of the U arm and table movements in hospital B. Usually, catheterizations are performed with a femoral approach. Characteristics about these hospitals are given in table 1.

DOSIMETRY

Individual dosimetry was carried out during circa 1000 procedures in hospital A and 250 procedures in hospital B. Combinations of filmbadge dosemeters and TLD dosemeters were worn by cardiologists and staff on the forehead, on the collar (outside apron), on the sternum (inside apron), the unshielded back, the tibia and wrists and indexfingers of both hands. All dosemeters were replaced every two weeks. In total 2000 dosemeters were distributed. Job dosimetry was performed with electronic personal pocket dosemeters worn in the breast pocket of lead aprons during 120 procedures in hospital A.

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[*] Hospital A: Catharina Hospital, Dept. of Cardiology, Eindhoven (NL 5602 ZA)
[**] Hospital B: Academic Hospital Maastricht, Dept. of Cardiology (NL 6201 BX)

Ambient dosimetry was carried out with integrating dosemeters, placed in 10 different positions in each laboratory. An antropomorph phantom was used in several series of exposure rate measurements in the laboratories during fluoroscopy and cineangiography in simulated cardiac procedures.

Using a Diamentor (PTW Freiburg) with a flat transmission ionisation chamber attached to the X-ray tube housing, independent measurements were made of the exposure-area product XAP (expressed in R•cm²) during both fluoroscopy and cineangiography in 200 examinations.

In addition, data were recorded about procedure type, team composition, patient data (weight, length and sex) and fluoroscopy time and cineangiography.

TABLE 1 DESCRIPTIVE DATA OF THE DEPARTMENTS OF CARDIOLOGY

	hospital A	hospital B
cardiologists/trainees	12	8
members of staff	9	12
procedures (1986):		
= total	2000	1200
= CAG	950	700
= PTCA	750	40
= EPS	-	250
= DSA	-	30
radiodiagnostic equipment	Philips Poly Diagnost; Maximus M200 gen.; C-arm with parallellogram support	Siemens Cardioscoop Pandoras generator U-arm
image intensifier	6½ and 9 inch	6½ and 9 inch
cine filmspeed	50 frames/s	50 and 25 frames/s

RESULTS: PATIENT EXPOSURE

Dose assessments for patients can be based on exposure-area product XAP measured with a Diamentor system [NR86]. Measurement results of XAP are given in table 2. The mean XAP for all cardiac procedures is circa 6100 R•cm². The relative contribution of fluoroscopy amounts to circa 40% per examination. The mean XAP-value per unit time is 3.8 R•cm²/s during fluoroscopy (range 0.4 to 9.3). During cineangiography at 50 frames/s, mean XAP per unit time is 63 R•cm²/s. (range 26 to 134). Differences in XAP-values per unit time during CAG and PTCA were statistically not significant.

TABLE 2 EXPOSURE-AREA PRODUCT XAP [R•cm²] PER EXAMINATION

	all procedures			CAG			PTCA		
	fluor.	cine	total	fluor.	cine	total	fluor.	cine	total
mean	2800	3500	6100	1900	3300	5000	3400	3700	6800
cv (%)	100	55	70	100	53	68	93	56	64
minimum	120	430	400	120	540	700	280	450	1200
1st quar.	820	2200	3300	560	2200	3000	1300	2300	3800
median	1900	3000	4800	1200	3000	4200	2300	3100	5500
3rd quar.	3500	4600	7700	2500	4100	6700	4300	4600	8900
95th %	9900	7300	15200	7500	6800	12700	10800	8100	17300
maximum	13800	8900	24000	8700	8900	16500	12000	8800	18800
sample size	169	170	209	79	81	98	71	70	84

Linear regression analysis of the measurement results showed that XAP per patient can be approximated with a formula in which fluoroscopy time is expressed in minutes and filmlength in meters (regression coefficient = 0.9):

$$XAP = 255 \bullet \text{Fluoroscopy time} + 60 \bullet \text{Filmlength} \quad (1)$$

Statistics about fluoroscopy time and filmlength during circa 1300 examinations are given in table 3. Mean fluoroscopy time during PTCA is about twice as long as during CAG-procedures. The difference in filmlength is negligible. Using these recorded data about fluoroscopy and cineangiography the distribution of XAP was calculated for 1300 procedures. This distribution is not significantly different from the distribution of measured XAP values (table 2).

Analysis of variance showed that differences in fluoroscopy and cine time between individual cardiologists in hospital B was not statistically significant. On the other hand, these differences were quite large in hospital A. During CAG mean fluoroscopy time varied between 5 and 16 min; mean cine time between 47 and 69 seconds. During PTCA mean fluoroscopy time ranged from 10 to 24 min and mean cine time from 50 to 67 seconds for different cardiologist.

TABLE 3 FLUOROSCOPY TIME AND CINE FILMLENGTH

	Fluoroscopy time (min) Hospital A			Hospital B		Filmlength (meters*) Hospital A			Hospital B	
	all	CAG	PTCA	all	CAG	all	CAG	PTCA	all	CAG
mean	11	7.7	15	10	7.7	59	57	61	49	48
s.d.	10	7.5	11	9	5.3	24	21	27	16	19
1st quartile	4.3	3.4	7.6	5	4	45	46	45	40	40
median	8.0	5.3	12	7	6	56	55	57	48	45
3rd quartile	15	9.3	19	12	9	70	68	72	50	50
95th percentile	33	23	36	30	18	101	90	110	75	90
maximum	66	59	66	63	42	243	216	243	120	120
sample size	1041	503	388	226	120	1031	502	387	124	79

*: Using a 35 mm camera at a filmspeed of 50 frames/s, one second of cine angiography corresponds with a filmlength of circa 1 meter.

RESULTS: OCCUPATIONAL EXPOSURE

Using General Linear Model procedures, dosimetry measurements were related to working conditions. Personnel dose data were fitted with XAP-values, totalized per monitoring period. XAP-values were calculated with formula (1) from fluoroscopy and cineangiography data about those examinations in which each person participated. Resulting estimates of organ dose equivalents for workers are presented in table 4 (normalized to XAP = 1000 R•cm^2). Jobdosimetry showed that mean dose equivalents of the cardiologist's eyes varied between 10 and 50 microsievert per 1000 R•cm^2 for different cardiologists.

Typical mean values for the dose equivalent of the eyelens for CAG procedures are 0.2 mSv for investigators and circa 0.05 mSv for assistants. During PTCA procedures, these estimates are circa 25% higher.

Using filmbadge dosemeter data the overall effective energy of scattered radiation in vicinity of patient was estimated at 35 keV.

TABLE 4 ORGAN DOSE EQUIVALENT TO WORKERS [microsievert] PER EXPOSURE-AREA PRODUCT XAP OF 1000 R•cm²

tissue	hospital A			hospital B		
	I	II	III	I	II	III
eye	30	10	10	20	10	3
thyroid	25	5	5	20	7	2
trunk (front)	-	-	-	-	-	-
trunk (back)	1	0.5	0.8	-	-	-
tibia	30	20	2	35	20	3
left hand	25	5	5	15	10	3
right hand	5	7	5	7	7	3

I = investigator ; II = sterile assistant ; III = circulating assistant

DISCUSSION

Using the method of dosimetry index described by Huyskens [Hu88], personnel dose data were evaluated. In hospital A the dosimetry index for investigators varied between 30 and 50% ; for members of staff of the catheterization laboratory between 10 and 20%. Due to a lower workload, the dosimetry index for cardiologists in hospital B was less than 25%. For members of staff in hospital B, the dosimetry index was only a few percent. For cardiologists and staff the dosimetry index for the eye is higher than any other dosimetry index. Using dosimetry index for classification of working conditions, it is concluded that cardiac catheterization procedures are to be classified as working condition A.

Calculated XAPs were compared with XAP-values of some common types of X-ray diagnostic examinations as reported by NRPB [NR86]. Ratios of mean XAP for cardiac procedures and mean XAP for diagnostic procedures are: chest 120; abdomen 10; lumbar spine 5 and barium enema 1.5.

The smaller amount of filmlength in hospital B is a result of a lower filmspeed during coronary angiography recordings (25 frames/s instead of 50 frames/s) [Ja85]. However, the expected reduction is partly compensated by taking more cine recordings per examination.
This study confirmed that during LAO-views (rotating the X-ray tube towards the investigator) radiation exposure to the investigator is much higher (10 times or more) than during RAO-views. LAO-views should be replaced as much as possible by RAO-views.
It was concluded that lead aprons of 0.35 mm lead equivalent, protecting a larger part of the body are to be preferred.

This study confirms that the practice of cardiac catheterization needs adequate individual dosimetry and intensive radiation protection managment.

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THE EFFECT OF THE CHERNOBYL ACCIDENT ON THE NUMBER OF RADIOGRAPHIC EXAMINATIONS IN GREECE

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ABSTRACT

The radioactive cloud of the Chernobyl accident reached Greece on May 2, 1986. Public anxiety, with many sings of panic, started on the night of May 5/6, when the first protective measures were officially announced, and lasted for about two months. Many people imposed to themselves, their families and particularly their children, additional countermeasures which included indoor staying, consumption of canned food only, intake of stable iodine etc. and avoided exposure to medical radiation, by refusing medically indicated x-rays. Since this latter "countermeasure" can be measured, we have checked the number of roentgenologic examinations of the Athens Children Hospital and of several General Hospitals over Greece, during the period May and June 1986 in comparison to the same months of the years 1984 and 1985.

The results showed that a significant decrease of the number of x-rays examinations was observed. This decrease was more marked in the Children Hospital than in the General Hospitals and in both cases was more marked for out-patients than for in-patients.

It is concluded that the Chernobyl panic effect decreased the overall medical uses of radiation and particularly this effect was more marked for the examinations taken in the more radiosensitive part of the population, namely the children, while it was less marked for the x-rays taken in the people in more need of medical attention, that is the in-patients, who also were under closer conduct with their doctors, and therefore probably they were influenced by them.

HIGH CONTRAST IMAGES FOR THE VERIFICATION AND DOCUMENTATION OF TREATMENT FIELDS DURING TELETHERAPY

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Dep. Radiotherapy

SUMMARY

High contrast images are necessary for the control and subsequent correction of the treatment area before radiotherapy begins, and to document the actual treatment field during radiotherapy itself. Several different combinations of film enclosed by metal foils were investigated to record the radiation field position. It was found that most commercially available types of X-ray diagnostic films especially those with a high gamma were suitable for verification of the field position (field localization). To document the field position during radiotherapy (field documentation) an other film with a characteristically coloured film base to differentiate this image from other types of image was prefered. For the foil, both copper and steel, enabled high contrast images from photons with energies of 1 to 15 MeV.

INTRODUCTION

In teletherapy, individual formed irradiation fields using irregular shields and blocks are routine. It is therefore necessary to be able to check the position of the radiation treatment beam and location of the shielding blocks with respect to the patients anatomy. Two separate types of image are necessary. Firstly an image is needed just before the start of radiation process (field localisation). It is taken with the X- or gamma radiation of the therapy machine. One exposure is given with the shielding blocks and the field size set as though the patient were to be really treated; the other exposure is given with a large open field and the schielding blocks removed.

Secondly an image is obtained during the irradiation process itself to document the actual treatment area (field documentation). This image is also a document of the actual radiation treatment and can be used to evaluate localisation erros which may occur during treatment or to provide information for any subsequent queries about the treatment procedure.

FOIL MATERIAL

Imaging with high energy X- or gamma radiation is a widely used nondestructive testing method for inspection of thick or metallic objects. Several types of image receptors used in nondestructive testings have therefor been used for treatment field imaging. However, the most commonly used technique employs a special radiographic film, often in ready pack, to use with or without lead screens of various recommended thicknesses.

Although lead has been recommended as a suitable foil material its use has limitations too. Its interaction with photons of lower energy, which are produced when the primary phontons are scattered in the patient, is much stronger than with the higher energy primary photons (Fig.1). This leads to images with an inherently poor definition. In contrast, steel or copper have generally a reduced amplification of photons especially for those with lower energies and therefor produce images with a higher definition. Steel has the additional advantage of an harder surface than copper thus being of better use for the daily routine work.

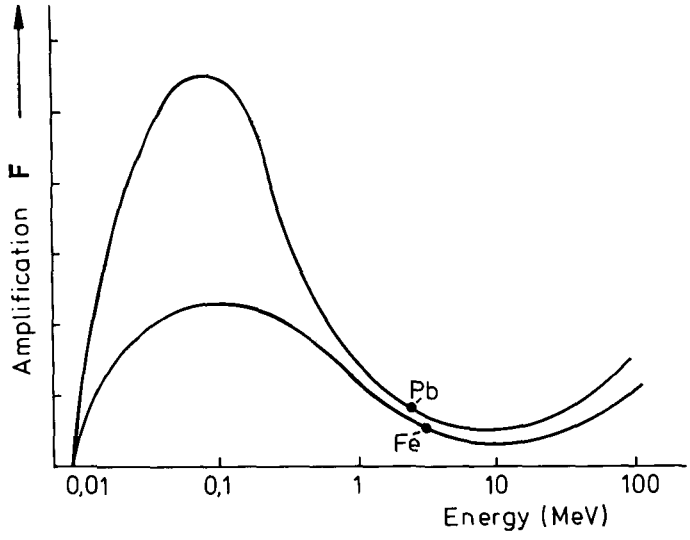

Figure 1:
Relation between relative image intensification and photon energy of lead and iron foils (taken from Beck 1982). Note the greater intensification when low energy photons (secondary scattered photons) are incident on the lead foil.

FOIL THICKNESS

In contrast to X-ray diagnosis, which uses an image intensifier foil that absorbs primary photons and emits secondary photons with a specific energy, the metal foil used during radiotherapy, produces compton electrons to interact with the film. The optimum intensification of the image occurs when the thickness of the foil is a little thinner than the mean free path of the compton electrons in the foil. Therefore the thickness of the foil must be matched to the energy of the primary photons. Practically a 1 mm thick foil in front of the film is suitabel up to 4 MeV-photons whereas for higher energy photons a 2 mm thick foil is necessary.

In addition to a metal foil in front of the film, there is also a metal foil at the back of the film which serves to back-scatter electrons which have passed through the film. The backscattering effect increases with the increase of the foil thickness up to a maximum (approx. 0.5 mm). Greater thicknesses do not offer any additional benefit.

Rear screens do not improve the contrast but there is a benefit in speed. The selection of metal screen materials and thicknesses are also influenced by practical considerations f. i. cassette weight, durability and the speed of the film being used.

FILM

To produce field localization images any diagnostic X-ray film may be used. However to reach best results films with especially high Gamma are needed. Several different and commercially available films were tested. Outstanding results were obtained using the Cronex 7 film (Du Pont).

To produce documentation images of field position during radiotherapy treatment a film with reduced speed wich can be processed automatically in 90 sec. is needed. To make identifications in the daily routine easier it would be useful to mark the film base differently to the usual film base. At present, this special characteristic is only offered for DOT-1 film (Du Pont).

Using films of either a 35 cm x 43 cm or 24 cm x 30 cm format all radiotherapy fields of interest can be studied.

CONCLUSION

Both, copper and steel foils enable high contrast images in an energy region from 60Co to 15 MeV-Linac, lead is not recommanded. Steel has the advantage of being more robust, and hence being more suitable for routine use.

Influenced by the large size of 60Co and 137Cs sources and the often used short treatment distance a significant reduction in resolution and image quality is unavoidable.

To make the identification of localization and documentation film easier differently colored film bases are helpfull.

A significantly improved reproduceability of the irradiation fields was found.

The technical personal becomes more actively involved in the treatment process and therefor becomes highly motivated.

ACKNOWLEDGEMENTS

The author wishes to thank Drs. G. Emmerich and G. Demuth (Du Pont Nemours, Anwendungstechnisches Zentrum, 6078 Neu-Isenburg) for assisstance with the investigation of film and foil materials.

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RADIATION SHIELDING DESIGN AND SURVEYING OF RADIODIAGNOSTIC AND RADIOTHERAPY INSTALLATIONS BASED ON SEVERAL COMPUTER CODES

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ABSTRACT

Radiation shielding is the most important tool for radiation protection of the public and the worker around radiation installation.
The shielding design of existing irradiation facilities require the use of a computerized approach for calculating the shielding for any medical installation.

The limitation of probable genetic effects is achieved by recommendation known as the ALARA concept. This requirement implies that further shielding additions, lead to exposure reductions which are not significant when compared with the increasing costs involved - "Optimization of radiation protection".

The starting point for our computerized approach will be the NCRP 49 and NCRP 51 protocols, while adopting several modifications needed by the ALARA concept and for multiple sources of radiation.

The method described should help develop framework and uniform approach for design shielding and prevent overshielding of radiation protection.

Irradiation facilities included in the present research are:

A. X-Ray Diagnostics - Radiography + Fluoroscopy units, CT scanners, Panoramic Dental Radiography units.

B. Radiotherapy - X-Ray Therapy Apparatus (50-500 kVp) Electron Linear Accelerators, Cobalt-60 Teletherapy Machines.

C. Industrial Radiography - X-Ray: Portable Directional Apparatus ($\leq$300 Kv), Electron Linear Accelerators.

   Gammagraphy: Co-60, Ir-192, Cs-137 .

   To perform the calculations, workloads are acquired for each of machine used in the room. Each source should be considered separately because of the different parameters associated with each source at its location in the room.

INTERNAL CONTAMINATION IN NURSES ATTENDING PATIENTS, THAT RECEIVED THERAPEUTIC AMOUNTS OF RADIOIODINE-131*

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INTRODUCTION

The most frequent and often very successful used unsealed source in Nuclear Medicine and Radiotherapy is the radioiodine-131 for the treatment of thyroid carcinoma and hyperthyroidism. Always there is a great concern about the health physics of radioiodine and possible internal contamination involved in high level 131-I thyroid therapy cases, in particular to the thyroid as target and limiting organ.
This report deals with 131-I air concentrations and internal contamination in nurses attending these patients under two different conditions.
During the past three years a change took place from the old building, where we had an unventilated two-bed nursing room, to a new building where we have rooms with forced ventilation and air-conditioning (refreshment five times per hour). From both external exposure caused by radioiodine treated patients and internal contamination due to ingestion and inhalation of 131-I, we calculated the dose-equivalent to the thyroid and the effective dose-equivalent to our health care personnel.

MATERIALS AND METHODS

In our hospital radioiodine-131 is given orally as a liquid in quantities of 185-925 MBq for the treatment of hyperthyroidism (uptake range 26-89%, mean value 67%) and up to 7 GBq for the treatment of thyroidcarcinoma and metastasis. In general the patients need only low care, which results in a maximum nursing time of 10 hours/week.
After drug administration the patient is not allowed to leave the room for three days on an average, until both the acceptable level of total body activity is reached (less than 370 MBq) and the fast excretion phase is over.
In both the old and new housing about 140 patients are treated yearly with 131-I for therapy, together with 65 patients diagnosted with amounts of 37 to 185 MBq. All health care personnel involved is classified as radiological worker category A, wearing personal filmdosimeters and as a standard undergoing periodically thyroid activity measurement. In this department the job is done by 20 colleagues.
Following ICRP-30 the annual limit on intake of 131-I amounts 1 MBq by ingestion and 2 MBq by inhalation, we use an investigation level amounting to 1/10 of the ingestion value.

*Acknowledgement:

This work was supported by the Dutch Ministry of VROM, Directory Radiation Protection.

MEASUREMENTS

In this study airborne 131-I contamination was measured with the Herfurth Air Sample Collector (H-1351N), intake at 2 metres above the floor of the two-bed nursing rooms.
Both volatile iodine and aerosols were measured periodically during the first 3 hours, 24 hours and patients' total residence time after drug administration. From the measurements we learned that the 131-I activity mainly consists of volatile iodine compounds or very low AMAD particles. Internal contamination control in both nursing personnel and administrators was performed with the Whole Body Counter (WBC) of the Institute for Radiopathology and Radiation Protection (J.A. Cohen Institute, Leiden, the Netherlands). This WBC consisted of four NaI-detectors Ø 4,75 x 4 inch placed in a measurement room having a wall thickness of 10 cm aged-lead. The sensitivity of the system amounts 30 Bq of 131-I for a 30 min counting time.

RESULTS

The degree of air contamination during above mentioned periods was correlated to both the administered activity and the excreted activity; in all cases the coefficient of correlation is bad ($< 0{,}750$). Table 1 deals with the 131-I concentrations in both the unventilated and the conditioned room at a mean administered activity of 3,8 GBq per patient.

Table 1. 131-I concentration in air in the therapy room.

Period	Unventilated room		Conditioned room	
	range Bq/m^3	mean value Bq/m^3	range Bq/m^3	mean value Bq/m^3
3 h	44 - 1428	400,7	2,2 - 114	43,6
24 h	3,9 - 737	236,6	0,7 - 92	24,2
total residence time	1,9 - 585	105,2	0,3 - 53	15,1

The distribution of the internal contamination of nursing personnel is given in figure 2, from which are calculated the mean and median values of this activity. From these data we learn that, in the new housing, this nursing personnel has a significant higher (although low) degree of contamination, probably due to little experience in contamination risk and unfamiliarity with this treatment.
Figure 3 shows the internal contamination in persons who dispense and administer the oral radioiodine quantity.

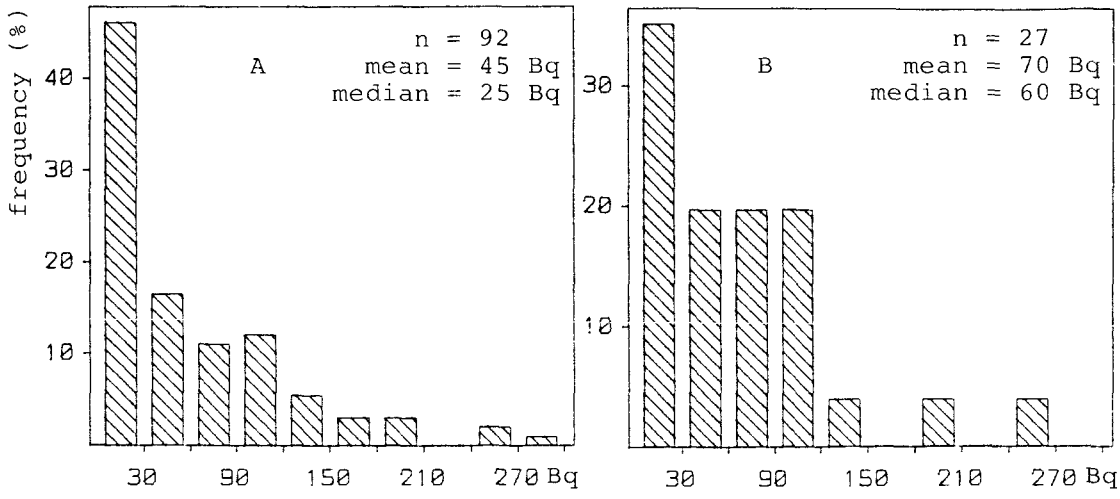

Figure 2 : Distribution of the internal contamination of nursing personnel in A: unventilated therapy room and B: the conditioned room.

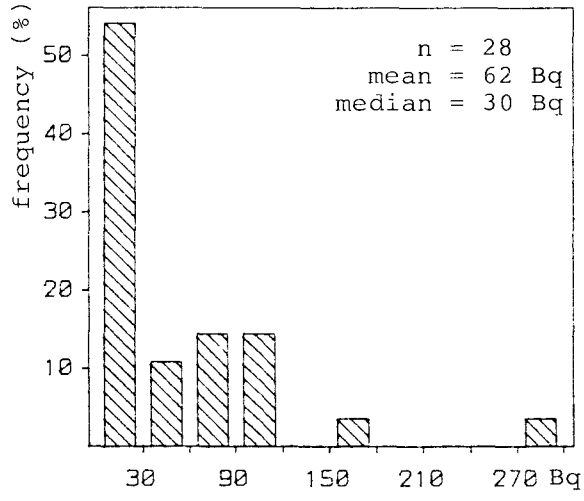

Figure 3 :

Distribution of the internal contamination in personnel dispensing and administering the liquid iodine-131 therapeutic quantities.

DISCUSSION

In case of nursing personnel we calculated, from the elapsed time between possible intake and WBC-measurements, a mean weekly intake of about 60 Bq I-131. From the airmeasurements, and the maximum total residence time of the personnel in the nursing rooms, we learned that most of the internal contamination can be ascribed to ingestion for lack of discipline; pointing out that the sanitary facilities are highly contaminated in this unit.
Following ICRP-30 the mean dose equivalent to the thyroid is 4,5 man.mSv/y, resulting in an effective dose equivalent of 135 man-μSv/y. Personal dosimeters indicate that the dose equivalent due to external radiation does not alter this figure significantly.

As can be seen from figure 3, even workers handling the therapeutic quantity are contaminated at a low level, resulting in an effective dose equivalent of 65 man.μSv/y, however this work is done by only a few men.
From the mean values of air-contamination and amount of refreshment together with the residence time of the patients and the volume of their room, can be calculated that the total airborne activity is about 10^{-3} of the administered quantity.
Besides that, no other waste is discharged directly to the environment, because we have decay-tanks and a decay-facility for solid waste.

CONCLUSION

In the present situation it is possible to nurse patients, having a low iodine uptake, treated with quantities up to 7 GBq without a significant radiation exposure to the health care personnel. However, it is necessary that the working conditions are judged by a health physicist, the discipline is near perfect and monitoring is performed on a regular basis.
Under these conditions, refering to the relative low air-contamination, even moderate care patients may be treated, mainly resulting in a higher external exposure of the personnel.

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"S" -ABSORBED DOSE PER UNIT CUMULATED ACTIVITY FOR SELECTED RADIONUCLIDES IN ORGANS OF PEDIATRIC MODELS

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ABSTRACT

The Center for Devices and Radiological Health of the U. S. Food and Drug Administration and the nuclear medicine community have been aware that size, structure, and functional differences between pediatric and adult patients influences absorbed radiation doses. The knowledge of pediatric radiation doses has been limited because of the lack of suitable mathematical models and the knowledge of the biodistribution of radionuclides in children.

Using a series of pediatric phantoms, explicit mathematical expressions (S-factors) have been calculated accounting for age differences in organ masses, size, shape, and position as determined from anatomical references including distribution of active marrow. As a result a number of changes in the S-factors differing significantly from those previously published for the adult have been developed. Models for the neonate, one, five, ten and fifteen-year old will be presented along with a description of differences in the S-factor data. Nuclear decay data have been updated in the revised calculations.

In addition, the S-factors have been tabulated for some 23 radionuclides, selected based on their actual or potential use in diagnostic nuclear medicine. An examination of these data reveals differences in absorbed doses between the ages ranging from two to ten for some emitters. When compared to the adult the pediatric organ doses are all greater than experienced by the adult. Clearly the earlier practice of applying adult data to pediatric patients resulted in underestimates of the doses to children.

PATIENT EXPOSURE IN GENERAL DENTAL PRACTICE IN THE NETHERLANDS.

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INTRODUCTION

Today radiology has become an important factor in dental diagnosis. Although the risk of a single radiograph is low if properly conducted, the frequency of these radiographs is rather high and still increasing.

To estimate the population risk due to dental radiography an investigation was started among 1200 dental practitioners. A questionnaire was set up to inventory commonly applied indications of X-ray examinations, the number of examinations and the organizational actions taken by the dentists to limit radiation doses to the patients. Information was gathered on the type of X-ray machines, the use of aiming devices, protective measurements for patients and dental staff, developing procedures as well as the type of films.

A number of practical tests was applied to obtain a quantitive impression of patient doses in accordance with special circumstances. For the practical tests films and lithium fluoride TLD-100 chips (Harshaw) were used to determine the beamdiameter, the exposure of the X-ray machine and the scatter at a set distance of the middle of the beam, developing circumstances as well as entrance and exit skindoses measured on the skin of a patient.

The results of 544 dental practices will be discussed. Finally an estimation of the possible extent of reduction in patient exposure in the Netherlands will be made.

RESULTS

Nearly 80% of the X-ray machines used were made by Philips, operating at 45, 50 or 65 kVp. Siemens X-ray machines were used in 7% of the dental offices while other manufactures were represented in less than 5% of the practices.

45% of the X-ray machines were operating at 65 kVp, 41% at 50 kVp. The kilovoltage used ranged from 45 to 90 kVp. X-ray machines of 65 kVp or more were all provided with open end cones. Still 30% of the X-ray machines were provided with circular pointed cones.

The mean exposure per radiograph of all X-ray machines was 142×10^{-6} C/kg. For X-ray machines operating at 50 kVp the exposure was 201×10^{-6} C/kg, for 65 kVp 81×10^{-6} C/kg. The mean entrance skin dose was 4.83 mGy, for X-ray machines of 50 kVp 6.70 mGy and of 65 kVp 2.80 mGy. X-ray machines provided with a circular pointed cone caused an exposure of 245×10^{-6} C/kg in comparison with 98×10^{-6} C/kg for open end cones. The entrance skin doses for these different types of cones were 8.35 mGy and 3.30 mGy.

At a fixed distance of the middle of the beam scatter was measured only when circular pointed cones were used. The mean scatter measered caused an additional entrance skindose of 0.20 mGy. No scatter was measured for open end cones.

Tabel 1: Mean exposure and entrance skindose for X-ray machines of 45 till 90 kVp.

kVp	Exposure C/kg	s.d.	s.e.	Skindose mGy	s.d.	s.e.
	$\times 10^{-6}$					
Mean	142	147	7	4.83	4.83	0.24
45	394	174	71	16.22	7.41	3.02
50	201	180	14	6.71	5.71	0.44
56	278	160	80	9.16	6.37	3.19
60	200	211	56	6.17	4.66	1.25
65	81	59	4	2.80	2.18	0.16
70	73	62	10	2.53	1.82	0.29
90	51	-	-	2.27	-	-

In 48% of the dental practices Kodak Ektaspeed films were used. The mean exposure measured in those practices was 100×10^{-6} C/kg instead of 178×10^{-6} C/kg in the other practices. The entrance skin doses were 3.23 mGy and 6.14 mGy. Theoretically a reduction of 40% in exposure by using Ektaspeed films instead of Ultraspeed films was expected. The reduction was even higher because of other influences: dentists using Ektaspeed films had more often X-ray machines operating at higher kVp's and had more often X-ray machines provided with open end cones.

The mean number of patients was 2055 per practice. The mean number of X-rays taken was 748 each year during 427 X-ray examinations: 1.75 exposures per examination. Approximately 20% of the patients were examined radiographically each year.

65% of the dentists made 2 bitewing radiographs of new patients routinely. In nearly 60% of all practices these bitewing radiographs were repeated at set intervals, more than 90% within 3 years.

Most radiographs were made of patients between 20 an 30 years of age (35%) and between 11 and 20 years of age (32%).

In 81 % of the dental practices radiographs were automatically processed while in 18% films were developed manually. Dentists using automatic film processing procedures caused less exposure and skindoses than other dentists. The low mean exposure in this group was caused by other influences: dentists using automatic film processing procedures had more often X-ray machines operating at higher kVp's, more often X-ray machines provided with open end cones and they used more often automatic timers than was used in the total group. No differences were found between both procedures if the results were corrected for the influences as mentioned above .

The mean exposure was 135×10^{-6} C/kg for automatic film processing and 220×10^{-6} C/kg for all kinds of manually film processing. The mean entrance skin dose for both kinds of processing was 4.34 mGy and 6.90 mGy.

The films used to control film processing indicated that the mean density of the X-ray films was less than films developed in a standardized way. More than 60% of the dentists could improve their developing procedures.

Lead aprons and thyroid collars reduce the effective dose-equivalent. The influences of these protective measures couldn't be measured directly during bitewing radiography with the LiF TLD-100 on the cheek of the patient. In the sample 54% of the dentists had a lead apron, 15% a lead collar. In 38% of the dental practices these protective measures were used during all exposures for every patient.

The use of aiming devices can cause a reduction in patient exposure of 40% if the beam is shielded. These reductions can not be measured during bitewing radiography by the measuring technique used in this study. In 8 % of the dental practices aiming devices provided with an extra diafragm were used.

DISCUSSION AND CONCLUSIONS

If all X-ray machines operating at 50 kVp provided with circular pointed or open end cones are replaced by X-ray machines operating at 65 kVp, a reduction in exposure of 58% is possible in 41% of the dental practices reducing the mean exposure to:

$$(1 - 0.58 \times 0.41) \times 142 \times 10^{-6} = 108 \times 10^{-6} \text{ C/kg}$$

If Kodak Ektaspeed films are used in all practices a reduction in exposure of 40% in 52% of the dental practices will result in a decrease of exposure to:

$$(1 - 0.40 \times 0.52) \times 142 \times 10^{-6} = 112 \times 10^{-6} \text{ C/kg}$$

Filmprocessing can be improved in 60% of the dental practices. However no differences in exposure were found as far as "good" or "bad" developing procedures were compared. In 51% of the dental practices the exposure was at least 32% higher than expected to receive a good radiograph. If the over-exposure is reduced, the mean exposure will decrease to:

$$(1 - 0.32 \times 0.51) \times 142 \times 10^{-6} = 119 \times 10^{-6} \text{ C/kg}$$

Finally a total reduction in patient exposure can be estimated by multiplying all possible reduction factors:

$$0.76 \times 0.79 \times 0.84 = 0.50$$

The mean exposure for a bitewing radiograph can therefore be reduced by 50% to 71×10^{-6} C/kg. It is possible to reduce the risk due to dental radiography in the Netherlands by 50% if all measures as discussed before will be carried out.

THE DEVELOPMENT AND OPERATION OF A DENTAL MONITORING SERVICE INVOLVING THE REMOTE DETERMINATION OF X-RAY BEAM PARAMETERS.

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INTRODUCTION

In 1973 NRPB introduced a postal service for the radiological safety assessment of dental radiography equipment and procedures. Experience of survey visits to dental practices prior to this had demonstrated that improvements to dental X-ray equipment and attention to correct operating procedures could result in the significant reduction of unnecessary exposure of the patient and dental staff, and also shown that to offer a service to the large number of dental practices in the country would require a postal method operated from a central laboratory. Advantages of such a service include the ability to maintain a low unit cost per X-ray set, the creation of a central point of reference for radiological safety in dentistry having a very comprehensive background of data, and minimum disruption to the normal practice routine.

A test cassette for exposure at the tip of the director 'cone' of the dental X-ray set was developed to acquire information regarding the operating parameters, ie operating potential, total beam filtration, patient entrance dose per exposure, and X-ray beam size[1]. The cassettes use Agfa-Gevaert double emulsion radiation monitoring film to obtain the necessary dynamic range of exposure sensitivity required for accurate and reliable assessment of these parameters by means of relative transmission measurements through a range of copper filters. For each X-ray set four cassettes are despatched to the practice, to be exposed at specified 'timer' settings, together with a questionnaire relating to general radiography procedures.

During the period 1978 to 1985 the Health and Safety Executive adopted the postal service to increase effectiveness in inspecting dental practices, and to obtain information for assessing standards in relation to the Euratom Directive on Radiation Protection. Some 7,000 X-ray sets were assessed in this period and the results of the survey have been analysed to provide an indication of the general standard of dental X-ray equipment and its operation, to identify the factors affecting both patient and staff doses, and to provide the data for a cost-benefit study of the service.

SURVEY RESULTS

The equipment deficiencies, reported as a percentage of the total number of X-ray set assessments are shown in Table 1.

Table 1. Summary of equipment assessment results

Equipment deficiency	% of sample
Operating potential < 50 kV	50.4
Total beam filtration < 1.5mm Al	11
Beam diameter > 7.5cm	12
Beam diameter between 7.5 and 6.0cm	9
No X-ray warning light	6
Incorrect operation of 'timer'	13

The significant percentage of sets operating at potentials less than 50 kV reflects the large number of older imported European dental units still in use in the UK. No general regulatory measures have been introduced to restrict their use, but because of the relatively high patient entrance dose necessary to achieve normal radiographic densities with such equipment, recommendations have been made to cease using sets operating at less than 40 kV. With the current availability of faster Group E dental film, a recommendation to adopt this film, to ensure patient entrance dose is not excessive, is now made for the remaining equipment operating below 50 kV. It is our experience that optimum diagnostic results are achieved with potentials in the range 60 to 70 kV, but the survey shows only 8.8% of sets operate within this range.

Where total beam filtration was assessed as inadequate the cause was generally the omission, or loss, of any additional aluminium filter. The inherent filtration of the tube assembly lies within the range 0.7 to 1.0 mm Al equivalent giving a patient entrance dose per radiograph a factor of 2 to 1.5 greater than that for equipment having the minimum requirement for beam filtration.

The current recommended maximum exit beam diameter for dental X-ray equipment is 6.0 cm. Equipment having X-ray fields greater than this value falls into two categories, very old equipment with beam diameters in the range 9.5 to 11 cm and equipment supplied with an exit beam diameter between 7.5 and 6.0 cm, reflecting earlier code of practice guidance. The use of a rectangular aperture to collimate the X-ray beam to just greater than the film size, with consequent dose savings to the

patient and a substantial reduction of scattered radiation, was only found in 6 X-ray units from the total sample of 7,000. From experimental results of measurements of scattered radiation to an operator positioned at 1.5 metres from the patient, for a range of X-ray beam diameters, a power law equation has been derived by means of which it is possible to assess the dose savings associated with correction of the deficiencies identified above.

As well as equipment deficiencies, inadequate radiography procedures may also contribute to unnecessary patient and staff exposure. The most frequently occurring procedural changes recommended are summarised in Table 2.

Table 2 Summary of recommended procedural changes

Recommendation	% of sample
Improve processing conditions	47
Direct X-ray beam away from inadequately shielded areas	43
Reduce radiographic exposure (patient skin dose exceeds 15 mSv)	40 (5)
Do not hold film or tube housing	9.5

Contributory factors to the high incidence of excessive radiographic exposure are varied and may not always be apparent to the operator. Where the patient skin dose exceeds 15 mSv, however, the density of a correctly processed radiograph should be such that most diagnostic information is obscured and the requirement to reduce exposure should be obvious, although in practice even this is not always appreciated. In all cases the adoption of 'optimum' exposure values will not only produce a significant dose saving but will provide radiographs of much improved diagnostic value. X-ray sets having anatomically pre-programmed exposure controls should lead to the use of acceptable dose levels, however, overriding controls such as film speed control and in particular long cone/short cone settings can, if incorrectly used, lead to excessive doses.

COST BENEFIT ANALYSIS

As part of a review of the efficacy of this service as a means of enforcement by the controlling authority (HSE), a cost benefit analysis was carried out. This indicated that if all the recommendations for dose reduction were implemented by the entire dental profession, the annual savings in collective dose for staff and patient doses would be 5 and 120 man.Sv respectively. Using the then (1984) prevailing NRPB provisional costing framework, these dose savings represented detrimental cost savings totalling £300k per annum. The cost of the dental monitoring

programme was approximately £70k per year and most of the improvements were procedural or incurred little cost. Whilst the assessment had other inputs, this strongly indicated the effectiveness of the programme.

THE CURRENT PROGRAMME

The introduction of the Ionising Radiations Regulations, 1985 in the UK provided a fresh impetus to review dental practices. At this time the Health and Safety Executive took the view that the cost of surveillance should fall on the dental profession. As a result, the Dental Monitoring Service operated by the NRPB was enhanced in 1986 to encompass all aspects of radiological safety in dental practice, and is now made directly available to the dental profession.

Among the enhancements is a check on the processing of radiography films, which is frequently not carried out to a standard that achieves optimum results. It is often the case, for example, that X-ray exposure is increased to compensate for inadequate development conditions. To provide the dentist with constructive comments relating to the processing of radiographs a pre-exposed film, a radiograph of a carefully chosen copper step wedge, is sent to each practice to be processed by the normal routine procedure. This film is returned to the laboratory where the optical densities of the filter areas are compared with a reference film processed under optimum conditions. The shape of the density difference curve provides the diagnostic information to identify the cause of any sub-standard processing.

The radiological safety assessment of panoramic X-ray equipment has previously been carried out by visiting the practice. A postal cassette has now been developed which enables a complete assessment of the equipment to be made following a simple exposure by surgery staff. The cassette is designed to locate and attach to the cassette holder of the X-ray set and contains a strip of X-ray film to record the beam size and position, together with an arrangement of two radiation monitoring films with various copper filters to enable the assessment of kV, total beam filtration and dose per exposure cycle at the receiving slit. In use, the postal cassette is exposed simultaneously with a normal diagnostic film. The image on this diagnostic film indicates any non-uniformity of radiographic exposure which may be caused by mechanical or exposure timing problems.

The Dental Monitoring Service now provides a cost-effective means for the dental practitioner to obtain all the information needed to comply with statutory requirements, and maintain quality assurance checks on equipment to ensure that maximum diagnostic information is obtained with minimum radiation dose to the patient, dental surgery staff and the public.

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DOSE DISTRIBUTION IN ORAL RADIOGRAPHY

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SUMMARY

Although the patient dose due to oral radiography is not very high per examination (at least in comparison to other radiographic procedures in medicine), the frequency of these examinations is high enough to warrant further investigations of dose distributions in standardized experiments.

Dose measurements were performed by means of LiF-100 thermoluminescence dosimeters in an Alderson Rando phantom, that was specially adapted to enable comprehensive measurements in the head and neck region. The measurements refer to several dental X-ray devices, projection techniques, kV-settings, film types and measures for dose reduction. In addition, some other X-ray techniques used to show a larger area of the skull were involved in this study. The effective dose equivalent was calculated for each of the techniques and the related parameters.

The H_{eff} ranged form .002 to .660 mSv. This corresponds to a fatal risk in the order of 10^{-6} or 10^{-7} per examination. The results of this study will be used to estimate the population dose due to oral radiodiagnosis in the Netherlands in a current investigation.

INTRODUCTION

Many publications have appeared concerning dose measurements and risk estimations in oral radiography, based on in vitro and in vivo studies. These studies show a large variance due to the small beam diameter as applied in oral radiography and the complicated anatomical structures in the facial region. These factors make that a small deviation of the projection geometry results in a considerably different dose distribution in the patient.

Although the patient dose due to oral radiography is not very high per examination (at least in comparison to other radiographic procedures in medicine), the frequency of these examinations is high enough to warrant further investigations of dose distributions in standardized experiments.[2] The aim of this study is to determine the dose distribution in the body, especially in the head and neck region during radiodiagnostic procedures as usually applied in dentistry. Factors that influence the dose should be considered as well. These factors include technical parameters like kV settings and aiming devices, differences between various projection techniques and the effects of radiation protection measures.[3]

MATERIAL AND METHODS

Dose measurements were performed by means of LiF-100 thermoluminescence dosimeters in an Alderson Rando phantom. Special provision were made in this

Table I, Effective dose equivalent for several intra-oral dental radiographic techniques. The kV and mAs values were adjusted to produce radiographs of constant density.

technique	kVp	mAs	H_{eff} (mSv)
complete mouth survey, bissecting angle technique, 20 radiographs, Ultraspeed film			
- GE-1000	50	356.5	0.284
- Philips Oralix	65	123.0	0.067
- Philips Oralix	65	127.0	0.067
- Philips Oralix	65	131.0	0.067
- Ritter Expl.	75	231.8	0.122
- GE-1000	90	46.1	0.050
- GE-1000	75	92.1	0.089
diaphragm x .5	75	92.1	0.082
with lead apron	75	92.1	0.084
with lead collar	75	92.1	0.064
complete mouth survey, paralleling technique, 20 radiographs, Ultraspeed film			
- GE-1000	50	477.4	0.173
- GE-1000	90	61.2	0.027
- GE-1000	75	122.4	0.051
with lead apron	75	122.4	0.047
with lead collar	75	122.4	0.046
complete mouth survey, bissecting angle technique, 20 radiographs, Ektaspeed film			
- GE-1000	50	200.1	0.160
- Philips Oralix	65	50.0	0.027
- Philips Oralix	65	57.0	0.027
- Philips Oralix	65	59.0	0.027
- Ritter Expl.	75	117.75	0.062
- GE-1000	90	25.7	0.028
- GE-1000	75	51.3	0.049
diaphragm x .5	75	51.3	0.046
with lead apron	75	51.3	0.047
with lead collar	75	51.3	0.036
complete mouth survey, paralleling technique, 20 radiographs, Ektaspeed film			
- GE-1000	50	283.1	0.103
GE-1000	90	36.3	0.016
- GE-1000	75	72.6	0.030
with lead apron	75	72.6	0.028
with lead collar	75	72.6	0.027
occlusal technique, Ultraspeed film			
- GE-1000	75	3.3	0.002

Table II, Effective dose equivalent for several extra-oral dental radiographic techniques.

technique	kVp	mAs	H_{eff} (mSv)
panoramic technique			
- extra-oral			
Siemens OP2	65	800.0	0.130
Panelete	73	240.0	0.030
- intra-oral	50	0.2	0.013
temporo-mandibular joint			
- Schüller	85	15.0	0.005
- Parma	65	23.0	0.660
- Parma	75	15.0	0.561
lateral skull			
- Siemens	96	175.0	0.015

phantom to enable comprehensive measurements in the head and neck region. The measurements refer to four different dental X-ray devices, five different projection techniques, four different kV-settings, two film types and four measures for dose reduction (table I). In addition to these, two panoramic X-ray techniques and four techniques to X-ray a larger area of the skull were involved in this study (table II).

The TLD measurements were converted into organ doses (mGy).[1] From these the H_{eff} and related fatal risk can be calculated for each of the projection techniques and parameters.

RESULTS

The H_{eff} ranged from .002 to .660 mSv (tabel I and II). This corresponds to a fatal risk in the order of 10^{-6} or 10^{-7} per examination. The kV-settings showed to influence the total dose considerably. A decrease of the kV from 75 to 50 kV almost doubles the H_{eff}. All measures for dose reduction proved to be effective.

CONCLUSION EN DISCUSSION

It can be concluded from the results, that dose reduction can be achieved by several measures:

- Dental X-ray apparatus should have a rectangular open ended tube instead of a closed pointed cone.
- Rotational panoramic radiographs are not a "cheap" substitute for full mouth surveys as far as dose is concerned. The dose in the rotation axis of the X-ray beam in rotational panoramic radiography is rather high and brings the total dose of this technique to the same level as that of a conventional full mouth radiographic survey.

- Radiographs of the temporo-mandibular joint made by the technique according to Parma should be rejected. The Schüller technique is to be prefered in these situation.
- Use of a lead apron or a cervical collar must be made obligatory.
- Tube voltages of less than 65 kV result in a much higher dose than higher kV settings and should be avoided therefore.

The results of this study will be used to estimate the population dose due to oral radiodiagnosis in the Netherlands in a current investigation (Velders and Selling, elsewhere in these proceedings).

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PATIENT EXPOSURES FROM INTRA-ORAL DENTAL RADIOGRAPHIC EXAMINATIONS

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ABSTRACT

This report describes a study to determine patient exposure to selected areas at the thyroid gland, the central chest area, the testes and ovaries, from conventional radiographic produres for dental patients. The study was designed also to compare the efficiency of the cervical lead shield with the lead apron which both serve as protection from radiation. A review of the literature indicates little agreement in doses reported, due to wide differences in machine factors, in systems of dosimetry or in anatomic location. No references has been found which compares the efficiency of the cervical lead shield with the lead apron. As phantom a Temex one was used. This tissue equivalent human phantom, developed by Stacey and Dickens , is made in the form of a woman and contains a rubber compound fluid which is a soft tissue equivalent. It is cast on a natural human skeleton and is made in such a way that air pores are located at the appropriate places similar to those of a living being. Full mouth examination was carried out, 14 periapical and 2 bitewing, using a X-ray generator type Siemens Helyodent, 50 KV, 7 mAs, and inherent filteration of 1.5 mm aliminium. The exposure time ran from 0.8 to 1.2 seconds. TLD-100 were placed upon the thyroid gland. In some situations extra pairs of dosimeters were placed upon the neck in the area of the thyroid gland in a scattered fashion along a length of approximately 2 cm. The measurements were conducted under three conditions: 1) with a lead apron; b) without a lead apron; c) with a cervical lead shield. At each of the 4 anatomic sites pairs of dosimeters were positioned adjacent to each other at the same level, and sealed in a poly-athylene envelope.
Results. The dose received by the thyroid gland area with a lead apron is approximately 300 mR and without the lead apron 350 mR. Both results are in the same range, so the dose due to scattering that reaches the thyroid gland is not larger than using a lead apron. The dose received by the thyroid gland area with the use of a cervical lead shield was decreased to 30-35 mR for a single X-ray. The cervical lead shield reduced the radiation dose as much as 90% in the Thyroid region.
The cervical lead shield proved to be more efficient than the lead apron in the thyroid gland region. We found adecreasing in the skin dose radiation by as much as 90%. Therefore we recommend the use of the cervical lead shield as the most effective protection measure against radiation when performing dental X-ray examinitions.

HANDBOOKS OF TISSUE DOSES IN DIAGNOSTIC RADIOLOGY

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Two additional Handbooks for determining tissue doses from diagnostic radiology procedures are now available. One permits estimates of glandular tissue doses in mammography (1), the other is an update of an earlier Handbook which permits estimates of several tissue doses for common projections in diagnostic radiology (2). Both Handbooks have been derived from calculations using a Monte Carlo computer code and associated anthropomorphic phantoms.

Experience with previous Handbooks of this type has demonstrated the practical value and versatility of such a format for medical and radiation protection personnel.

HANDBOOK OF GLANDULAR TISSUE DOSES IN MAMMOGRAPHY

The Handbook of Glandular Tissue Doses in Mammography contains data applicable to a wide range of techniques in present day mammography. In mammography the glandular tissue is the tissue considered vulnerable to radiation-induced breast cancer. The Handbook presents the absorbed dose to glandular tissue in the breast per unit entrance exposure (free-in-air) as a function of breast size, breast composition, breast thickness, breast compression and x-ray beam quality. These data permit absorbed dose to glandular tissue to be computed readily for the array of current mammography techniques.

From the absorbed dose computed in the whole breast tissue (excluding the skin layer), the glandular tissue dose (D_G) is determined by multiplying the dose to the whole breast (D_B) by the ratio of the mass energy absorption coefficient of glandular tissue $(\mu_{en}/\rho)_G$ to that of the whole breast $(\mu_{en}/\rho)_B$. The mass energy absorption coefficient for the whole breast is derived from the weight percents of glandular (G) and adipose (A) tissues and the mass energy absorption coefficients of the two tissues $(\mu_{en}/\rho)_G$ and $(\mu_{en}/\rho)_A$. Therefore:

$$D_G = D_B \frac{(\mu_{en}/\rho)_G}{(\mu_{en}/\rho)_B},$$

where $$(\mu_{en}/\rho)_B = G(\mu_{en}/\rho)_G + A(\mu_{en}/\rho)_A.$$

A sample of the Handbook data is given in Table 1 for the craniocaudal view and a uniform firm compression of 6 cm.

Table 1. Glandular Tissue Doses in Mammography

Craniocaudal view, 6-cm thickness, glandular tissue content between 5 and 100 percent (by weight)

HVL	Glandular tissue dose (mGy) for 1 mC/kg entrance exposure (free-in-air)[a]				
	Glandular tissue content				
(mm Al)	5%	25%	50%	75%	100%
0.2	2.6	2.4	2.1	1.8	1.6
0.4	7.1	6.4	5.8	5.2	4.9
0.6	10.8	9.9	9.0	8.4	7.8
0.8	14.0	13.0	12.0	11.3	10.7
1.2	18.5	17.5	16.5	15.7	15.1
1.6	21.5	20.5	19.4	18.7	18.0
2.0	23.9	22.8	21.8	21.0	20.4
2.4	25.7	24.6	23.6	22.8	22.3

[a] Multiply table entries (mGy per mC/kg) by 25.8 to obtain previous units in mrad per R.

To compute glandular tissue doses from specific mammography applications, the user applies measured or estimated values of entrance exposure (free-in-air) and x-ray beam quality (half-value-layer) relevant to the actual clinical conditions of interest.

UPDATE OF HANDBOOK OF TISSUE DOSES FOR COMMON PROJECTIONS IN DIAGNOSTIC RADIOLOGY

The update of the Handbook of Tissue Doses for Common Projections in Diagnostic Radiology contains an expanded collection of data applicable to a reference adult patient. The range of x-ray beam qualities (half-value-layers) covers from 1.0 to 6.5 mm Al, as appropriate. Additional tissues, notably the female breast, have been included, and distinction is made between tissue doses to males and females when appropriate.

The Handbook table for each projection includes an indicator of cancer detriment from the aggregate of these tissue doses, based on current risk coefficients for various cancers induced by radiation and the severity of those cancers.

A Cancer Detriment Index (I_c) has been formulated as follows:

$$I_c = \sum_{i=1}^{n} \left[r_i(f) + s_i \, r_i(c) \right] D_i$$

where $r_i(f)$ is the lifetime risk coefficient for fatal cancer i (per mGy),
$r_i(c)$ is the lifetime risk coefficient for "curable" cancer i (per mGy),
s_i is the relative severity associated with successful treatment of cancer i, and
D_i is the average absorbed dose in the appropriate tissue for cancer i (mGy).

The risk coefficients and relative severities for treatment used in the Cancer Detriment Index are given in Table 2.

Table 2. Lifetime Risk Coefficients[a] for Induction of Fatal and "Curable" Cancers and the Relative Severities for Treatment of "Curable" Cancers

	$r_i(f)$[a]		$r_i(c)$[a]		
Cancer (i)	Male	Female	Male	Female	s_i
Lung	2.0	2.0	0.1	0.1	0.95
Leukemia	2.4	1.6	0.12	0.08	0.95
Thyroid	0.33	0.67	6.3	12.7	0.05
Breast	---	5.0	---	3.0	0.60
Other	5.0	5.0	1.5	1.5	0.75

[a] Multiply table entries for $r_i(f)$ and $r_i(c)$ by 10^{-6} to obtain risk coefficients per mGy.

The entries in Table 2 and the formulation of the Cancer Detriment Index take advantage of the rationale and discussion by Pochin in ICRP 45 (3), with modifications to distinguish between risks to males and females. The breast cancer risk coefficient is asssigned entirely to females, therefore, the value found in ICRP 45 that is for a mixed occupational population is doubled. The risk coefficients for leukemia and thyroid cancer have been adjusted to reflect the observed differences in risk to males and females (4).

The Cancer Detriment Index therefore reflects both the detriment from fatal cancers, and the detriment from cancers that can be treated sucessfully, the latter being given a weight equal to the ratio of fatal to fatal plus "curable" cancers (3). The result is an indication of the overall detriment from all potential cancer risks.

A typical Handbook tabulation, presenting the tissue doses and the resulting Cancer Detriment Index for a sample x-ray projection, is given in Table 3. All values are presented per unit entrance exposure (free-in-air).

Table 3. Tissue Doses (mGy) for 1 mC/kg Entrance Exposure (free-in-air)[a] and Cancer Detriment Index (I_c) (per mC/kg)[b] - AP Thoracic Spine, SID = 102 cm, 17.8 x 43.2 cm field size, 80 kVp

HVL(mm Al) ⟶	2.5		3.5		4.5	
Tissue	Male	Female	Male	Female	Male	Female
Lungs	5.5	4.1	7.0	5.2	8.4	6.3
Active Bone Marrow	0.7	0.7	1.0	0.9	1.2	1.0
Thyroid	3.7	3.7	4.6	4.6	5.4	5.4
Breasts	---	14.0	---	16.0	---	17.5
Other Tissue	1.9	1.5	2.3	1.9	2.7	2.2
I_c (multiply values by 10^{-4})	0.28	1.19	0.34	1.39	0.41	1.54

[a] Multiply table entries (mGy per mC/kg) by 25.8 to obtain previous units in mrad per R.

[b] Multiply table entries for I_c (per mC/kg) by 0.258 to obtain I_c for previous units (per R).

To compute tissue doses and the Cancer Detriment Index for a reference adult patient from specific diagnostic projections, the user applies measured or estimated values of entrance exposure (free-in-air) and x-ray beam quality (half-value-layer) relevant to the actual clinical conditions of interest. For example, an examination consisting of 3 AP thoracic spine films at HVL = 2.5 mm Al and 0.1 mC/kg (0.39 R) each would yield Cancer Detriment Indexes of 8.4×10^{-6} (male) and 3.6×10^{-5} (female).

Consideration is being given to formulating corresponding detriment indexes for hereditary effects and effects to children irradiated in utero.

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PERFORMANCE EVALUATION OF DSA UNITS

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ABSTRACT

The performance of the digital subtraction angiography(DSA) units in Finland has been measured by means of the DSA phantom and measuring method standardized by the American Association of Physicists in Medicine (AAPM).The performance evaluation method is non-invasive and involves the use of a patient simulating phantom with different test inserts.The 8 units, including both continuous and pulsed mode units,were evaluated applying a technique normally used in each x-ray department.Image quality including low and high contrast performance, the response of log-amplifier,image homogenity, artifacts and misregistration were measured.Besides the AAPM test inserts, a Burger-type hole pattern insert for contrast detail determination was also studied.

The preliminary results show reasonable differences in contrast and spatial resolution.The high contrast resolution varied between 0.8-1.4 lp/mm, iodine contrast spatial resolution between 0.35-0.7 lp/mm with 10 mg/cm2 iodine consentration and 0.175-0.5 lp/mm with 5 mg/cm2 consentration, when ⌀ 25 cm field size and 512x512 matrix size were used. Also ⌀ 2-4 mm iodine vessel were detected with 10 mg/cm2 and ⌀ 2-4 mm with 2.5 mg/cm2 consentration. The results agree well with the results presented in literature. The dose rate in front of the grid varied between 1.5-10 mR/s on continuous mode units and dose/frame between 0.1-1.0 mR on pulsed mode units. The hole pattern test insert proved to be a sensitive method for showing differences in contrast details. Some deficiencies in logaritmic amplification and image homegenity were also observed.

AN EVALUATION OF THE VICTOREEN "NERO" (NON-INVASIVE EVALUATOR OF RADIATION OUTPUTS)

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INTRODUCTION

There is increasing concern about the radiation dose received by patients and staff during diagnostic and interventional radiology. Significant reductions in dose can be achieved by, amongst other measures, implementing a rigorous Quality Assurance (QA) program to optimize the performance of the x-ray equipment. Most QA programs require measurement of tube potential (kVp), exposure and exposure rate. The development of non-invasive methods to measure the tube potential has facilitated the implementation of such programs since, unlike a voltage divider, connection to the high voltage circuit is not required.

The Victoreen NERO is one such instrument which measures quantities, including the average of the tube potential peaks (kVp Avg), exposure and exposure rate. We have determined, using a voltage divider, the accuracy of the kVp Avg measured by three NEROs in routine use, and studied the effect of added filteration on kVp Avg. In addition, the exposure and exposure rate accuracy of these NEROs was also determined using a digital exposure/exposure rate meter.

METHODOLOGY

Calibration of kVp Avg: A Siemens Tridoros 5S, three phase, six pulse generator was used to evaluate the kVp Avg measured by NERO, with its detector placed 66 cm from the focal spot. The kVp Avg from a voltage divider (Siemens "Muffs") connected in parallel to the high voltage circuit, was determined simultaneously using a storage oscilloscope. The kV peaks used by NERO to compute kVp Avg were identified on the oscilloscope. The corrected (as detailed below) average of these peaks is termed the "Muffs" kVp Avg.

Applying a known voltage in the range 0-3 kV across the "Muffs", the dividing ratio was measupred to be 100,340:1±0.04% as against 100,000:1±2% quoted by the manufacturer. In measuring the output voltage on an oscilloscope which has an input impedance of 1 MΩ, coupled with a 10.525 kΩ output impedance of the "Muffs", the effective dividing ratio is increased by 1%, becoming 101,400:1±0.05%; and was the value used.

The effect of added filteration on kVp Avg was determined by inserting a 3.3 mm thick type 1100 aluminium filter (Victoreen, USA) in the x-ray beam (it was taped to the light beam diaphragm).

Evaluation of exposure and exposure rate: A calibrated 30 ml ionization chamber, recommended for radiological measurements (PM-30, Capintec, USA) and a Capintec Model 192 exposure/exposure rate meter, were used to evaluate the NERO measurements.

When measuring exposure, both the NERO detector and PM-30 were placed in the beam to overcome machine output variations between exposures. The detectors were placed at marked points at right angles to the tube axis to negate the heel effect. Furthermore, spatial nonuniformity was overcome by making identical paired exposures with the detector positions (A & B, say) interchanged. The relative sensitivity was calculated from the exposure measurements as follows. Let N and C be the NERO and Capintec response respectively. The radiation exposures at positions A and B at which the detectors are placed are a and b respectively. The outputs of the NERO N_A (at position A) and Capintec C_B (at position B) for the first exposure are :

$$N_A = Na_1 \text{ and } C_B = Cb_1$$

On interchanging the positions of the two detectors, the outputs for the second exposure are :

$$N_B = Nb_2 \text{ and } C_A = Ca_2$$

Combining all four measurements, we obtain the relation :

$$\frac{N_A N_B}{C_A C_B} = \left(\frac{N}{C}\right)^2 \frac{a_1 b_2}{a_2 b_1}$$

The x-ray output may vary considerably from the first exposure to the second. However, the ratio of exposure at positions A and B will remain constant so that $a_1/b_1 = a_2/b_2$. Given this, we may write:

$$\frac{N}{C} = \left(\frac{N_A}{C_A} \quad \frac{N_B}{C_B}\right)^{1/2}$$

The percentage NERO error was determined from this ratio. It was determined that for the geometry used, cross-scattering between the detectors was not measureable and could be ignored.

For determining exposure rates in the fluoroscopic mode, only one detector was placed in the beam at a time, and the measurement made well after the beam had stabilized.

RESULTS AND DISCUSSION

Evaluation of kVp Avg: Four exposures were made with each of the three NEROs, for every set of exposure factors. The mean of four kVp Avg values measured by a NERO was expressed as a percentage of the corresponding mean value (n=4) from the "Muffs" and the percentage error was calculated. This is shown in Table 1, along with the mean values provided by the "Muffs" (n=12). The latter indicates the absolute kVp Avg values. It can be seen that only NERO 1 measures kVp Avg to within 3% of the "Muffs". This is in agreement with the Victoreen claim that the accuracy of kVp Avg measured by NERO is within ±3% or ±3 kVp, whichever is greater. The other two NEROs were not within the specifications - NERO 3 being up to 16% out from the true value. This raises concern regarding the maintenance of accuracy of NERO and highlights the need for continual checks and calibration.

TABLE 1 - Percentage Error in kVp Avg measured by NERO

Exposure factors	"Muffs" kVp Avg (n=12)	Percentage error in kVp Avg		
		NERO 1	NERO 2	NERO 3
50 kVp, 10 mAs	46	0	-1	-7
75 kVp, "	73	+1	-1	-12
100 kVp, "	98	-3	-5	-16
75 kVp, 20 mAs	64	-1	-2	-11
100 kVp, "	91	-2	-3	-14

When a 3.3 mm Al filter was inserted in the beam, the kVp Avg of each of the three NEROs increased by 2% or, alternatively, 0.6%/mm of Al at 75 kVp (Half-Value Layer = 2.4 mm Al). This is twice the reported value of 0.3%/mm Al (Simon

1985). This discrepancy could, perhaps, be accounted for by possible differences in beam quality due to kVp and Half-Value Layer which are not specified by Simon (1985). This effect of filteration may, however, be ignored in testing diagnostic x-ray machines, since the variation in filteration in these is not large, typically between 1.5 - 2.5 mm of Al (Putney and Raymond 1984).

Evaluation of exposure/exposure rate: The accuracy of the exposure measurements was good for all NEROs, being better than 5%, when the tube potential was varied from 50 to 100 kVp and when mA was varied at 75 kVp.

Exposure rate was measured with the generator factors maintained at 106 kVp and 3.2 mA. The accuracy of the exposure rate was +2.9%, +5.8% and -33.7% for NEROs 1, 2 and 3 respectively. Thus, NERO 3 was not within the 15 percent accuracy specified by Victoreen. Such an underestimate in exposure rate is significant, given that most QA programs require that the maximum tabletop exposure rate during fluoroscopy should not exceed a specified limit.

In conclusion, this study emphasizes the need for regular checks and calibration of NERO and similar instruments. The effect of added filteration on kVp Avg may be ignored, since the variation in filteration in diagnostic x-ray machines is not large.

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RADIATION PROTECTION IN THE MEDICAL FIELD

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ABSTRACT

Medical sources of radiation produce the most significant contribution to the collective dose of any population; therefore efforths should be done to reduce such contribution as much as reasonably possible keeping at the same time every real benefit to the health of people.
The non thereshold linear hypotesis between risks and doses is the present support of radiation protection philosophy as recommended by the International Commission of Radiological Protection (ICRP). The paper analyze its implication on individual and collective basis; is shown that even when a single radiological procedure con generate a "negligible" risk for a given person, the collective dose may be relevant if many people are irradiated in similar procedures and the associated detriment, expressed as the mathematical expectation of harm, may be significant.
Such considerations provides basis for a better judgment of justifications and optimization of practices, particulary when generalized procedures of radiological survelliance are imposed to groups of populations. It is shown that reduction of unnecesary doses in radiation medicine results in appreciable reduction of collective dose and its associated detriment.
Such reductions can be achieved by promotion and enforcement of utilization of appropiate equipment and technichs as well as a good policy for training of physicions and physicist. Argentine experience on adoption of standards of quality and training programs is commented.

RADIATION PROTECTION SURVEYS - AS AN OPTIMISING PROCEDURE IN DIAGNOSTIC RADIOLOGY

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Radiation Protection Division (RPD) in Kuwait operates primarily for the benefit of health authorities in regulating the use of ionizing radiation in medical field. There are approximately 250 radiology and 40 dental X-ray units in the country. Radiation protection surveys are a regular feature of RPD and include planning, pre-commissioning, routine and after repair checks on the newly installed as well as in-use X-ray units. The aim of these surveys is to ensure standard X-ray outputs and adequate safety measures so as to enhance the image quality and avoid the unnecessary exposure to both the patients and operators.

Optimization in diagnostic radiology implies maximising the benefit/cost ratio. In diagnostic radiology, the benefit is the correct medical diagnosis and the cost includes the indirect costs arising from radiation induced determinent. Since there is no definite way of evaluating image quality in clinically meaningful terms, the physicists, engineers, techicians and doctor interpret it within their own professional ways. Ultimate goal, however, is to enhance the information content of the images with the least amount of radiation administered to the patient during the examination. In order to achieve this goal the RPD adopted two courses of action, i.e, regular radiation protection surveys of the equipment and training courses for the technicians, engineers, physists and radiologists. The surveys include reject film analysis and comprehensive performance check on tube and generator, image intensifier, and automatic exposure control systems. Training courses consisting of 3 to 5 days programme of lectures and demonstrations are held every two years.

REJECT FILM ANALYSIS

Each radiology department is encouraged to conduct a reject film analysis to determine the incidence of and reasons for retake examinations. Initially two 700-bed hospital (Hospital 1 & 2) were examined by RPD over a 6 month period. Each hospital carried out 125,000 radiographic examinations annually. The study showed that the average retake rate were 9% and 6% and the reject rate were 13% and 9% from Hospital 1 and 2 respectively (1). In Hospital 1 positioning and underexposures were the causes of over 50% of the rejected films while equipment faults and both over and underexposed film were the major causes in Hospital 2 (Table I). The information generated by this analysis of retakes has helped both hospitals to take corrective action and in some instances to replace equipment, such as faulty cassettes and timers.

Table I: The Percentage Frequency of Principal Causes for Rejects

	Hospital 1	Hospital 2
Positioning Error	32	22
Too Light	23	21
Too Dark	17	16
Fog	17	11
Others	11	18

X-RAY TUBES AND GENERATORS

Comprehensive performance check including checks on the tube kilovoltage, timer accuracy and radiation output consistency and linearity with changes in tube current, is carried out. Defects, faults or non-compliances with standard, based on ICRP and NCRP recommendation (2,3) are conveyed in writing to the user and to the supervising authority responsible for maintaining the X-ray equipment in state hospitals. In private clinics, the user is asked to contact the appropriate manufacturer or agent for rectifying the reported defects. Follow up inspections carried out to ensure that the reported faults have been satisfactorily rectified play an effective role in the optimization process.

Findings of the survey that were reported to the user in the last 6-month period are presented in Table II. 65 general purpose radiography & dental X-ray units were surveyed during this six month period. Out of this 29 units were found to have one or more than one defect. Four units were taken out of service and one facility was redesigned for better protection measures.

Table II: Findings of Radiation Protection Survey of Radiography & Dental X-ray Units in a 6 Month Period

Performance parameter	Number of Faults or Non-Compliances Observed
Light Beam Alignment	8
Exposure Timer	9
Tube Output Consistency & Linearity	12
kVp	12
FFD	1
Total Faults	42

Over the three years during which these surveys has been performed, there has been a consistent reproducibility in the number of defects/faults detected per 6-month period. Around 40% unit are consistently found unsatisfactory for one or the other reason in the specified period.

IMAGE INTENSIFIER SYSTEMS

In fluoroscopic units, the measurements/checks mainly include field size determination, table top exposure, cumulative timer activation and contrast resolution. The field size determination ensure confinement of the X-ray field size to the size of the input surface of the image intensifier. Table top exposure exceeding 50 mGy/min are reported to the user as a measure to optimize fluoroscopic outputs. While 50% units had larger field size, the table top exposure exceeded the acceptable level of 50 mGy/min in more than 35% units surveyed in a 6-month period (Table III). One C-arm unit was found without any cumulative timer, whereas in 15% of the units the timer did not get activated during fluoroscopic exposure. Both high and low contrast resolution capabilities of the imaging system are checked with the RMI resolution test tool (Model 141A and 151). While resolution capabilities were found below aceptable levels in 20% of the surveyed units , the majority of the remaining 80% showed the bare minimum acceptable levels (holes/sq.in) for the corresponding image intensifier size.

Table III: Findings of Radiation Protection Survey of Image - Intensifier Systems in a 6-Month Period

Performance Parameter	% of Faults or Non Compliances Observed
Field Size	50
Table Top Exposure	35
Cumulative Timer	15
Contrast Resolution	20

AUTOMATIC EXPOSURE CONTROL (AEC) SYSTEMS

A survey of the performance of the AEC system throughout the region revealed that 90% of these systems were either not functioning correctly or had not been installed correctly. Among the 10% units in which AEC was found operating were either newly installed conventional units or the chest X-ray units. Chest X-ray examinations constitute 90% of the total X-ray examinations, conducted in the country and majority of these examinations are performed on A.E Controlled units. In addition to the normal tube and generator performance check, a comprehensive measurements of entrance skin exposure (ESE) is also undertaken (4). The wide variations in ESE which were observed (Table IV) and the subsequent investigation to account for these results, has led to the servicing of the film processing units, replacement of the intensifying screens and recalibration of the tube and generator parameters.

Table IV: Range of Entrance Skin Exposure and Associated Beam Parameters for the Surveyed AE Controlled Chest X-ray Units

	Photofluorography Units	AEC (Large Film) Chest X-ray Units
Range of Min. and Max. Entrance Skin Exposure (mGy)	0.80 - 1.4	0.15 - 2.1
kVp range	65 - 81	125
FFD	105 - 150	200 cm

CONCLUSION

Optimization in diagnostic radiology implies maximising the benefit/cost ratio. This was effectively achieved by conducting regular radiation protection surveys which included comprehensive performance check on X-ray tube and generator, automatic exposure control systems and image intensifier systems. These surveys have resulted in

- increased staff awareness of faults and non-compliance in the X-ray units
- increased confidence in the performance of the equipment
- technical improvement of the equipment, and
- reduced dose to the staff and patients.

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OPTIMUM X-RAY ENERGY IN DIAGNOSTIC RADIOLOGY: MONTE CARLO SIMULATION STUDIES

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INTRODUCTION

Diagnostic X-ray examinations are the principal source of population exposure to ionizing radiation. ICRP recommends the limits in dose equivalent that are based on the risk estimates for each organ and tissue. For this reason a realistic knowledge of dose values to the organs involved in X-ray examinations and more efficient methods for dose reduction and image optimization are necessaqry. In this work a new methodology is proposed to correlate organ dose with irradiation photon energy and image quality. To recognize the influence of inhomogeneities on energy deposition in the body due to low energy photon exposure the new sex specific heterogeneous mathematical phantoms, GSF Adam and Eva,(1) have been used. A Monte Carlo method has been applied to the mathematical models to calculate organ doses and to evaluate the quantity of photons impinging on radiographic plane in diagnostic radiology under conditions of partial body irradiation. The technical parameters for a typical radiographic technique have been reproduced. External irradiation with monochromatic photon energies have been considered in the range 20-100 keV. Organ dose values have been normalised to the number of photons transmitted without interactions on radiographic plane. The doses so obtained against photon energy have been considered as ones necessary to produce the same useful diagnostic information. Moreover the ratio between the number od scattered and transmitted photons on radiographic plane have been calculated. Finally a discussion on the optimum photon energy for every organ has been carried out comparing these data.

MATHEMATICAL EXPOSURE MODEL FOR PHOTONS

Mathematical heterogeneous phantoms have been applied to calculate organ doses in diagnostic radiology under conditions of partial body irradiation. They are very similar (2) to GSF Adam and Eva and comprise 24 internal organs each of which is characterized by quadratic equations as shown in figure 1. This mathematical exposure model combines a Monte Carlo technique to the anthropomorphic heterogeneous phantoms. The physical processes treated are limited to the photoelectric and Compton effect since X-ray photon energies in diagnostic radiology range are generally less than 150 keV. Photon history are determined using linear attenuation coefficients, for each organ and tissue, calculated applying White and Fitzegerald (3) polynomial coefficients. The geometrical condition of a radiographic examination can be reproduced in particular the field size in the plane of image receptor, the focus to skin distance and the

direction of incident photons. The radiation beam is considered uniformely distribuited and the source as point source. It is possible to use any radiation quality. The principal restriction associated with this exposure model is the rigidity of the phantoms that cannot consider special position used during X-ray examinations.

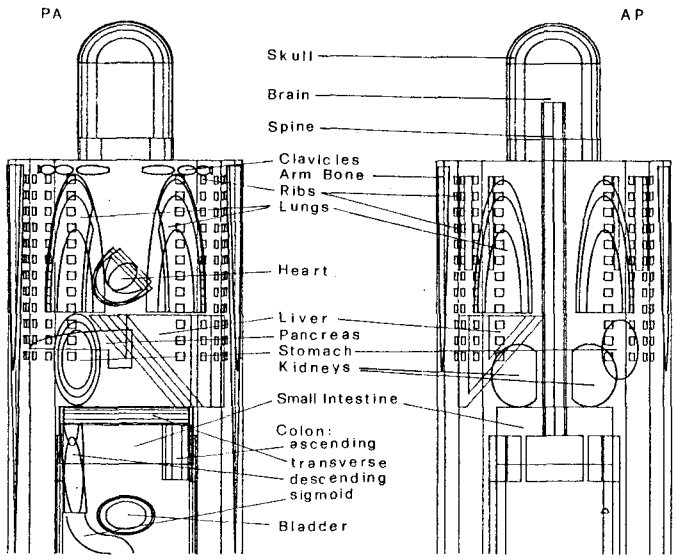

Fig.1 Two dimensional plot for defined vertical planes through the male phantom.

MONTE CARLO METHOD

The Monte Carlo method was applied to follow the transport of each photon through the phantom determining scattering angle, absorption site and to record the resulting energy depositions at the site of physical interaction. The Monte Carlo program used in this study is based on similar principles of ORNL code. The potential side of an interaction is chosen by the usual procedure of taking the distance traversed as:

$$d = (-\ln R) / \mu_o \qquad (1)$$

in which R is a random number (0-1) and μ_o is the attenuation coefficient of skeletal tissue. Determined the region of interaction one then plays a game of change with probability μ_i/u_o where μ_i is the total attenuation coefficient of region i. An

internal cavity is regarded as an organ with zero cross section. A new energy and direction are chosen on the basis of Klein-Nishina differential cross section formula. In particular Compton polar angle is chosen with Kahn algorithm and azimuthal angle with Neumann algorithm. Absorption due to photoelectric interaction is simulated by reduction of the statistical weight that is expressed by:

$$W_n = W_{n-1} (1 - \mu_\tau / \mu) \qquad (2)$$

where μ_τ and μ are the attenuation coefficients respectively for photoelectric and total process before the nth collision. The total flight history of a photon is termined if it is escaped from the phantom or if its energy falls below 1 keV or if its weight falls below 10-4 (in our calculations). In the latter two cases the energy was considered as locally absorbed. The energy deposition for the nth interaction E_n is:

$$E_n = W_{n-1} \left[\frac{\mu_\tau(E_{n-1})}{\mu(E_{n-1})} E_{n-1} + \frac{\mu_c(E_{n-1})}{\mu(E_{n-1})} (E_{n-1} - E_n) \right]$$

RESULTS AND DISCUSSION

Using the mathematical phantoms and the exposure model described above the geometrical conditions of external irradiation as in a diagnostic X-ray examination were accurately reproduced in the simulation. Furthermore the photons that passing through the phantom with or without interactions and impinge on radiographic plane were also recorded. To analyze the influence on organ doses of photon energy incident on the phantom, a simulation with monochromatic energies was carried out for a lungs PA examination. For each monochromatic energy the organ doses were normalised to the number of photons transmitted on radiographic plane without interactions with the phantom. This procedure allows to analyze the organ doses under equal conditions of radiological informations of the image. In this analysis image differences due to the contrast and energy response of image detector system were neglected for simplicity. Figures 2 and 3 show the results of this analysis. The organ dose values are compared with the curve A that represents the ratio between the number of scattered photons and transmitted without interaction on radiographic plane in function of photon energy. The doses to the lungs, the spine and the spleen have a maximum between 25-40 keV, that probably represents the optimum photon energy, then the doses decrease with photon energy. For the lungs and the spleen, down to 60 keV, the dose is independent on photon energy. For the other organs analyzed the doses increase with the photon energy, in particular for the heart where the dose increases as the scattered radiation on radiographic plane. These preliminary results are interesting because they shown that the variation of organ doses with photon energy can be foreseen with difficulty. In particular the decreasing of the dose with the increasing of photon energy is not absolutely true.

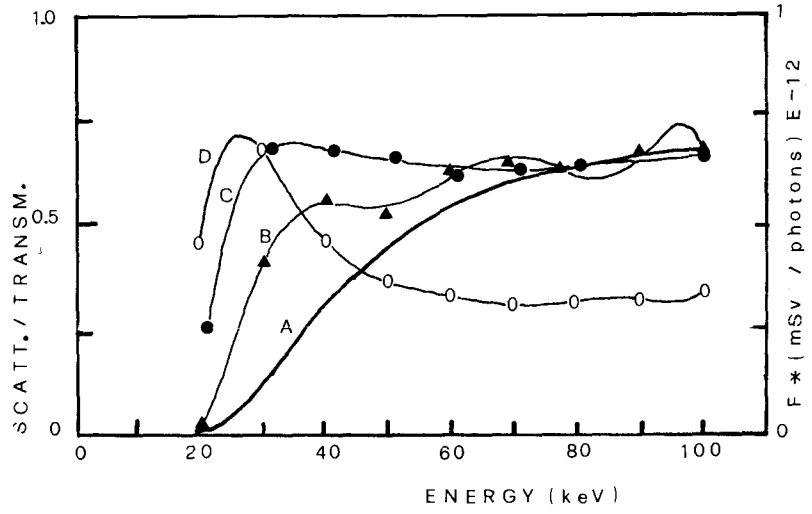

Fig.2 Lungs PA projection. Organ doses vs incident photon energy. Lungs (curve D);liver (curve C); pancreas (curve B); ratio between scattered and transmitted photons (curve A). F is a normalization factor.

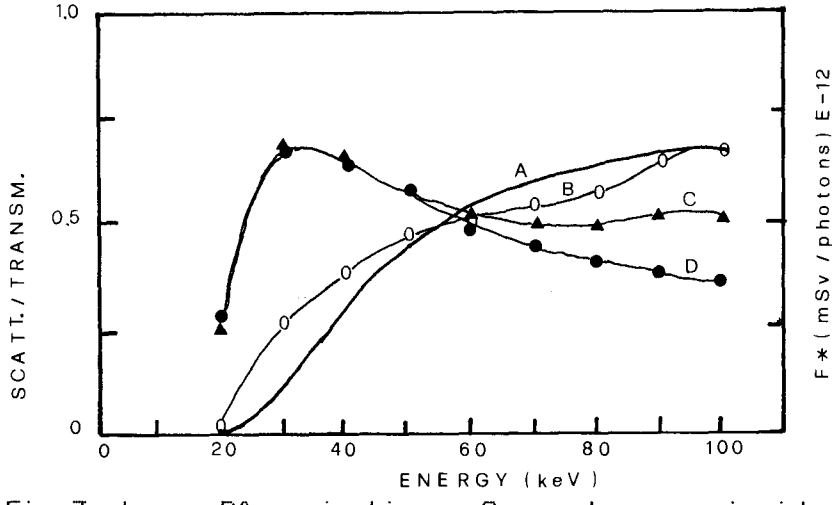

Fig.3 Lungs PA projection. Organ doses vs incident photon energy. Spine (curve D);spleen (curve C); heart (curve B); ratio between scattered and transmitted photons (curve A). F is a normalization factor.

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EXIT SPECTRA FROM THE PATIENT IN DIAGNOSTIC RADIOLOGY: MONTE CARLO SIMULATION STUDIES

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INTRODUCTION

Differential absorption of the beam of X-ray, as it passes through the patient, results in the creation of the radiographic image. To optimise the radiographic image ,the photon energy response of the screen/film system has to match the spectral properties of the radiation beam emerging from the patient. The exit spectrum consists of two components: 1) transmitted without interactions with the body, which contains the usefull diagnostic information, and 2) Compton scattered. The analysis described in this work is based on mathematical phantoms developed at GSF (Gesellschaft fur Strahlen und Umweltforschung) (1). Monte Carlo technique applied to the GSF anthropomorphic heterogeneous has been realized in our laboratory (2). They are similar to the GSF phantoms but contain some variations in elemental composition of organs and tissues. The linear attenuation coefficients for each organ and tissue calculated from White and Fitzgerald (3) have been introduced in our mathematical models. In this work a detailed analysis of exit spectra is presented by means of male and female mathematical human phantoms (Adam and Eva). The technical parameters for two typical radiographic techniques have been reproduced. Transmitted and scattered radiation spectra impinging on radiographic plane have been also simulated. Calculations have been carried out for external irradiation of the phantoms with monochromatic and distributed photon energy.

THE MATHEMATICAL PHANTOM

The heterogeneous mathematical phantoms presented in this work are very similar to GSF Adam and Eva. They comprise 24 internal organs each of which is characterized by quadratic equations. Volumes and masses of organs of the two sex-specific adult mathematical phantoms are based on average values given by ICRP 23 (4). The phantoms have different dimensions with 19 regions of different composition and density. For each organ and tissue percentage by weight of 51 elements and specific gravity from ICRP 23 have been considered. Organs tissues and contents that are not well specificated in ICRP 23 have been assumed as soft tissue. They are : breast, uterus, ovaries, gall bladder, urinary bladder and the contents of intestine and stomach.In the phantoms all skeletal components such as compact bone and active bone marrow are homogeneously distribuited in the skeleton.

RESULTS AND DISCUSSION

The results consist of an accurate analysis of the intensity and the quality of radiation that is responsible of the image forming process during a diagnostic X-ray examinations. To this aim the irradiation geometrical conditions of the radiological examination were accurately simulated. The theoretical spectra incident on the phantoms were taken from HPA Report (5) with costant potential and total filtration of 2.5 mm Al. For each examination two different components of radiation transmitted on image detector system were analyzed: 1) the radiation that passes through the phantom without interaction ("transmitted") and 2) the scattered radiation ("Compton"). The first produces the diagnostic informations of the image , the second causes its impairing. Figure 1 shows the Compton spectra produced from monochromatic photons of 50 keV and 80 keV in abdomen AP projection for Adam and Eva. No appreciable difference can be noticed on the radiation qualities due to the different dimensions of the two phantoms. In figure 2 transmitted spectra are shown for 90 kV incident radiation in lungs AP projection. A wide energy range of the incident spectrum (15-35 keV) is not usefull to produce the image and contributes mainly to the dose. In the conditions described above and to obtain a more general analysis of correlations between incident photon energy and transmitted spectrum on radiographic plane were carried out only calculations with monochromatic photon energy. Furthermore was developed a procedures , starting from the informations obtained from monochromatic energies to obtain the transmitted and Compton spectra produced by any incident distributed photon energy . To calculate the transmitted spectra a simple polynomial interpolation was applied to Monte Carlo results for monochromatic energies. The calculation of Compton spectra was more complicated because every monochromatic energy produces a distributed spectrum on image detector system as shown in figure 1. A method called "decrement method" was applied. It relates each energy Ed of Compton spectrum to monochromatic energy Em by the equation: Ed = Em - ΔE where ΔE is the decrement that increases with 1 keV step. This method takes into account the particular energy distribution of Compton spectra in which the numbers of events increases with Ed. Interpolating by polinomial expressions the numbers of the events of energy Ed obtained from a discrete number of monochromatic energies for each decrement ΔE it is possible to obtain the following mathematical expression:

$$N_c\left(E_j - \Delta E_k\right) = \frac{N(E_J)}{N_0} \sum_{i=0}^{g} A_i\left(\Delta E_k\right) E_J^{(i)}$$

where E_j and $N(E_j)$ are the energy and the number of monochromatic photons j emitted from the X-ray source respectively. N_c is the number of Compton photons at energy $E_j - \Delta E_k$ and $A_i(\Delta E_k)$ are the polynomial coefficients. N_0 is a normalization costant. The result of the procedure applied to 90 kV spectrum in a lungs AP examination is shown in figure 3. The procedure allows to obtain a good accuracy fitting the values of Compton spectrum, obtained from the simulation of a distributed energy emission of the source. Finally figure 4 shows the complete result of intensity and radiation quality on image detector system for the transmitted and Compton photons obtained

applying the procedure described above.

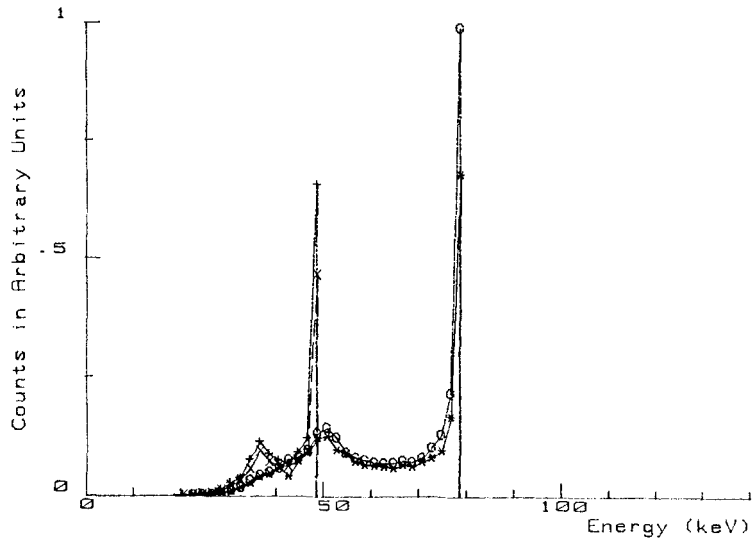

Fig.1 Simulation of Compton spectra produced from monochromatic photons of 50 keV and 80 keV in abdomen AP projection.

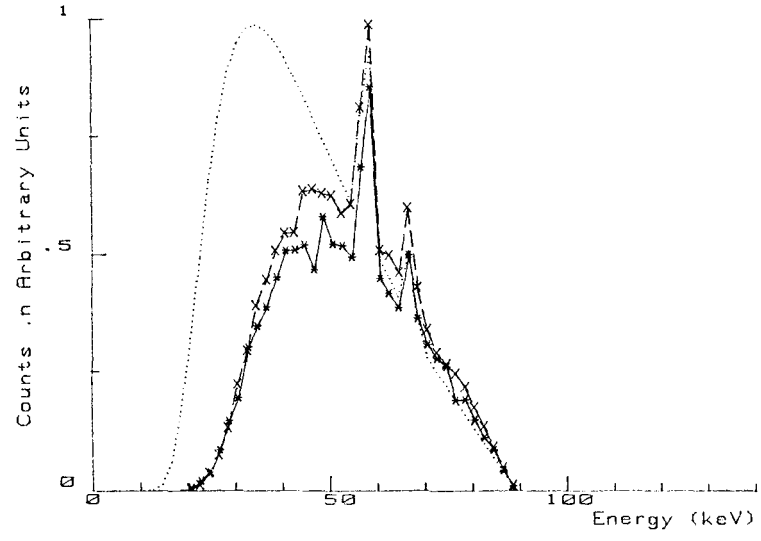

Fig.2 Transmitted spectra for 90 kV incident radiation in lungs AP projection for Adam (asterisc) and Eva (cross) compared with the incident photon spectrum (dot).

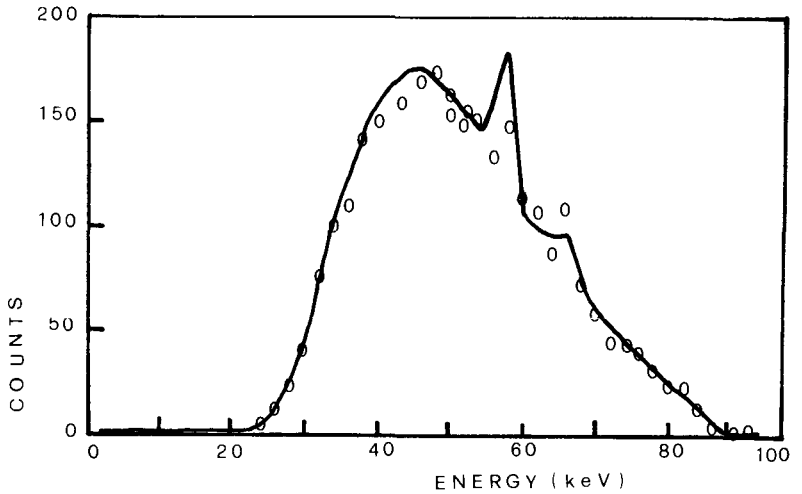

Fig.3 Projection: Lungs AP 90 kV.
Comparison between Compton spectra obtained with Monte Carlo simulation (circle) and applying the decrement method (full line).

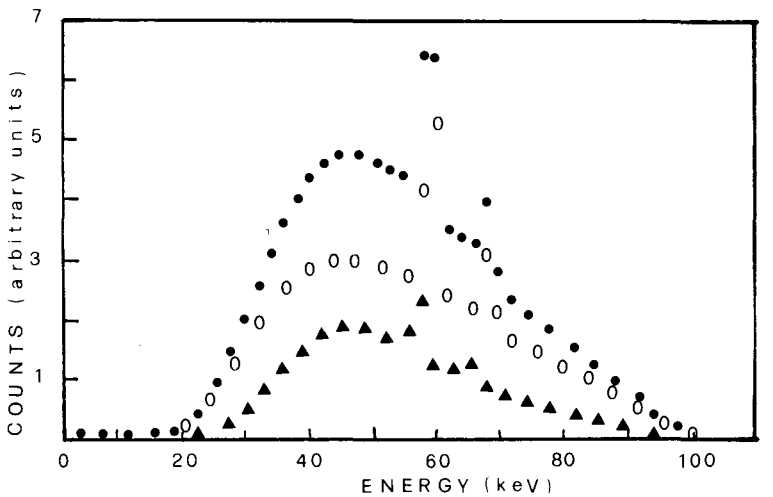

Fig.4 Projection: Lungs PA 100 kV.
Transmitted (circle), Compton (triangle) and total spectra (dot) obtained applying the decrement method.

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A NEW POSTAL QUALITY ASSURANCE PROGRAM FOR DIAGNOSTIC RADIOLOGY DEPARTMENTS.

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The Federal Government of Canada has responsibility for the design and performance of new x-ray equipment, but the Provinces have responsibility for controlling the installation, maintenance and use of the equipment. For the past 30 years Saskatchewan has had regular inspection of X-ray units by radiation health officers who have ensured that deficiencies are corrected. With the present emphasis on quality assurance procedures and hospital accreditation requirements some additional needs have had to be met. These are primarily due to unusual geographical factors. The province covers a quarter of a million square miles and has a continental climate with very severe winters. This used to impede winter travel and many small communities therefore established their own hospital. Today there are a large number of very small hospitals (four beds and upwards) which are geographically widely spread and which, with current resources, can only be visited rather infrequently. A mail out test package, primarily designed to give early warning of the existence of radiation related problems, has therefore been introduced. If a test gives an unexpected result, a prompt visit can be made to the centre concerned to carry out whatever more sophisticated test procedures may be necessary. Experience has shown however that the postal test package has, in itself, a high diagnostic potential so the source of the problem can often be identified without a visit becoming necessary.

Medical radiography is basically an imaging procedure and any comprehensive test procedure must be capable of assessing image quality and detecting degradation of the imaging capacity. The program has been designed around a radiographically realistic, low contrast, resolution test phantom which has been made available to all hospitals and medical clinics conducting diagnostic radiology. It consists of a test plate of approximately tissue equivalent lucite plastic in which an array of cylindrical cavities has been drilled (Fig. 1). These cavities vary in depth from 12.5 mm to 1 mm in one direction, and in diameter from 6 mm to 0.5 mm in the other. The type of anatomical feature in which a radiologist might be interested are typically soft tissue structures of irregular shape and very little contrast, located deep within the body. Consequently their image is overlaid by that of other structures, and degraded by scatter from soft tissue masses above and below. To make a test which is as meaningful as possible the complete phantom consists of the test plate, which was developed to provide an analogue of the structures of interest, sandwiched between two plastic scattering blocks (Fig. 2). The complete test procedure requires a pack of eight TLD dosemeters that are posted out to the centre concerned, and an aluminum step wedge which is supplied with the test phantom (Fig. 3). The test is

carried out by putting the dosimeter pack on top of the clear end of the test phantom so that it does not obscure the image of the resolution test pattern, and then placing the step wedge over it. This whole assembly must then be radiographed twice; one radiograph is placed on the departments Q.A. file and the other (fig. 4) is posted back, along with the exposed dosimeter pack, for laboratory evaluation. The four pairs of dosemeters in this pack record the radiation dose in the direct beam and under 2mm, 4mm and 6mm aluminum absorbers. The test assembly requires an X-ray exposure similar to that used for an AP abdomen. Site specific protocols are established for each installation participating in the program and must be exactly duplicated for successive tests. They are selected to make the densities of the radiograph under the four steps of the step wedge all fall on the linear part of the H & D curve for the emulsion. This requires a maximum density of about 1.3 on the radiograph. To evaluate a returned test package the radiograph is placed on a standard viewing box and the image resolution is evaluated by eye. This is expressed numerically by counting the number of cavities which can be identified in each row of the test pattern. Next the photographic densities of the radiograph under each step of the test wedge are measured, as is the base plus fog level for the film. Finally the TLD dosemeters are read out. From these readings the half value layer for the X-ray beam and the slope of the characteristic curve for the processed film are both calculated. In this way four major parameters of the X-ray installation may be monitored on an ongoing basis without any need for a physicist to make specific site visits for this purpose. These parameters are:

(a) The imaging capacity or image resolution.

(b) The X-ray beam intensity at the control settings that have been specified for the test.

(c) The X-ray beam quality or half value layer.

(d) The consistency of the photographic processing procedures followed.

When carrying out the test the technologists concerned place metal markers at the corners of the light field to check the accuracy of adjustment of the light beam diaphragm. In addition, since most of the test radiograph should be of uniform density, experience has shown that many other equipment faults can often be diagnosed. For example a periodic density variation usually indicates incorrect Bucky grid operation. Faulty screens and degenerating table tops have also been noted.

Most technologists have welcomed the new test procedure. They are encouraged to radiograph the test phantom themselves periodically and to compare the results with the reference films which they have on file. This has frequently proved useful at times when they have been suspecting abnormal results. Tests are normally scheduled to be carried

out annually and it is hoped to increase this to bi-annually as soon as possible. Technologists who feels that their own test radiograph may indicate a possible problem are encouraged to write and ask for a dosimeter pack so that an additional unscheduled test can be carried out as promptly as possible. As a result technologists who are working on their own under rather isolated conditions are experiencing an increased interest in keeping all X-ray installations in optimum condition.

To minimise the staff time required for administering this new program the handling of the returned test packets has, as far as possible, been automated. Manual TLD readout is still used but it is hoped to replace this by a semi-automatic system in the near future. All the TLD dosemeters have an individual sensitivity factor stored in a microcomputer, this is used to automatically correct the readings obtained from the test packs; the microcomputer also carries out all the numerical calculations associated with measuring the half value layer and the slope of the characteristic curve of the processed radiograph, and then, from a comprehensive selection of word processor text files, prepares the final report sent out to the centre concerned. Additionally it is programmed to carry out the scheduling of the test procedure, to provide the mailing labels for the test packages and to flag any instances where the test results show marked deviations from those of previous tests. This enables a Radiation Health Officer to plan to visit the centre concerned and carry out more comprehensive tests on site. In this way the best possible use is made of available radiation physicists.

For the benefit of technologists who have not yet had a great deal of experience in carrying out this new test the radiation safety unit has produced an instructional video tape which can be sent out with the dosemeter pack and which shows the required procedures in considerable detail. A copy of this tape can always be enclosed with one of the postal test packages if so requsted by the technologist who will be responsible for carrying out the test.

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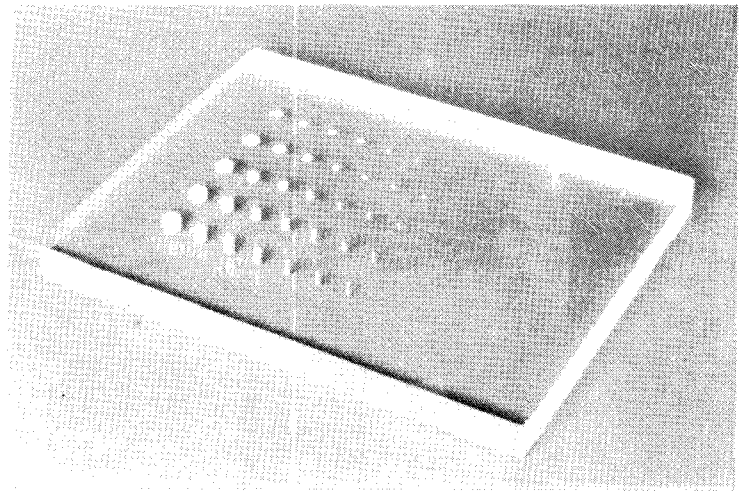

Figure 1
Resolution Test Plate.

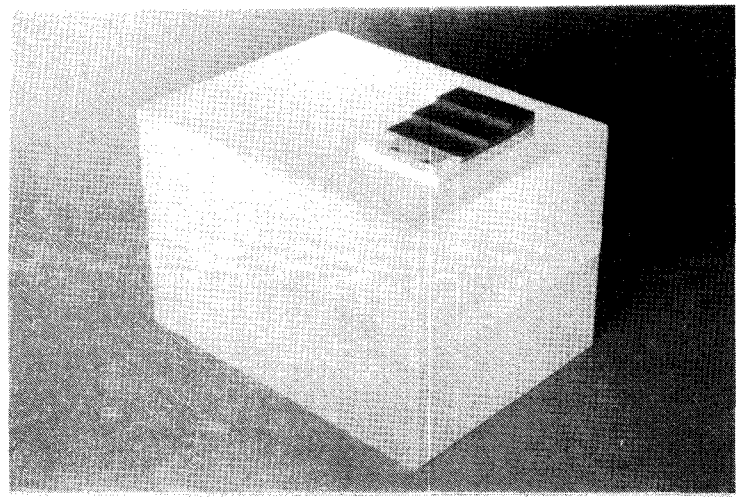

Figure 2
Assembled Phantom and Test Pack.

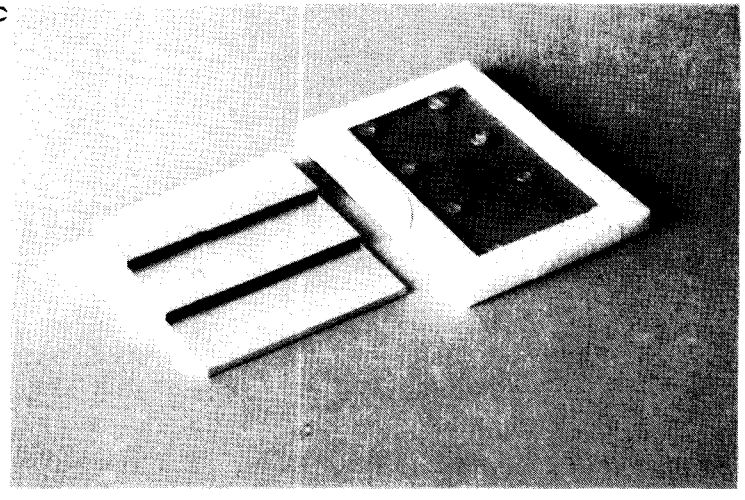

Figure 3
Dosemeter Pack and Step Wedge.

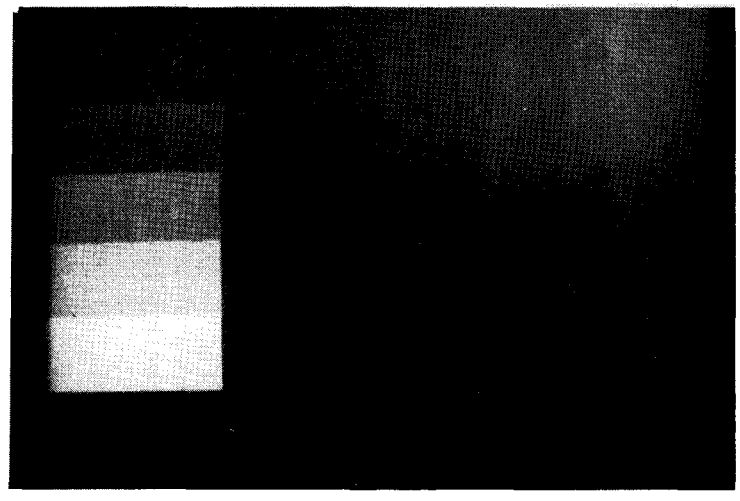

Figure 4
Test Radiograph.

A MICROCOMPUTER BASED X-RAY QUALITY ASSURANCE PROGRAM.

L.Denis Brown.

Radiation Safety Unit. Saskatchewan Occupational Health Branch.

Provincial Radiation Health Officers in Saskatchewan carry out routine inspections of all medical X-ray departments and are responsible for providing medical physics services to all small hospitals where there is no staff physicist; this includes supervising the establishment of an effective quality assurance program. Saskatchewan is a part of Canada where travelling in winter used to be difficult, and where, in consequence a large network of very small hospitals has been established to meet the needs of local communities. Today the number of small X-ray departments and the travelling time required to visit them are so great that, although large departments are visited at least once a year, in many small centres routine inspections are possible only once every two or three years. To minimise the likelihood of serious faults or maladjustments remaining undetected for long periods between inspections, a microcomputer system has been developed to collate and analyse data from six different provincial programs with the objective of identifying any centre where conditions that should be corrected may have developed. This provides early indication of where potential problems may exist, and permits inspections to be carried out promptly in the centres concerned.

The six programs which generate the data for the computer system are:

(a) A service for advising on the design and selection of equipment for any new facility, or any facility making major changes, and for approving the relevant plans before work proceeds.

(b) An annual registration program which ensures that central records relating to all X-ray equipment in use in the Province are kept fully updated.

(c) A statutory Safety and Preventative Maintenance (SPM) program which requires all X-ray equipment used for medical purposes to be tested for electrical and mechanical safety by qualified personnel at intervals based on the age and the usage of the equipment.

(d) A mail out diagnostic radiology test package originally developed by the Radiation Safety Unit in 1981 to supplement the relatively infrequent field inspections which it is able to carry out. This utilises a specially developed resolution test phantom combined with a step wedge and a pack of TLD dosemeter elements which together enable the operation of many parts of the complete system, including the film processing, to be assessed without the necessity of visiting the centre concerned (1)(2).

(e) An advisory service operated by experienced technologists known as Regional Consultants, operating from nearby Regional Hospitals, which support the isolated local technician who encounters any problems in obtaining consistent or high quality radiographs. Whilst the other programs described above are administered by the Radiation Safety Unit, this program is operated through the Regional Services division of the Department of Health.

(f) Acceptance testing of new equipment, followed by regular field inspections, carried out by the staff of the Radiation Safety Unit.

Over the years extensive independent data files have been generated for each of these programs. All of these files need to be analysed if the Radiation Safety Unit staff are to plan their hospital inspection schedules on the basis of need rather than by calendar only. As the available records have become more extensive it has become clear that this is only possible if they can be readily assessed and cross-correlated by means of a computer. It has also become clear that a physicist making a field inspection in a remote hospital would benefit from direct access to all relevant material in any of these data files whilst on site. Computer generation of data and record handling make this possible and in addition permit final hard copy reports to be prepared directly by the officer concerned, thus reducing the load which must be placed on secretarial support services. During the past three years the Radiation Safety Unit has been planning and developing a microcomputer system designed to meet these needs. This system is now in place, and has already proved of considerable value although the full benefit will only be realized after all existing files have been converted to the form of computer based data - a process which will necessarily take a considerable time to complete.

The data flow chart illustrates the structure and purpose of the system as it is currently being developed. The key part of the system is readily accessible computerised data files stored on a 20MB hard disc, backed up by 13 cm. floppy discs each covering a specific region of the Province. All data is first being entered on the floppy disc files that are ultimately used for archival purposes.

Apart from data storage the computer serves four other basic functions in the operation of the inspection program:

(i) When planning his inspection schedule a Radiation Health Officer can access a list of facilities where the computer has flagged abnormal data from any source which appears to indicate that a visit is desirable. Generally speaking visits to facilities listed in this way would take precedence over visits scheduled purely on the basis of elapsed time since the last inspection. We are currently working to improve our in house software which evaluates input data from all the various Provincial programs, to identify facilities where there are potential problems, and to direct the attention of our physicists to possible faults or maladjustments before they visit a facility.

(ii) Each physicist in the field is provided with a small portable

computer loaded with a software program which prompts him as to the various measurements normally made during the course of a standard inspection. As each measurement is made the reading is entered directly into the computer using the keyboard. Thus the program becomes a data disc which subsequently conveys all the readings back to the office following the inspection.

(iii) When making his inspection at the hospital it is frequently helpful for the physicist concerned to have access to relevant information obtained during previous inspections both in order to interpret his current measurements correctly and to decide whether any additional tests are desirable. To facilitate this the portable field computers have been fitted with modems which can be used to access this data through the office computer. Until all data for the entire Province has been entered on a single hard disc this requires manual assistance in the office to load the appropriate floppy disc into the disc drive; eventually it is intended that this system will be fully automated.

(iv) Software has been prepared to enable the office computer to read out the data disc produced during the inspection and print out a report listing all the numerical data in standard format. The covering letter making specific recommendations to the hospital Administrator can also be generated by the computer through word processing software which utilizes a menu of text files covering many commonly encountered deficiencies, supplemented by any necessary additional comments entered directly from the keyboard.

One of the sub-programs which provides data to this complete system is a mail out test package recently developed by the Radiation Safety Unit. Evaluation of the returned test package involves readout of TLD dosemeters and densitometry of the test radiograph. From these dose readings and film densities the microcomputer can calculate, using a "least squares" fit procedure, the half value layer for the X-ray beam and the slope of the H & D curve for the processed emulsion. This was formerly one of the most time consuming aspects of the work of the unit's physicists, and was therefore the first part which was fully automated following the acquisition of the computer.

Development of this computerised system has enhanced the quality of the service which the Radiation Safety Unit offers to hospitals in the Province, and we anticipate that it will enable the staff of the unit to monitor the effectiveness of quality assurance procedures in diagnostic X-ray departments much more closely than in the past.

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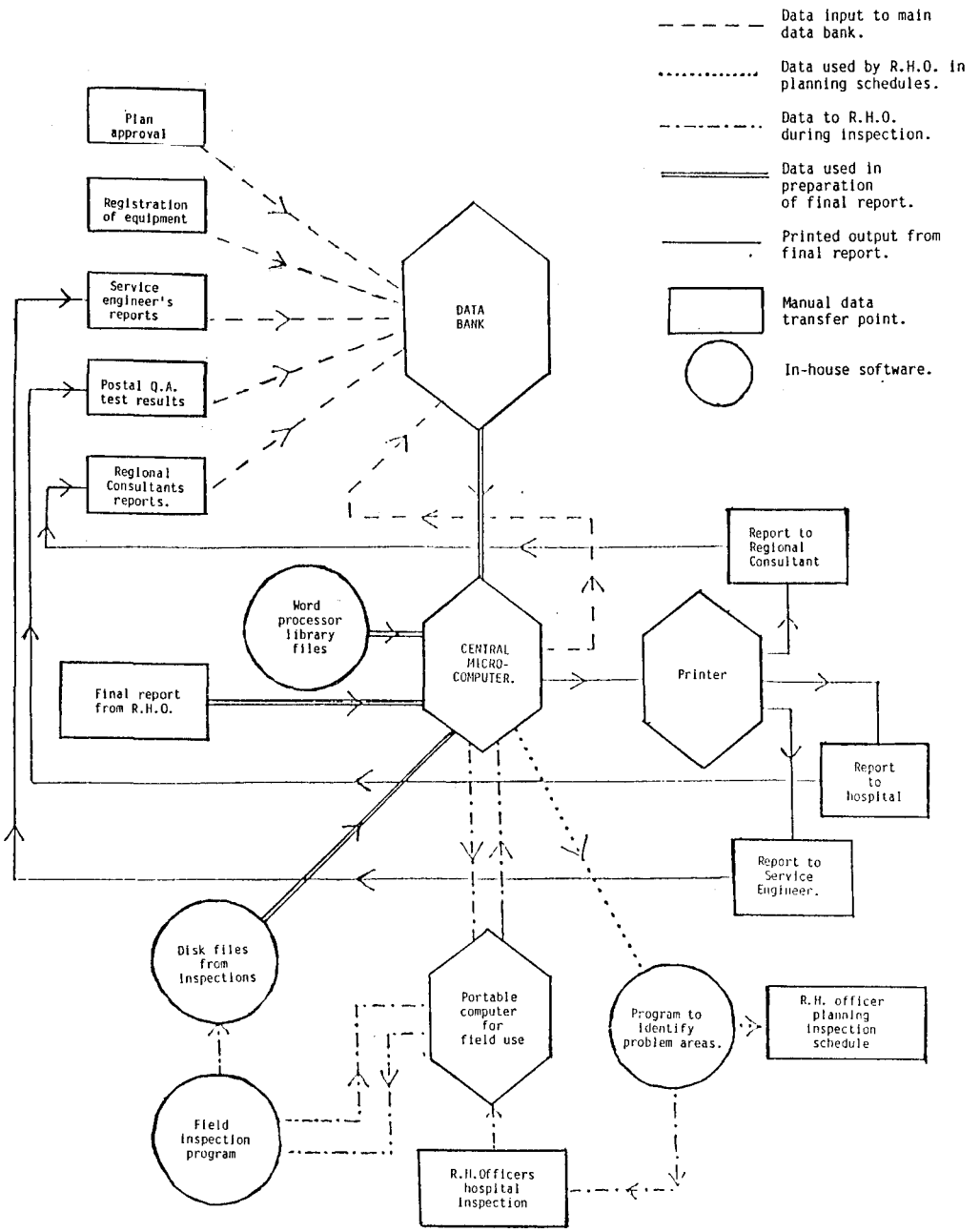
DATA FLOW CHART FOR THE

SASKATCHEWAN HOSPITAL RADIATION SAFETY PROGRAM.



CURRENT STATE OF A DOSIMETRIC EVALUATION PROGRAMME IN DIAGNOSTIC RADIOLOGY INSTALLATIONS IN SPAIN

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INTRODUCTION

The Medical Physics Group at the School of Medicine of the Complutense University of Madrid, started a programme on the study of radiation doses in relation to Diagnostic Radiology in the area of Madrid in 1986, in cooperation with the Department of Health and Consumer Affairs, and several Madrid area hospitals as well as some Outpatient Centers.

In Spain, the National Health Service (NHS) (through the National Institute of Health, INSALUD), potentially cares for the health of approximately 94% of the population. This figure reaches 99% at the Community of Madrid . Radiological examinations are performed mainly in Hospitals and Outpatient Centers (the latter making up a first link in the patient's radiological diagnosis). Private Diagnostic Radiology is used by the remaining 6% of the population (not taking into account the population attended in military hospitals), and by patients who in spite of having access to NHS Diagnostic Radiology Services, prefer to choose the private sector for different reasons.

Besides the data we obtained during the first year of study from four large Madrid-area hospitals (and a few outpatient centers); we have used data furnished by the Department of Health and Consumer Affairs, the INSALUD (1, 2) and other sources (3).

Hospitals involved in the project serve 36% of the population of the Autonomous Community of Madrid (CAM). The data available from outpatient centers, refer to 31% of CAM population. Those from the private sector were obtained from 16 centers (according to the documentation presented at the CSN by these centers, for licensing of their corresponding installations) (4).

The studies and reports from the National Institute of Statistics (INE) enabled us to know the population values of the different Autonomous Communities, their age distribution and number of offspring, which made it possible to derive the child expectancy as a function of age (which we assumed identical for the whole of the Spanish territory, due to lack of better data sources) and to calculate the genetically significant dose (GSD).

NUMBER OF RADIOLOGICAL EXAMINATIONS

The number of annual examinations per 1000 inhabitants in the Autonomous Community of Madrid has been estimated from detailed information obtained up to present as the programme runs its course

and confirmed with film area used, together with the data from Outpatient Centers contributed by different sources (1, 3). This figure has been extrapolated to the rest of Spain from the data supplied by the INSALUD (2).

The following figures could then be established with this information:

a) Comm. of Madrid: 4 825 000 inhabit.(12.4% of the whole of Spain)

200 examinations/1000 inhabit./year (in hospitals)
230 examinations/1000 inhabit./year (in outpatient centers)
150 examinations/1000 inhabit./year (in private sector)

TOTAL: 580 examinations/1000 inhabit./year
(not considering labor control, military or dental examinations)

b) Spain as a whole: 38 800 000 inhabitants

140 examinations/1000 inhabit./year (in hospitals)
223 examinations/1000 inhabit./year (in outpatient centers)
127 examinations/1000 inhabit./year (in private sector)

TOTAL: 490 examinations/1000 inhabit./year
(not considering labor control, military or dental examinations).

FREQUENCY OF SOME EXAMINATIONS PER 1000 INHABITANTS

In order to perform the analysis of frequency of hospital radiological examinations, we mainly used data obtained from the San Carlos University Hospital (HUSC), since we considered it a center not presenting distortions with regard to its functioning and patient population served during the year of the analysis (1986). Also, this hospital has a very good data processing center which enabled us to compare through different routes, some of the data obtained.

Examination frequencies in Outpatient Centers were analysed using data from reference (3). A total number of 302 893 examinations/year referred to 9 outpatient centers in Madrid. For the private sector, data from the CSN were used (4) likewise some other information supplied by our colleagues.

PATIENT DOSE ESTIMATION

Information on over 60 000 hospital examinations with details of the technical parameters employed, operator and equipment involved, place where performed, patient sex and age, besides a smaller number of examinations performed in Outpatient centers and private offices were obtained during the first year of the project.

Using mean and range values of these parameters (kVp, mAs, screening time, etc.) measurements were made on exposure at skin level (without backscatter) in several rooms of the participating centers and for a large number of examination types (6), with properly calibrated ion chambers. In some complex examinations

(digestive tract, urinary tract, etc.), measurements of the area x exposure product were also performed, by using DIAMENTOR equipment from PTW-Freiburg. Absorbed dose (in muscle) values at the entrance were obtained in urography (7) by positioning on patient skin LiF TLD-100 chips from Harshaw. This type of measurements shall be completed shortly in a coordinated manner among some E.E.C. states.

Also, organ dose measurements have been initiated on a REMAB phantom from Alderson, especially designed for Diagnostic Radiology, utilizing between 40 and 100 TLD-100 chips, for dose estimations in each radiological examination. We have results from digestive tract examinations (following introduction of the corresponding contrast medium into the stomach), chest, conventional chest tomography, computerized tomography and urology.

With all this information and keeping in mind the variations recorded within the same center and between the different centers participating in the project, we have made an estimation (which should still be considered at the preliminary level and not as final result), of the organ doses (using phantom measurements and tables from Jones and Wall (8, 9). It can be concluded in a first estimation that the mean effective dose equivalent per radiological examination in the Madrid area is 1.81 mSv, that means 1.04 mSv per inhabitant and year (table 1). GSD values of 0.2 μGy for chest and 17 μGy for adult urinary tract examinations were also estimated.

TABLE 1

ESTIMATION OF EFFECTIVE DOSE EQUIVALENT FOR DIFFERENT MEDICAL DIAGNOSTIC X-RAY EXAMINATIONS IN THE COMMUNITY OF MADRID

TYPE OF EXAMINATION	ESTIMATED EFF.DOSE EQUIVALENT (mSv)(a)	MEAN EFF. DOSE EQ. WEIGHTED BY FREQUENCY				
		Hospital (mSv)	Outpat. (mSv)	Private (mSv)	Total (mSv)	(%)
Skull	0.2	0.006	0.006	0.006	0.006	0.3
Spine	1.0	0.070	0.250	0.250	0.189	10.4
Chest	0.16	0.050	0.043	0.032	0.043	2.4
Abdomen	1.5	0.120	0.150	0.135	0.136	7.5
G.I. tract	10.2	0.255	1.020	1.122	0.781	43.1
Urology	7.0	0.105	0.210	0.210	0.174	9.6
Hip and pelvis	2.8	0.084	0.084	0.084	0.084	4.6
Extremities	0.1	0.015	0.014	0.020	0.016	0.9
CT	5.0	0.160	-----	0.100	0.080	4.4
Others	3.0	0.774				
	0.5		0.030	0.025	0.282	15.6
SUBTOTAL (mSv)		1.640	1.810	1.980	1.810	
EXAMINATION (%)		34%	40%	26%		
EFFECTIVE DOSE EQUIVALENT/INHABITANT				1.04 mSv		

(laboral, military and dental radiology not included)

Assuming that the 1.81 mSv value can be extrapolated --once properly corrected with the percentages of hospital, outpatient and private radiology-- to the national whole (1.71 mSv), the 490 examinations/1000 inhabitants and year would imply 0.84 mSv/inhabitant and year in Spain.

The collective dose equivalent which would be attributed to Diagnostic Radiology in Spain would then be 32 510 man.Sv.

This study was partially supported by the E.E.C. (contract DI6-0214-EA), the CAICYT, the Department of Health and Consumer Affairs and by the Nuclear Safety Council (CSN).

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TWO IMPORTANT FACTORS INFLUENCING THYROID INTERNAL DOSE FOR RADIOIODINE

lei Kui, Zhang Zhonghou and Li Shijun

ABSTRACT

Radioiodine urinalysis of 24-hr integral urine sample and extracount detection were carried on daily throughout 15-day period in kinds of thyroidal function women who received diagnosis ^{131}I(NaI). By these, the internal radiation doses for each subject were back-calculated. The results show: (1) the thyroid internal dose factors yield as a linear function of thyroid iodine uptake rates, $D_{Thy} = 2.75 + 15.2K_{Thy}$. (2) taking Methimazolum together with Thyroideum can accelerate ^{131}I release from thyroid, increase ^{131}I urine output and decrease the thyroid internal dose.

HOW SAFE CAN WE BE?

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To live safely does not mean to live without risk. We wish to take some risks in order to enjoy particular benefits. Other risks we cannot avoid. Our death, at some time, is not a "risk", it is a certainty. Our likelihood of dying before age 100 is close to 100 per cent, but also at young ages we run some risk per unit time of dying.

An empirical approximation of the age-specific death probability rate, G(u), as a function of age (u), is known as the Gompertz-Makeham expression. It can be written

$$G(u) = A\, e^{Bu} + C$$

where A and C are constants which differ between populations while the exponent coefficient B is usually about 0.1 per year.

Since we are all certain to die, there is no way in which protection can "save lives". All we can do is to save years of life and to give these years as much of mental and physical well-being as possible. This is a multidimensional objective and can hardly be expressed in terms of a number. We must never forget that saving years of life must be supplemented by commensurable efforts to make these years worth living.

The age-specific mortality may be the best quantity to express the risks in life in a quantitative way. From the objective point of view, it may be said that we have no cause for increased concern about our risk situation as long as G(u) does not change significantly. When asked about our annual risk of dying, we are likely to accept a two-digit answer, e.g. "0.45 per cent", rather than to insist on knowing whether it is, in fact, 0.448 per cent or 0.453 per cent. A total risk increment of the order of G(u)/100 might, therefore, be an objective "de minimis" risk. The lowest values of G(u) are found for ages of about 10 years, at which G(u) may be as low as 0.01-0.02 per cent per year in countries with low risk levels. The corresponding "de minimis" increment of the annual probability of dying would be of the order of one per million.

In countries with a low standard of living, the age-specific annual probability of dying at young ages is quite high, mainly because of malnutrition and infections. It is not unusual that the lowest value is as high as 0.1 per cent per year. It is obvious that priorities must be given to reducing the main causes of death but it would be wrong to set a correspondingly higher "de minimis" value for risks of concern in poor countries if it is agreed that the primary causes of death in these countries are unacceptable.

"Objective" de minimis values for risk increments of individual significance are often misinterpreted. One misconception is that such values could be applied to each source of risk, e.g. to radiation risks or even to a particular source of radiation risk

(e.g. by assuming that a source-related annual de minimis dose would therefore be of the order of 100 microsievert). However, it is not a straightforward exercise to derive a source-related de minimis value even if there might be general agreement on a de minimis value of total risk increment.

The problem is even more complicated. It is true that there is an objective de minimis value of risk increment in the sense that a smaller risk increment would not change the individual's overall risk situation significantly. But this does not necessarily mean that he is not concerned about a small risk and is willing to accept it unhesitatingly. Our willingness to accept a risk depends on many factors, such as our prospect of ensuing benefit, our view on the need and value of the source of the risk, our trust in those responsible for risk estimates and protection, and the degree of voluntariness. Few people would accept an unnecessary risk, even if very small, forced upon them for a cause for which they feel no sympathy and by persons they dislike or distrust.

Most people may feel that the loss of a penny or a cent is not worth any further thought, but if all the pennies lost by a large number of persons happened to build up an available asset, they might nevertheless think it worth-while to use it for a good cause rather than throwing it away. In a similar way, a very small risk to a large number of people may mean a mathematical expectation of a finite number of injuries. That number will be small and insignificant in comparison with the total background of harm, but it may not be small in comparison with the effort that could reduce it. It is the latter comparison that matters. If some harm, in the absolute sense, can be avoided at a reasonable cost (including the "cost" of evaluating the situation), why should it then not be avoided? This is the thought behind the ICRP principle of optimization of protection: to keep all exposures as low as it is reasonably achievable, economic and social factors being taken into account (1).

One method for optimization of protection is the differential cost-benefit analysis described in a number of ICRP documents (2). This method is based on minimization of "costs" exclusively related to protection, namely, the cost of protection and a detriment "cost" which is calculated as αS, where S is the collective effective dose equivalent and α is the amount of money that society is willing to pay, marginally, per unit of collective dose eliminated.

The value of α is supposed to be given by national authorities. The two methods usually mentioned for deriving this value are the "human capital" approach and the "willingness to pay" approach. These methods are generally applicable in protection against stochastic harm, not just in radiation protection. In the more universal approach α is replaced by a corresponding value q, which is the marginal sum that society is willing to pay in order to "save" a life in a statistical sense (to save perhaps twenty years). If r is the risk coefficient for radiation detriment (i.e. about 0.02 per mansievert), the implied value of q is α / r. It is often recommended that α be of the order of US$ 10,000 per mansievert, in which case q = $ 500,000.

In the "human capital" approach, q is derived on the assumption that it would be stupid for any government not to pay at least as much per life saved statistically as the per caput gross national product for the years saved. Any lower ambition level would be a direct loss of resources from the economical point of view.

In the "willingness to pay" approach, q is deduced from information on what people might be willing to pay in order to avoid risks to themselves.

Both these approaches have been criticized for setting a price on a human life. This is a valid criticism in the case of the human capital method, which is only defendable as a means of finding a lower limit for q. However, the objective of optimization of protection can also be seen as an attempt to save the maximum number of lives with the resources that have been made available. The fact that society's resources are not infinite does not imply a valuation of life but only a limitation of what is achievable (3).

Conceptually, therefore, there is a third approach for deriving the appropriate value of q. Let us assume that a parliament or a government decides to allocate a sum of money, Q, to be used for statistical life-saving. If protection is optimized on the basis of a very low value of q, the money will not be spent, because very few lives would be saveable on this policy. If, in contrast, the value of q is taken to be very high, for example equal to Q, the whole sum may be spent to save only one life. Between these extremes, there may be a value of q which would save the maximum number of lives. That is the value that should be used, and it would, in fact, be unethical not to use it.

Assume that the saveable number of lives, N(q), is a function of the value of q that is chosen ("saveable" means that a risk source may be eliminated so that individuals who would otherwise have died from that source will have the life expectancy that is normal at their age). If we then save all lives that can be saved with q as the marginal cost per life saved, we shall find that we have, on the average, paid less than q per life. If the average sum that we have paid is $a(q) < q$, the total expense for society is $N(q) \cdot a(q)$. We save the maximum number of lives if that sum is equal to the sum Q that we have at our disposal. This will be when $N(q) \cdot a(q) = Q$. If we knew the function N(q), we would thus be able to calculate q; however, that function is not known.

It is nevertheless possible to make some upper estimate of N(q). There is not an infinite number of lives available to be saved. We may be able to guess what the maximum number, M, may be. We also know that the average cost per life (a) cannot exceed q. We can therefore assume that, at the best choice of q, we have

$$M\,q > Q$$

From the Swedish statistics it can be seen that the parameters A and B in the Gompertz-Makeham expression have not changed much since the beginning of the century, while the parameter C has decreased by more than one order of magnitude (4). This is what one might expect, since the age-independent term would mainly

describe the death probability from accidents and infections, i.e. risks that seem to be controllable or "saveable".

One potentially controllable cause of death which is rapidly increasing in many countries is lung cancer. It shows a death probability which is strongly age-dependent. This seems to be masked by a decrease in other age-dependent causes of death. The value of C, therefore, is likely to somewhat underestimate the potentially avoidable risk. The value of C ranges from about 0.0003 per caput and year for males in "safe" countries to about 0.003 per caput and year in those "high-risk" countries for which statistical information is available (5). To "save a life" amounts to different things in these two cases. In a "safe" country it means, on the average, saving 10-20 years of life and in a "high-risk" country 30-40 years in spite of the fact that the mean life expectancy is shorter (the main life-saving in poor countries is in very young years). A more thorough analysis indicates that about 0.1 manyears per caput and year might well be "saveable" for both men and women in many countries, rich as well as poor. This order of magnitude estimate may suffice to give an indication of the implications of the inequality $M\,q > Q$.

For example, with q = $ 500,000 per life, as implied by the radiation protection practice, which may mean a cost ranging from $ 10,000 to $ 50,000 per manyear saved, and with M = 0.1 manyear per caput and year, the protection cost expressed per caput in the whole population would be less than a number ranging from $ 1,000 to $ 5,000 per year, probably being much less because all potentially "saveable" lives may not be saveable at these values.

If the marginal sum to be paid to save a manyear were chosen on the basis of the human capital method and therefore equal to the per caput annual GNP, the actual protection expense would be less than 1/10 of this GNP, i.e. less than a number ranging from $ 50 to $ 1500 per caput and year. The radiation protection ambition for rich countries, therefore, is not much higher than that behind the human capital approach. A rich country should therefore afford to adopt the radiation protection ambition in all fields of protection against stochastic risks, and the cost of doing so may not exceed a few per cent of the GNP. However, a poor country may find that the expenses drawn by such ambitions would be prohibitively high to meet without help. Any ethical problem will not relate to the optimization procedure, but to the political decision to allocate resources (Q), necessarily limited, for lifesaving purposes.

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OPTIMISATION IN MEDICINE : THE POTENTIAL AND THE PROBLEMS

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THE POTENTIAL

High profile uses of ionising radiations, such as nuclear power automatically get significant attention from the radiological protection community. Conversely the resources and attention paid to the use that has been with us the longest, medical diagnostic procedures, are lower. In the light of this it is perhaps instructive to look at the levels of individual and collective doses in the medical sector. In almost every country the largest component of the population dose from artificial sources is to patients from medical diagnostic procedures. Over the last few years the NRPB has carried out studies in the UK to determine the frequency with which various examinations are carried out [1] and the associated patient doses [2]. The total annual frequency of all X-ray examinations, excluding dental, for 1983 were 488 per 1000 inhabitants. Dental X-rays provided a further 156 per 1000 inhabitants.

Table 1
Collective doses to the Great Britain population from diagnostic medical radiology

Practice	Collective effective dose equivalent (man Sv)
Medical X-ray (excluding CT)	15 500
Computerised tomography (CT)	500
Dental X-ray	200
Nuclear medicine	950
All diagnostic radiology	17 150

From Table 1 it can be seen that the annual collective dose is of the order of 17 000 man Sv, which accounts for approximately 90% of the UK exposure arising from artificial origins. The individual effective dose equivalents from different types of examinations cover a tremendous range; from dental and chest radiography at ≤ 0.05 mSv per examination to between 5 and 50 mSv for barium enema X-ray examinations and nuclear medicine scans of the thyroid and pancreas. The per caput annual dose was about 0.3 mSv. A particularly important part of the studies was the determination of the distributions of doses for the same types of examinations conducted on different patients and in different hospitals; examples are shown in Fig. 1.

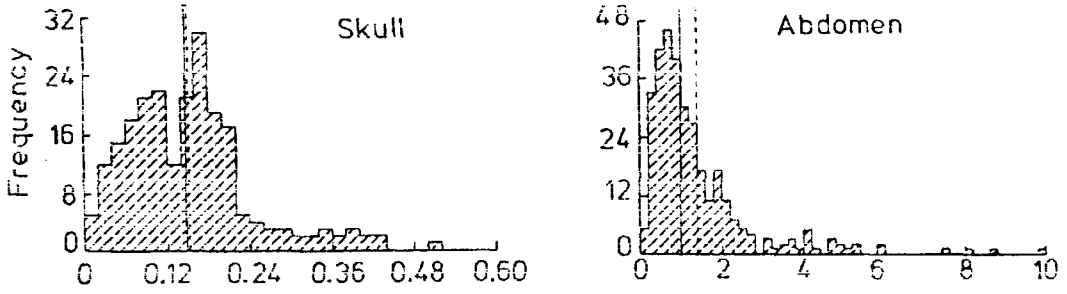

Fig. 1 Examples of distributions of effective dose equivalents for adult patients undergoing simple types of examinations

It is clear that there must be potential for eliminating the high doses from the distribution of patient doses and thus shifting downwards the mean doses for each type of examination. Given the size of the collective dose from medical examinations this potential assumes very significant proportions. For example, every 1% reduction in the annual collective dose is just under twice the current annual collective dose in the UK from occupational exposure in the nuclear power industry (92 man Sv). Many patient dose reduction options, e.g. rare earth screens and carbon fibre cassette facings, grid facings and table tops are known to be cost effective but have not been widely used. Indeed in the statement from the ICRP Paris meeting, 1985, the Commission positively expressed regret that such changes were not being introduced as rapidly as possible. It is also relevant to note here the striking disparity between the sums spent on radiation protection in the nuclear power industry (£10^4 to £10^6 man Sv^{-1}) and those spent in medicine £10 to 10^3 man Sv^{-1}.

ADDRESSING THE PROBLEMS

The numerical values of the dose limits do not apply to patient exposure but the basic principles of Justification and Optimisation do apply, and as shown above there is a significant potential for dose reduction. In the light of this the NRPB has a programme of work with two broad elements that are attempting to focus attention on this area. Firstly, it is considered that in common with other sectors the use of decision aiding techniques such as cost benefit analysis (CBA) together with formal ALARA Audits, have a valuable role to play in the assessment of competing protection options and indeed in decisions on justification and the allocation of resources. The ALARA Audits are essentially systematic and regular reviews to determine the relevance and adequacy of the existing radiation protection measures. They provide a mechanism for identifying problems that need to be addressed, whilst CBA, used within a structured 'ALARA Procedure' [3] provides a means to aid the subsequent decisions that have to be made. Fundamental to the use of CBA is the need for a value of unit collective dose that

is appropriate in the context. Russell and Webb [4] have recently considered the special factors that apply here, in particular

1. the magnitude and range of effective dose equivalents associated with particular types of examination

2. the age distribution of the irradiated population

3. the direct benefit to the patient

4. the general view of medical exposures as beneficial.

The age distribution is particularly important. In diagnostic radiography, exposure is not uniform throughout life. In children in their first decade, the frequency of examination is about one third of that found in those aged over 60, and in general the radiation dose for an examination is less in children. A large proportion of a lifetime's exposure to diagnostic X-rays occurs in the last illness when radiogenic cancer is irrelevant. Conversely, the doses given in paediatric and obstetric radiography will be associated with higher somatic and genetic risk factors than would apply to an average population. Russell and Webb concluded that a value in the range £5,000-£10,000 man Sv^{-1} would seem reasonable for most purposes, i.e. a general radiography department. However, doses in paediatric and obstetric departments should be valued 5 times higher, in the range £25,000-£50,000 man Sv^{-1}.

Although many optimisation case studies have been done in the nuclear power sector, there are relatively few examples in the medical sector. The NRPB has therefore put effort into encouraging such studies. One aspect of this has been to demonstrate the technique by performing a case study [5] for a hypothetical 6 room X-ray department considering expenditure on the protection options of slot radiography, rare earth screens, carbon fibre table tops, grid facings and interfacings, etc. This produced incremental cost effectiveness figures of between £2 and £8,900 man Sv^{-1}. Apart from indicating the desirability of implementing the options it also provided a ranking or implementation priority listing. Further studies are being planned and it is hoped that such work will encourage others to put the techniques into practice. Although the main focus of such studies is likely to be on equipment and facilities, it is felt that the techniques could also be used in quality assurance, training and enforcement programmes, and as part of the justification process when considering routine X-ray screening programmes.

Whilst encouraging further studies, it must be recognised that within the medical sector there is reticence to accept the use and validity of these techniques. This is probably because many of those with direct budgetary control are making overt short term life and death decisions daily and do not recognise a reduction in long term detriment as a sufficiently real source of income to be balanced against direct expenditure. This problem may partly be attached by looking towards the higher decision levels concerned with resource allocation, but at a

practical level we need to integrate protection with existing facets of medical programmes. This is where the second element of the NRPB's programme of work is relevant. Quality Assurance (QA) programmes are becoming widely implemented in radiological departments, partly due to the radiologist's prime interest in image quality. The encouragement of routine patient dose monitoring as a part of QA programmes is considered of prime importance. Also to demonstrate the cost effectiveness of radiation protection measures relative to medical procedures, other economic techniques are being explored. Cost utility analysis provides one method of doing this in terms of the cost per life year saved and this has the advantage of being a technique that is already in use to some degree within medical circles.

In common with other sectors, the availability of good data bases is a major problem and if we are to make use of the various techniques, action is required now to build these up, particularly in relation to workload and doses per examination. Work is in hand to develop a simple patient dosimetry protocol to enable hospital X-ray departments routinely to monitor patient doses as part of their quality assurance programme. Two of the options being pursued are entrance skin dose measurements using a TLD system and the use of Diamentor ionisation transmission chambers to deduce the total energy imparted. Shrimpton et al. [2] have shown that there is reasonable correlation between the total energy imparted and the effective dose equivalent over a range of two orders of magnitude and for examinations of widely varying complexity.

CONCLUSION

Overall the control of patient exposure from diagnostic radiology has not achieved the objective of ALARA and significant potential exists for dose reductions. This is an area that warrants increasing attention from the radiation protection community.

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INTERNATIONAL APPROACH TOWARDS TREATING RADIATION EXPOSURE WITH PROBABILITY LESS THEN ONE

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ABSTRACT

In recent years there have been growing concerns on the 'interface' between the systems of control of normal and accidental exposures to ionizing radiation. The ICRP has developed consistent radiation protection principles applicable to exposures which are assumed to occur with unit probability during normal operation of a radiation facility. Based on the ICRP principles the IAEA prepared, jointly with the WHO, ILO and the OECD/NEA, the Basic Safety Standards for Radiation Protection, Safety Series No. 9, 1982.

The situation is rather different for exposures which may or may not occur with a given probability and this usually refers to accident conditions. So far, this problem has been discussed internationally only in the case of radioactive waste repositories. Therefore, generally applicable international recommendations, guides or norms do not exist that are consistent with the ICRP principles for actual exposures from normal operations.

The paper discusses the current IAEA efforts to develop guidelines for a unified approach for the application of radiation protection principles to radiation exposures assumed to occur with certainty and exposures which are not certain to occur. A useful criterion is that of a limit on individual risk. A simple approach would be to define separate limits for normal and accident situations. For waste disposal ICRP has suggested a risk limit for accident situations to be of same magnitude as that for normal operation. IAEA is considering a risk limit of 10^{-5} in a year for consistency with general safety standards of dose limitation. ALso, is needed a source-related upper bound which has to be apportioned from the risk limit in order to take into account the presence of other sources.

ACCIDENTAL RELEASE OF UF_6 AT THE SEQUOYAH FUELS CORPORATION FACILITY AT GORE, OKLAHOMA, U.S.A.

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INTRODUCTION

On January 4, 1986, a 14-ton UF_6 cylinder ruptured at the Sequoyah Fuels Corporation (SFC) site at Gore, Oklahoma, U.S.A., resulting in a massive release of uranium hexafluoride (UF_6). One SFC worker was killed and some workers were hospitalized. About 100 offsite individuals were admitted for medical "screening" in a local hospital.

Shortly after the accident, the Nuclear Regulatory Commission (NRC) gathered a group of experts (the Ad Hoc Interagency Task Force) to assess the potential health impact to the general public from this accident. In addition, the NRC formed a group on "Lessons Learned from the Accident" to review and improve the protection of the general public from future accidents.

A description of the accident, the public health impact, development of procedures for hospital "screening", suggestions for future improvements, and the requirement for emergency planning for this type of facility is discussed in the following sections.

DESCRIPTION OF THE ACCIDENT

On Saturday, January 4, 1986, at approximately 11:30 a.m., a cylinder containing about 29,500 pounds of uranium hexafluoride (UF_6) ruptured at the SFC site, releasing its contents into the atmosphere. According to statements from SFC officials, plant employees had inadvertently overfilled a 14-ton cylinder (maximum limit is 27,560 lbs.) and were in the process of heating the cylinder to facilitate removing the excess UF_6 when the rupture occurred.

As a result of overpressurization, the wall of the 12-foot long cylinder split over a length of about 52 inches parallel to the axis of the cylinder forming an opening about 8 inches wide at the midpoint of the split. Because of the high pressure in the cylinder, the large size of the opening, and because the cylinder rotated so that the split was on the lower side, much of the UF_6 came out of the cylinder rapidly.

Released UF_6 and its reaction products, UO_2F_2 and HF, made a dense white cloud which, pushed by the wind, quickly engulfed the process building (the Sequoyah Facility's principal structure) and formed a plume expanding to the south-southeast of the plant site. At the time of the incident, there were 42 workers at the plant site. One employee was in a scrubbing tower adjacent to the release point; his location was promptly enveloped in the initial plume from the release, and he died within a few hours of respiratory injuries. Approximately 37 people, mostly company employees, were hospitalized as a result

of the incident, but only a few were diagnosed as suffering from respiratory problems related to inhalation of hydrogen fluoride. About 100 offsite individuals reported to the local hospital for medical "screening".

PUBLIC HEALTH IMPACT

On January 14, 1986, the Nuclear Regulatory Commission (NRC) gathered a group of experts (the Ad Hoc Interagency Task Force) to assess the potential health impact to the general public from this accident. A report was published in March 1987 (NUREG-1189).[1] Based on the available data at that time, the Ad Hoc Interagency Task Force made the following conclusions and recommendations:

SFC Workers

The workers involved in this incident encountered exceptionally high uranium exposures and intakes (as estimated by urine uranium levels). These exposures were at levels at which one would expect from past experience at least some temporary renal (kidney) injury in a significant number of those exposed. No conclusions could be drawn with regard to more permanent and serious injury to some of the more highly exposed workers because they were exposed to levels at which there are no reliable data relating to human exposures.

The Ad Hoc Interagency Task Force recommended that workers exposed during the incident should be monitored carefully for evidence of pulmonary and renal injury. Medical surveillance of these individuals should be performed for a period of 1 to 2 years or until such time that renal and/or pulmonary function is determined by physicians to be acceptable in their professional opinion. These individuals should receive periodic evaluations, preferably every 6 months.

The maximum uranium intake among onsite workers was approximately 28 mg. The estimated effective whole-body equivalent dose for this maximally exposed worker was about 0.46 mSv (46 mrem).

Offsite Individuals

Around 100 offsite individuals submitted urine samples; of these, most showed no evidence of exposure, while 7 received low-level uranium exposure. The intakes for these individuals ranged from 0.1 to 0.9 mg uranium. The effective whole body equivalent for the maximally exposed offsite individual was about 0.014 mSv (1.4 mrem). This is insignificant compared with background radiation of 1.06 mSv/yr (106 mrem/yr) in the area. The population dose for the affected population sector to 80 km (50 miles) from the site was about 0.0048 person-Sv (0.48 person-rem) which was insignificant compared with the annual natural background radiation of 8.4 person-Sv (840 person-rem) received by the same affected population.

Offsite Contamination

A comprehensive survey was conducted after the accident covering affected areas up to 10 miles from the site. The survey included soil, vegetation, and surface water sampling, and aerial and in-situ radiological surveys. The results indicated that no significant uranium or fluoride deposited offsite. Most of the contamination was found onsite within the site boundary.

LESSONS LEARNED FROM THE ACCIDENT

After the accident, the NRC formed a Lessons Learned Group to review and improve the protection of the general public from future accidents. A report (NUREG-1198)[2] titled "Release of UF_6 from a Ruptured Model 48Y Cylinder at Sequoyah Fuels Corporation Facility, Lessons-Learned Report" was published in August 1986. Some of the major recommendations have been reflected in the following topics.

EMERGENCY PLANNING FOR FUEL CYCLE AND OTHER RADIOACTIVE MATERIAL LICENSEES

In the NRC's current proposed rules for emergency preparedness for fuel cycle and other radioactive material facilities (nuclear power plants are not in the same category)[3], the proposed requirements for emergency planning if it is needed, should include the following items:

1. Facility description: A brief description of the licensee's facility and area near the site.

2. Types and classification of accident: An identification of each type of accident for which protective action might be needed.

3. Detection of accident: Identification of the means of detecting each type of accident.

4. Mitigation of consequences: A brief description of the means and equipment for mitigating the consequences of each type of accident.

5. Assessment of releases: A brief description of the methods and equipment to assess releases of radioactive materials.

6. Responsibilities: A brief description of the responsibilities of licensee personnel should an accident occur.

7. Notification and coordination: A commitment to and a brief description of the means to promptly notify offsite response organizations and request offsite assistance, including medical assistance for the treatment of contaminated and/or injured onsite workers when appropriate.

8. Information to be communicated: A brief description of the types of information on facility status radioactive releases and recommended actions, if necessary, to be given to offsite response organizations and to the NRC.

9. Training: A brief description of the training the licensee will provide workers on how to respond to an emergency and any special instructions and orientation tours the licensee would offer to fire, police, medical, and other emergency personnel.

10. Recovery: A brief description of the means of restoring the facility to a safe condition after an accident.

11. Exercises and audits: Provisions for conducting quarterly communication checks with offsite response organizations and annual onsite exercises to test responses to simulated emergencies.

PROCEDURES FOR MEDICAL SCREENING FOR ACCIDENTAL EXPOSURES

After the SFC accident, it was determined that emergency chemical and radiological bioassay procedures to evaluate the exposure of individuals to these materials under accident conditions need to be developed. The NRC has contracted the Brookhaven National Laboratory (BNL) to develop emergency bioassay procedures for medical "screening" for accidental exposures to alpha-emitters. A table is to be developed for each radionuclide, (UF_6, Am-242, Pu, Cm, and Po-210,) showing the recommended bioassays. The table will outline the bioassay program and follow-up action recommended for three groups of people by level of exposures: (1) people with exposures believed to be far above 10 CFR Part 20 limits; (2) people believed to have significant exposures, but not greatly above Part 20 limits; and (3) people believed to have low or no exposure. Guidance will be provided for determining assignment to these groups. For each exposure group, the recommended bioassay schedule should be presented for the following time periods: (1) the first day; (2) day 2 and 3; and (3) day 4 through 7. Beyond 7 days, it is anticipated that the NRC and the licensee will be able to assemble whatever resources are necessary to deal with the specific accident. Instructions concerning proper specimen submittal, prevention of specimen contamination, and specimen preservation will be included so that appropriate measurements can be performed either promptly or later. The work will be published as an NRC NUREG report.

CONCLUSION

After the SFC accident, the NRC has made additional requirements on the emergency planning for fuel cycle and other radioactive material licensees. Guidelines are being prepared to provide procedures for medical "screening" to further improve the protection of the general public from future accidents.

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REAC/TS: A COMPREHENSIVE PROGRAM FOR MEDICAL MANAGEMENT OF RADIATION ACCIDENTS

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ABSTRACT

The events immediately following the Three Mile Island (March 28, 1979) and Chernobyl (April 26, 1986) nuclear power plant accidents caused a considerable reassessment of medical planning and mass casualty problems respectively. Emergency planners in countries relying on electricity generated by nuclear power, and many other countries as well, have undoubtedly reviewed and are revising emergency response planning concepts as a result of the Chernobyl accident. This paper reviews the readiness for nonmilitary nuclear disaster management as a part of the overall programs of the Radiation Emergency Assistance Center/Training Site (REAC/TS) located in Oak Ridge, Tennessee. The REAC/TS program is funded by the U.S. Department of Energy, the U.S. Federal Emergency Management Agency, and also serves as the Western Hemisphere Collaboration Center as designed by the World Health Organization (WHO). The current status of trained (over 2,000) physicians, nurses, paramedics, and health physicists will be reviewed as well as data from the REAC/TS Registry system for documentation of worldwide radiation accidents. The level of REAC/TS involvement (in Moscow, Kiev, Bucharest, and Warsaw) following the Chernobyl accident will also be discussed. Lessons learned from medical problems associated with the Chernobyl nuclear power plant accident will be emphasized as well as their implications for future response in the event of similar accidents.

This work is based on work performed under Contract No. DE-AC05-76OR00033 between the Department of Energy, Office of Health and Environmental Research, and Oak Ridge Associated Universities.

Medical treatment of whole-body overexposed individuals

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1. Introduction

A severely irradiated casualty is a person who has received a radiation dose of a level such as to endanger his or her life. The whole-body dose which kills 50 % of victims within two months lies somewhere between 2.5 and 5 Gy [1, 2]. The differences in dose separating probable survival (mortality lower than 10 % for example) from more or less certain death (mortality of over 90 % for example) is small because the dose to the bone marrow has to increase from 2 to 5 Gy. The hematopoietic bone marrow is one of the most sensitive organs to radiation. After a latent period (the higher the dose, the shorter this phase), signs of medullary hypoplasia or aplasia appear ; their intensity is related to the mean absorbed dose to the bone marrow.

Irradiation is never uniform when an accident happens. The mean dose reflects an average level but areas which have been heavily exposed are critical points which can exacerbate the clinical pattern. The heterogeneity of irradiation is a very important factor, because it determines whether spontaneous medullary repair can take place or whether a bone marrow transplant is advisable. Even a few, small isolated patches of marrow which have received doses of less than 2 Gy will be enough for the repopulation of heavily irradiated areas. Repopulation will be possible only if the stroma and microenvironment have not been destroyed by excessively high doses. As with any damage to the bone marrow, priority when treating a severely irradiated casualty must go to prevening and curing the consequences of the insufficient production of the damaged blood cell lines. In addition to dealing with hematopoietic deficiency, the two main forms of treatment are therefore the prevention and treatment of infection and maintaining the nutritive and electrolyte balance.

The irradiation of the intestine with doses of more than 6 Gy results, within two weeks, in a serious digestive syndrome, the main complications of which are perforation, invagination and occlusion. The irradiation of the brain with doses higher than several tens of Gy can kill the patient in a few days. When doses are lower, the neurological symptoms are transitory and are mainly due to cerebral oedema.

The survival of a radiation victim therefore depends (1) on the absorbed dose to vital sensitive organs (human reproductive organs are extremely sensitive to radiation, but damage to them does not affect chances of survival), (2) on the distribution of irradiation in the body and (3) in time (for the same level of severity, the dose must be doubled or quadrupled if it is spread over two weeks or one month). Prognosis is based on clinical signs which reflect the extent to which to various sensitive systems are affected :

- brain : headache, obnubilation, or even coma and neurovegetative dysfunction, irregular electro-encephalogram ;
- intestine : nausea, stomach pains, diarrhoea, digestive complications ;
- blood : . platelet deficiency with the outward symptoms of hematomata, petechiae, ecchymoses and hemorrhaging,
  . anaemia with the external symptoms of pallor, dyspnea and tachycardia,
  . leukocyte deficiency with local or generalised superinfection or even septicaemia.

The level of the mean absorbed dose to the bone marrow is assessed from paraclinical tests such as working out the speed of decline of lymphocytes and platelets or the level of chromosomal abnormalities in lymphocytes in the circulating blood and estimating the level of exposure of various medullary areas from the results of marrow punctures in various carefully chosen areas and of cultures of stem cells, to mention only the main tests.

2. Treatment of hematopoietic damage

- *The problems : anaemia and thrombopenia*

Substitution therapy is the only treatment which can correct disorders connected with anaemia and thrombopenia [3, 4, 5]. Hemoglobin must be kept at a level which makes it possible to remedy disorders due to anoxia (limit approximately 7 to 8 g/100 ml. There is a distinct risk of hemorrhage once the level of platelets drops below 50,000/µl. It is a major risk below 10,000/µl. Platelet transfusions are carried out if these values are reached and in the event of a hemorrhagic syndrome, which may be revealed by hemorrhagic blisters, hemorrhages at the back of the eye, hemorrhaging in the skin or mucous membranes or visceral hemorrhages.

- *Transfusions*

A complete check must be made before the transfusion in order to reduce the risk of immunological complications, this includes determining or confirming ABO groups, the Rhesus factor together with the complete phenotype in other blood group systems and tests for irregular agglutinins and the HLA complex. The transfusion products consist of phenotyped packed red cells with a low leukocyte content and with compatible HLA platelets from a single donor. One red cell pack must contain at least 45 g of hemoglobin. In an adult, the transfusion of 200 to 250 ml of packed red cells gives a 2 to 2.5 % rise in the hematocrit. Platelets increase by approximately 10,000 per transfused unit and per m^2 of body surface. The frequency of transfusions varies with the daily readings of blood counts, the purpose being to maintain a level above that at which anaemia or hemorrhages could occur. Tests must be carried out for antibodies in the various systems one week after transfusion. No product may be taken from a person who might be a donor of a bone-marrow transplant, because a transfusion can lead to the formation of antibodies against the donor's private antigens.

- *Bone-marrow transplantation*

Bone-marrow transplantation has become enshrined in the literature as a proposed final successful therapy for the Acute Radiation Syndrome. In the **Chernobyl** accident, bone-marrow transplantation was attempted in 13 patients with one survival. None of the transplants have taken definitively. In the opinion of USSR physicians, this therapy resulted in worsening of the conditions of their patients and they advice against its use [6, 7, 8]. The overall marow transplantation results suggest that it will be of benefit to few exposed individuals in similar types of radiation accident [9, 10]. However, it is impossible to say for certain whether the transplants carried out in Moscow were warranted, as many cases among the most heavily irradiated developped complications chiefly due to the associated burns. It must be borne in mind that a bone-marrow transplant must still be regarded as an *exceptional measure* and that it is *dangerous* [6, 11, 12]. The decision as to whether it is indicated must be taken as soon as possible, for any delay prejudices the chances of the transplant taking (blood transfusions, infections, etc...).

The indication for transplanting bone-marrow to a radiation casualty rests on [10, 11, 13, 14] : (1) high doses received by all medullary areas and which are incompatible with spontaneous recovery ; (2) the absence of any relatively protected areas (less than 4 Gy) which could provide a basis for hematological recovery ; (3) the results of the immunological examination ; (4) a very poor general state, reflecting a very low tolerance of aplasia. If doses at these levels have been received, it must not be forgotten that the neurological and digestive syndromes are likely to cause grave concern and to require more urgent treatment than the hematopoietic syndrome.

The possibility of conducting a transplant must first be discussed bearing in mind the biological criteria of serious medullary aplasia combined with : (1) reticulocytosis of less than 20,000/µl ; (2) neutropenia below 500/ µl ; (3) thrombopenia below 10,000/ µl ; especially if these values are continuing to fall. A transplant is indicated only if marrow punctures at different places which are assumed to be the least damages (following the establishment of a dose-rate contour map based on the reconstruction of the accident) show a barren marrow after an interval of several days. Moreover, these examinations must be backed up by medullary biopsies, stem cell cultures and blood cell cultures which show whether or not any repair is in progress. As the results of these

examinations are not known for one to two weeks, they must be carried out as soon as possible. All the criteria for selecting the donor have now been established and the same rules must be followed as with bone-marrow transplants based on more traditional indications.

3. Preventing and treating infections

- *Prevention*

The risk of infection is probably the most therapeutic challenge initially confronting the patient. The patient must be kept in a protected environment. No matter what techniques are used (from taking ad hoc isolation measures to laminar flow beds or sterile rooms), the situation is critical and necessitates measures against exogenous and endogenous infections.

Exogenous contamination can come from the immediate environment or from food. As far as the first source is concerned, strict isolation must be coupled with the observance of extremely strict rules and the daily disinfection of any traps in drains. As regards the second source, it is illusory to hope that sterile food or even food with a low germ content can be obtained by using the hospital's catering system. The preparation of sterile food should therefore be arranged. It must be varied as these patients rapidly become anorexic.

The risk of endogenous contamination must be assessed as soon as the patient is admitted. It is standard practice to carry out digestive decontamination which must be closely supervised and adapted during aplasia. The treatment must be both anti-bacterial and anti-fungal. The former is selected in the light of the results of the coproculture. The latter is especially important because damage to flora from antibiotic treatment encourages the settlement of yeasts which colonise the upper respiratory tract. From here, they will regularly reinvade the alimentary canal. The oral cavity must be alkalinised and the yeasts must also be eliminated before the bacterial vacuum is created by removing enterobacteria. A residue-free diet helps to reduce secretions. Nothing must be prescribed to slow the speed of passage in the event of diarrhoea.

- *Treatment of infection*

Close supervision (at least every two days) based on coproculture and samples taken from the pharynx makes it possible to detect the emergence of a potentially pathogenic strain of bacteria. If there is fever, the source of the infection must be sought as a matter of urgency, especially if there is severe granulocytopenia. The treatment is traditional : a massive dose of antibiotics is always given at the site of the infection in accordance with the disc sensitivity test. Granulocyte transfusions are reserved for patients suffering from severe granulocytopenia, if the signs of infection continue after one or two days of intensive antibiotic therapy which has been especially worked out for that patient's needs or if the patient's life is threatened by an acute infection.

4. Maintaining the nutritive and electrolyte equilibrium

Whole-body irradiation with a lethal dose leads to a syndrome of serious undernutrition. The nutritive and caloric equilibrium must therefore be maintained without waiting until undernutrition sets in [5, 6, 13]. Normal feeding through the mouth is preferable to parenteral feeding whenever possible : it is physiologically natural and is psychologically reassuring. During the critical phase, it is often impossible to avoid artificial feeding. The sound is introduced deeply into a vein in absolutely aseptic conditions. An adult needs approximately two litres of liquid per day : the caloric intake is gradual and is maintained at 2,000 to 3,000 calories per day, made up of 50 % carbohydrates, 30 % of lipids and 20 % of proteins. Regular metabolic supervision is necessary and must form part of the daily examination so that any losses can be offset by adding the appropriate substance [5, 13]. Hydroelectrolytic supervision comprises a daily electrolytogram of the patient's blood and urine and the determination of the blood plasma effective osmotic pressure.

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EMERGENCY PREPAREDNESS : MEDICAL MANAGEMENT OF NUCLEAR ACIDDENTS INVOlVING LARGE GROUPS OF VICTIMS

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1. INTRODUCTION

The treatment of overexposed individuals implies hopitalization in a specialized unit applying hematological intense care. If the accident results in a small number of casualties, the medical management does not raise major problems in most of the countries, where specialized units exist, as roughly 7 % of the beds are available at any time. But an accident which would involved tens or hundreds of people raises much more problems for hospitalization. Such problems are also completely different and will involve steps in the medical handling, mainly triage, (combined injuries), determination of whole body dose levels, transient hospitalization. In this case, preplanning is necessary, adapted to the system of medical care in case of a catastrophic event in the given Country, with the main basic principles : emergency concerns essentially the classical injuries (burns and trauma) - and contamination problems in some cases -, treatment of radiation syndrom is not an emergency during the first days but some essential actions have to be taken such as early blood sampling for biological dosimetry and/or for HLa typing.

2. TRIAGE

This term refers to the sorting of patients into classes of injury and/or disease for the purpose of expediting clinical care. One of the main tasks is to determine the emergency of the cases. A simple way as a first step could be to divide the virtually exposed groups into three main categories :

- individuals overexposed or not, with combined injuries (wounds, burns...) and/or contamination ,
- individuals who are likely to have received doses at such levels that they will require medical management at some degree,
- individuals who have received only low doses and who are free of any other injury.

(i) individuals overexposed or not with combined injuries : these victims should be managed as in the case of conventional emergency. They do not represent a medical emergency for treatment of radiation injury, but some specific investigations should be carried out without delay in order to assess the dose and to get the necessary data for the treatment which will be initiated later. In France, the management is based on the set-up of :- "Petite Noria" : a shuttle constituted of first aid specialist who have in charge evacuation of the victims from place of accident to the "Poste Médical Avancé".

- Poste Médical Avancé : (Medical Advance Unit) : It has in charge emergency care after quickly established diagnosis according to a preplanned file (Fig. 1).

Results are transfered to a medical regulation center which finds the hospital and special unit needed. At the same time or during evacuation, a file concerning radiation problems is filled up (Fig. 2). This medical unit is constituted by Medical staff of FIREMEN or SAMU (Mobile Emergenccy Medical Unit - Assistance Médicale d'Urgence) from the regional or national levels.

(ii) - individuals who have or are suspected to have received doses. On the same way, preplanned actions have to be set up. In France, these victims will be regrouped in a center where they will be treated as described on the same file (Fig. 2) for radiation indications and which presents two parts. The first one is fulfilled by technicians. The second one by the medical staff. A first triage is made at this level in three categories. Global overexposed, local overexposed and internal contaminated. At the same time, the medical regulation center checks the available medical facilities in the National Public Hospitals. It is admitted and planned that during the first days (day 1 to + day 10), they can be hospitalized in classical medical units provided that the necessary follow-up and examinations are carried out. For this purpose, a special file will be available giving the basic examinations that should be made : blood typing on day 1 (A,B,0, rhesus, erythrocytes phenotype), HLA-A,-B-DR typing on day 1 and lymphocytes freezing on day 1, cytogenetic dosimetry on day I if not done on day 0, electroencephalogram for dosimetry on day 1. A further classification based on results of clinical, biological and dosimetric examinations will be made by an expert medical group in the four classical categories of dose levels; on these basis, decisions for hospitalization will be taken as follows :

< 1 Gy = hematological and clinical follow-up as out patients in an hospital.

1 - 2 Gy = follow-up in a general medical department.

2 - 4 Gy = strict follow-up in an hematological department, and with specific instructions in case of unfavorable hematological evolution.

> 4 Gy = hospitalization in hematological intensive care department (controlled rooms).

(iii) - individuals who evidently have received low doses. Their hospitalization is not justified but during a few days they have to be followed as out patients.

3 - CLASSIFICATION IN DOSE LEVELS CATEGORIES

For this purpose, it seems that three important steps may be defined :

- 1 : very early classification based on clinical symtoms and eventual data on physical dosimetry.

- 2 : confirmation and more precise classification based on hematological counts.

- 3 : later confirmation by biological dosimetry.

- Early classification : nausea, vomiting, diarrhea and fever are the most important clinical findings. They should be carefully recorded, including their time of appearance, frequency and severity. By this mean, it is possible to classify at least in 2 categories (less or more than 2 Gy).

- Secondary classification : hyperleucocytosis within the first hours if any early blood cell count is available, speed of decline of lymphocytes and platelets within the first 2 or 3 days, will allow a good classification within the categories above 2 Gy.

- Third step : it should be considered as a confirmation of the previous one, using essentially chromosome analysis, and neurophysiological response (EEG), and possibly some hematological specific technics.

The first two simple steps have been found effective in all radiation accidents. At Chernobyl, where over 350 persons were examined in the first 36 hours, these procedures were specially practical and proved their efficiency. But the third step which will lead to accurate estimates of dose and dose distribution, completed by other examinations such as bone marrow quantitative scintigraphy, stem cells cultures, is essential for decision on treatment specially in case of indications for bone marrow graft. This step can be foreseen because only a few tens of victims will rely on this kind of treatment.

4 - CONCLUSION

Preplanning the medical response after the alert is given should be done in details, file and records have to be prepared. The efficiency of the planning will depend on two essential points : it has to be incorporated in the usual medical planning for catastrophic events and its performances have to be tested in real size exercices. The basis for preparedness should be : use of three categories for triage, classification of overexposed individuals in four ranges of doses based on simple, rapid and reliable method by well trained specialized physicians. The decision on the hospitalization methods, i.e. small groups of patients in several hospitals or all patients in a centralized hospital will depend upon national resources and possibilities, but the need of specialized units in hematological intensive care may imply international cooperation. Another important point is that such preplanning implies extensive formation and education of the medical and non-medical staffs involved.

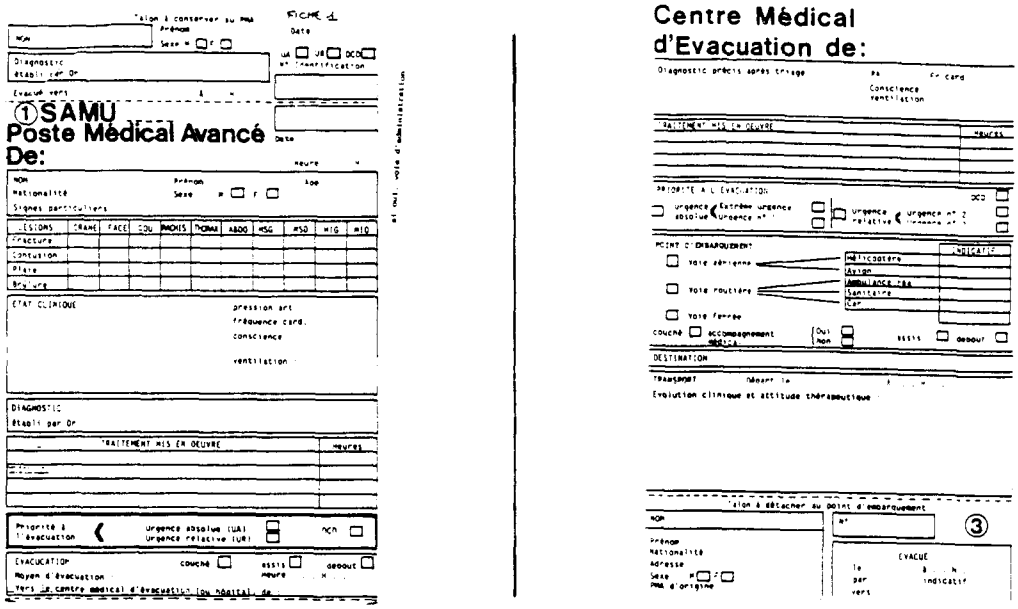

Fig:1

FICHE D'IDENTIFICATION MEDICALE EN CAS
D'IRRADIATION ACCIDENTELLE DE GRANDE AMPLEUR

NUMERO D'IDENTIFICATION AVEC
AUTOCOLLANTS

J 0

IDENTITE DE LA PERSONNE EXAMINEE

- NOM (majuscule) : .....
- PRENOM USUEL : .....
- DATE DE NAISSANCE : /_/_/_/_/_/_/_/_/
- SEXE : M F (*)

IDENTITE DE LA PERSONNE REMPLISSANT LA FICHE

- NOM (majuscule) : .....
- FONCTION : .....
- SERVICE : .....

DATE ET HEURE D'ETABLISSEMENT DE LA FICHE

- DATE /_/_/_/_/_/_/_/_/ HEURE /_/_/_/_/

ACCIDENT

- DATE DE L'IRRADIATION : /_/_/_/_/_/_/_/_/ HEURE /_/_/ PRESUMEE
- CONDITIONS D'IRRADIATION :
  Durée d'exposition : /_/_/
  . Eventuellement heure : * d'arrivée /_/_/
  * de départ /_/_/

ou était la personne examinée ? .....
.....

Que faisait-elle au moment
de l'accident ? .....
.....

DOSIMETRIE
la personne examinée avait
elle un dosimètre : oui non
Dosimètre (s) récupéré (s) : oui non
si oui numéro (s) /_/_/_/_/_/_/_/_/
/_/_/_/_/_/_/_/_/

PORT D'APPAREIL
RESPIRATOIRE : oui non

CONTAMINATION
VETEMENTS (si détection
faite) : oui non

PREMIERS SYMPTOMES

ETAT DE L'INTERESSE (E) :

			Heure d'apparition	nombre ou durée
Nausées	: oui	non	/_/_/_/_/	_____
Vomissements	: oui	non	/_/_/_/_/	_____
Plaie	: oui	non		
Traumatisme	: oui	non		
Brûlure	: oui	non		

EXAMEN MEDICAL (PARTIE REMPLIE PAR LE MEDECIN)

Nom du médecin (majuscule) : .....
Nom du Malade (majuscule) : .....
Prénom du malade : .....

(*) BARRER LA MENTION INUTILE

. DATE DE L'EXAMEN /_/_/_/_/_/_/_/_/ HEURE /_/_/_/_/
- INTERROGATOIRE MEDICAL

. Asthénie	: oui	non		
. Céphalées	: oui	non		
			Heure d'apparition	
. Nausées	: oui	non	/_/_/_/_/ durée	: .....
. Vomissements	: oui	non	/_/_/_/_/ nombre	: .....
. Diarrhée	: oui	non	/_/_/_/_/ quantité	: .....
. Température	: /_/_/ ' /_/			
. Pouls	:			
. TA	:			
. Conscience	: normale	anormale (préciser)	. agitation	
			. délire	
			. somnolence	
			. coma	

. troubles de l'équilibre : oui non
. Troubles de coordination: oui non
. Examen de peau et
muqueuses : * oedème oui non
* erythème oui non
. Autres (préciser) : .....

MESURES MISES EN OEUVRE

- DESHABILLAGE : oui non
- DECONTAMINATION : oui non
- UTPA : oui non

si oui, voie d'administration : . AEROSOL . BUCCALE . INTRAV.

IODE STABLE : oui non

PRELEVEMENTS

- PRELEVEMENTS DE SANG
  1° prélèvement (si possible avant la 1ère heure qui suit l'irradiation)

* Date : /_/_/_/_/_/_/_/_/ heure /_/_/_/_/
* NFS Plaquettes : oui non
* Cytogénétique (10 cm3) : oui non
* Prélèvement pour Spectrométrie : oui non

2° prélèvement (si possible 2 heures après le premier) heure /_/_/_/_/

* NFS Plaquettes : oui non
- PRELEVEMENTS TYPAGE H.L.A : oui non

- RECUEIL DES URINES : oui non
  (si possible, pour mesure spectrométrie GAMMA)
  S'agit-il de la première miction depuis l'accident ? oui non

CONCLUSION DU MEDECIN

.....
.....
.....

DESTINATION DE LA PERSONNE EXAMINEE

.....

Fig:2

ACCIDENTS AND INCIDENTS OUTSIDE THE NUCLEAR ENERGY SPHERE

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In summary, we show that 10-20 incidents that do not at all involve nuclear power are reported to us each year. The doses are usually trivial, but one or two of the incidents could - with a bit of bad luck - have turned into rather nasty accidents.

All radiation incidents/accidents in Sweden must be reported to the licensing authority (i.e., us). Table 1 lists the reports received 1984-87. 5 of the 47 reports are trivial false alarms.

The remaining 42 incidents should be compared to our 3 500 industrial nuclear gauges etc, 800 radiography devices for non-destructive testing, 700 laboratories using radioactive substances and medical and dental services for 8.3 million people.

Only some three reports concern doses exceeding ICRP limits for workers, and no dose causes real concern. Others handling radiation incidents, such as NAIR in the UK, similarly observe that most events have trivial consequences (1). Nevertheless, some incidents could have turned rather worse.

Thus, both staff and members of the public could have been seriously overexposed in accidents quite similar to the incidents which actually occurred in industrial radiography. Most other industrial incidents are not potentially serious, but rather show that industrial users tend to report even trivial events. However, the case where a source torn from its holder in a mechanical accident wound up in a workman's pocket could have led to substantial damage. The policeman who allowed a person to pass through an X-ray machine for luggage inspection showed surprising lack of judgment, even if the likelihood of serious exposure was remote.

The package handling system at railway stations, where trucks repeatedly run over containers of radioactive substances, could be worth a closer look. Careless handling of radioactive substances led to contamination in a number of laboratories and hospitals, but there is no obvious common denominator.

Electrical faults occur repeatedly in diagnostic X-ray machines, particularly dental ones. Besides overexposure, such faults could cause electrical shock as well as overheating and mechanical damage (X-ray equipment sometimes hurts patients by literally dropping onto them).

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Table 1. Incidents reported to NIRP 1984 - 1987 (15 November)

Event	Source	Consequence
Industrial radiography:		
1985: Source jams in tube due to incorrect mounting	Ir-192 350 GBq	No person dose
1986: Military airfield guard lets 30 visitors into locked hangar during radiography	Ir-192 600 GBq	30 WB doses of max 0.4 mSv each
1986: Technician enters radiography area, believes his alarm dosemeter to be faulty when it sounds	X-ray tube	WB dose 12 mSv
1987: Operator changes films during exposure	X-ray tube	Hand dose 2 mSv
Industrial and similar equipment:		
1984: 3 persons enter field of radiation of level gauge	Co-60 7.4 GBq	3 WB doses of 0.03 mSv each
1985: Lead shield melts when level gauge overheats due to fan failure	Co-60 11 GBq	No person dose
1985: Malfunction of thickness gauge shutter when operator inserts object	X-ray tube	No skin damage detected
1985: Technician enters X-ray room during operation (interlock failure)	X-ray tube	WB dose 0.1 mSv
1985: Operator adjusts X-ray diffractometer cameras while X-ray is on	X-ray tube	No person dose
1985: Operator holds X-ray fluorescence analyzer with shutter failure to palm of hand	Fe-55 + Cd-109 1.7 + 0.19 GBq	Skin dose a few mSv
1986: Customs policeman allows person who claims to have metal object in body to pass through luggage X-ray inspection device	X-ray tube	WB dose 0.01 mSv
1986: 3 persons enter field of radiation of level gauge	Co-60 740 MBq	3 WB doses of 0.01
1986: Sheet metal tears area thickness gauge from holder on production line	Kr-85 9.3 GBq	No person dose
1986: Worker puts area thickness gauge, torn from production line by sheet metal, in pocket	Kr-85 9.3 GBq	Skin dose max 6 Sv, no skin damage detected
1987: 1 person enters field of radiation of level gauge	Co-60 740 MBq	WB dose 0.1 mSv

(table 1, cont'd)

1987: Radar device emits X-rays after rewiring	Electronic vacuum tube	No person dose
Transport, loss, theft:		
1984: Truck crushes dropped package on railway platform	I-125 5 x 3.7 MBq	No person dose, no
1985: Density gauge gone (with burglar 1982?)	Cs-137 110 MBq	?
1985: Sealed source for research work stolen	Ra-229 95 MBq	?
1985: Car with 2 radiography machines stolen	X-ray tubes	?
1985: 3 area thickness gauges gone (properly scrapped?)	Tl-201 3 x 930 MBq	?
1986: Car with static electricity eliminator stolen	Po-210 6.2 GBq	? (Car and source found unharmed)
1987: Truck crushes dropped package on railway platform	Tl-201 74 MBq	No person dose
1987: Truck crushes dropped package on railway platform	Tl-201 74 MBq	No person dose
1987: Truck crushes dropped package on railway platform, 10 MBq released	Cr-51 200 MBq	No person dose, local health physicist decontaminates
Other mishaps with open sources:		
1984: Iodine therapy patient vomits 40 MBq on carpet at home	I-131 400 MBq	60-yr old husband inhales max 2 ALI
1985: Nurse gets spray in face during ventilation scintigraphy and 200 kBq do not wash off	Tc-99m 1 MBq	Skin dose 2 mSv (area 200 cm^2)
1985: Wrong label causes overdose to 8 clinical trial volunteers	Fe-55 240 MBq	8 WB doses of 0.08 instead of 0.01 mSv
1985: 1 person contaminated with 850 Bq after cleaning chemical hood	I-125	WB dose 0.01 mSv
1985: Bottle falls on hospital floor and bursts	Tc-99m 23 GBq	No person dose
1985: Bottle bursts during thawing	I-125 2.7 GBq	No person dose
1986: Dirt in leak detector causes release of 200 GBq	Kr-85 740 GBq	No person dose
1986: Nurse accidentally squirts 65 MBq from syringe	Y-90	No person dose
1986: Faulty seal causes leak of max 3.7 GBq from tritium	H-3	No person dose

(table 1, cont'd)

device at radiation lab		
1986: Researcher gets contamination on skin, all washes off	I-131	No person dose
1987: Researcher gets slightly contaminated when melting metal	Yb-169 190 MBq	No person dose
Medical X-rays and similar:		
1984: Engineer makes faulty connection, gets exposed	Dental X-ray tube	WB dose max 1 mSv
1985: Therapy patient gets overdose due to incomplete notes	Linear accelerator	Patient complains of diarrhoea
1985: Engineer makes faulty connection, exposes himself, pregnant dentist and 2 nurses	Dental X-ray tube	4 WB doses of max 1 mSv
1986: Shortcut causes continuous exposure to dentist and 1 patient	Dental X-ray tube	2 WB doses of 1 mSv + 0.6 Gy skin, 20 mGy salivary and 5 mGy thyroid patient dose
1987: Relay failure causes continuous patient exposure	Fluoroscopy X-ray tube	50 mSv skin dose
1987: Shortcut causes continuous exposure to 1 nurse and 2 patients	Dental X-ray tube	3 WB doses of 1 mSv + 50 mSv to head of 1 patient

False alarms and hoaxes:

1984: An informant to the police claims that a named person has stolen a sealed source and will use it for sabotage purposes

1984: 40 MBq I-131 reported lost during transport are later found in addressee's own store-room

1986: Worker stands close to thickness gauge for a few seconds (dose rate 0.02 mSv/h), fears overexposure

1986: Operator of X-ray fluorescence analyzer complains of erythema (calculated skin dose 0.01 mSv)

1986: Film dosemeter exposed to 60 mSv turns out never to have been worn by anybody

ASSESSMENT OF RADIATION EXPOSURE TO A NON-RADIATION WORKER IN AN INDUSTRIAL RADIOGRAPHY SOURCE TRANSPORT ACCIDENT

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ABSTRACT

A non-radiation worker got exposed to Ir-192 gamma rays from a radiography source pencil in an accident during transport of container with source pencil from one radiography site to another. Dose calculations were made taking into account the activity of the source, the duration and the geometry of exposure conditions. The problem of dose assessment was complicated by the varying distance of source from the body surface, lack of information about exact duration of exposure in different positions etc. Dose estimates were done considering the above factors. Experimental measurements simulating the exposure conditions were made to arrive at dose distributions at different depths in the thigh and to assess the dose to the bone. A thigh phantom of tissue equivalent material and Thermoluminescent dosimeters in the form of $CaSO_4$:Dy powder were employed in the simulation studies. Appropriate correction factors were employed for the energy dependence of the dosimeters for the primary and backscattered gamma rays from Ir-192. The doses to different regions ranged from 500 - 11,500 rads. Dose estimates correlated well with the clinical findings. The sequence of events, experimental and calculation aspects are elaborated in the paper.

DIAGNOSTIC PROCEDURE AND TREATMENT OF A RESIDUAL DADIOACTIVE CONTAMINATION WITH Co-60

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The diagnostic and therapeutic procedure applied to a worker carrier of a residual Co-60 contamination on his left hand for 9 years is described. For several years, this worker has been performing his tasks without exposing to ionizing radiation. When he circumstantially entered into a controlled area, a localized contamination was detected on his hand. According to his professional record, on April 2nd, 1977, while he was performing maintenance tasks on a steam generator, he suffered a cutting wound with the edge of the manway plug.
As the evaluation done in the present installation estimated a high beta dose, a medical examination was decided.

Diagnostic Stage

Physical and clinical studies were done. The former comprised an estimation of the body burden, of the spatial distribution of the burden, and of the resulting dose and dose rate by beta and gamma radiation.
With the aim of decreasing the geometrical error, measurements were carried out by using a INa(Tl) detector 3"x 3" located at 17.5 cm from the hand and centered at the wound.
Calibration was done with a Co-60 point source, measured with the same geometry. The measured activity was 1202.5 Bq which carried to time zero, and assuming only physical decay allowed an estimated burden of 3526.1 Bq. The measured burden after the accident was 3811 Bq.

Estimation of the spatial distribution of the burden

A INa(Tl) detector, 1" in diameter and 3" thick, provided with a lead collimator was used. It had been built on top of a lead semicylinder and it had a rectangular window 0.3 cm width, 1.8 cm long and 2.8 cm thick. Measurements were done over the wound at intervals of 0.5 cm. Fig.1 shows the obtained distribution and the one corresponding to the point source placed 0.5 cm from the collimator.

Estimation of the dose and the beta and gamma dose rate.

The worker had a scar on his palm, aproximately 4 cm long measured from the wrist articulation to the hand's center.
In order to dermine the dose rate in contact, 60 CaF-Fy 200 (3x3x1) mm^3 thermoluminiscent dosemeters were homogeneously distributed on a dosemeter holder sealed with two lucite plates 19x38x2 mm to the hand's palm. Total inmobilization was attained by means of a bandage and through a fingers splint. Besides, dosemeters were distributed on the hand dorsal surface and on the internal surface of the forearm. The detectors remained in the described position for 15.83 hours. The dose rate over the scar

ranged from 0.014 to 0.047 micro Gy/h.
Dose rates on the hand and forearm are below 0.6 nGy/h. These results correspond to an assumed lineal source uniformly distributed around a space 1 cm long and 0.6 cm in depth beneath the wound, that contains an activity of 1202.5 Bq.

Beta decay for Co-60 shows two maximum energies and percentages: E1 max.= 0.314 MeV (99%) and E2 max.= 1.498 MeV (0.1%). Ranges of E1 and E2 on soft tissues are 0.0836 g cm^{-2} respectively. The contributions to the dose of E1 and E2 are considered separately according to the assumed source distribution. The electron range of 0.314 MeV in soft tissue defines an irradiated mass of 25.6 mg. Mean beta dose was 364.24 Gy between February 1977 and February 1985. Dose rate in this mass, calculated for t=0 was 2.8×10^{-7} Gy/h, value that is similar to the natural exposure rate.

The gamma dose on skin and in a mass of tissue next to the assumed source were calculated. The doses over 10 cm^2 of skin over the scar and per annum ranged between 2.3 and 5.7 cGy. Mean calculated dose per annum in a 1 cm^3 volume of soft tissue ranged between 2.3 and 5.7 cGy.

Clinical Studies

A clinical examination and another one localized on the hand were done. Clinical, haematological and cytogenetic manifestations were not observed.
Dermoepidermic trophic alterations were neither observed nor touched. Neither the pletismography nor the vascular centellography showed changes in the hand's vessels. The mean gamma dose on skin, in a volume of 1 cm^3 was below the threshold dose for late non-stochastic effects for a volume of 100 cm^3 (55-70 Gy), even when the latter corresponds to a larger dose rate and a smaller total exposure time.
The beta mean dose is over the threshold dose, whatever the assumed source distribution; whereas the beta mean dose rate is below the upper limit of the dose rate for subletal harm reparation (2.10^{-3} Gy/min) (1). Nevertheless, late non-stochastic effects are to be expected. There is a low probability of stochastic effects due to the low radiosensibility of the irradiated tissue.

Therapeutic Stage

The only treatment was the surgical removal of the contaminating radionuclide. Benefits of the surgical removal were: a) prevention of the spontaneous and eventual translocation of it, b) prevention of the fiber nodules development that might interfere with the free prehension function of the hand.
Risks were: a) the systemic dispersion of the radionuclide, b) the production of sequels that might alterate the prehension function of the hand. Since these risks were not very significant, surgical removal was decided. To prevent systemic dispersion a quelating agent (0.5 g of DTPA-Na3-Ca) was

administered by slow intravenous injection.
According to the physical studies done, the removal of the contaminated tissue was decided. Tissue with 902.8 Bq, other three small portions of tissue with 5.92 Bq, and finally the outer rims of the scar with 119.14 Bq were removed. All tissues were processed with paraffine wax and stained with eosin-haematoxilyn. Histopathologyc examinations of the removed tissue allowed the observation in all the skin sections of: a) slight epidermis engrossment, b) slight dermis fibrosis with unmodified sebaceous and sweat glands, c) absence of arteriolocapillary injuries.
Three days urine samples were required and measured with a detection limit of 0.037 Bq/l with a GeLi detector.
Activity values were below the detection limit and this allowed as to assume that there was not dispersion of Co-60. The remainig burden was 108.78 Bq, while the removed activity was 1132.2 Bq. This confirms the previous estimation of 1202.5 Bq.

Conclusions

The validity of the physical procedure applied was confirmed by : a) the coincidence between the first burden estimation and the one done before surgery that confirmated no transportability of Co-60, b) the physical delimitation of the contamination that enabled the removal of 80% of the deposit, c) the coincidence between the values of the initially measured burden and the values of the remaining and the removed activities altogether.
The slight damage observed could be due to the low dose rate that would allow the reparation processes, and in a lesser degree, repopulation process.
In other cases with higher doses of alpha emissions, development of nodules of doubtful malignant evolution was observed (2,3).

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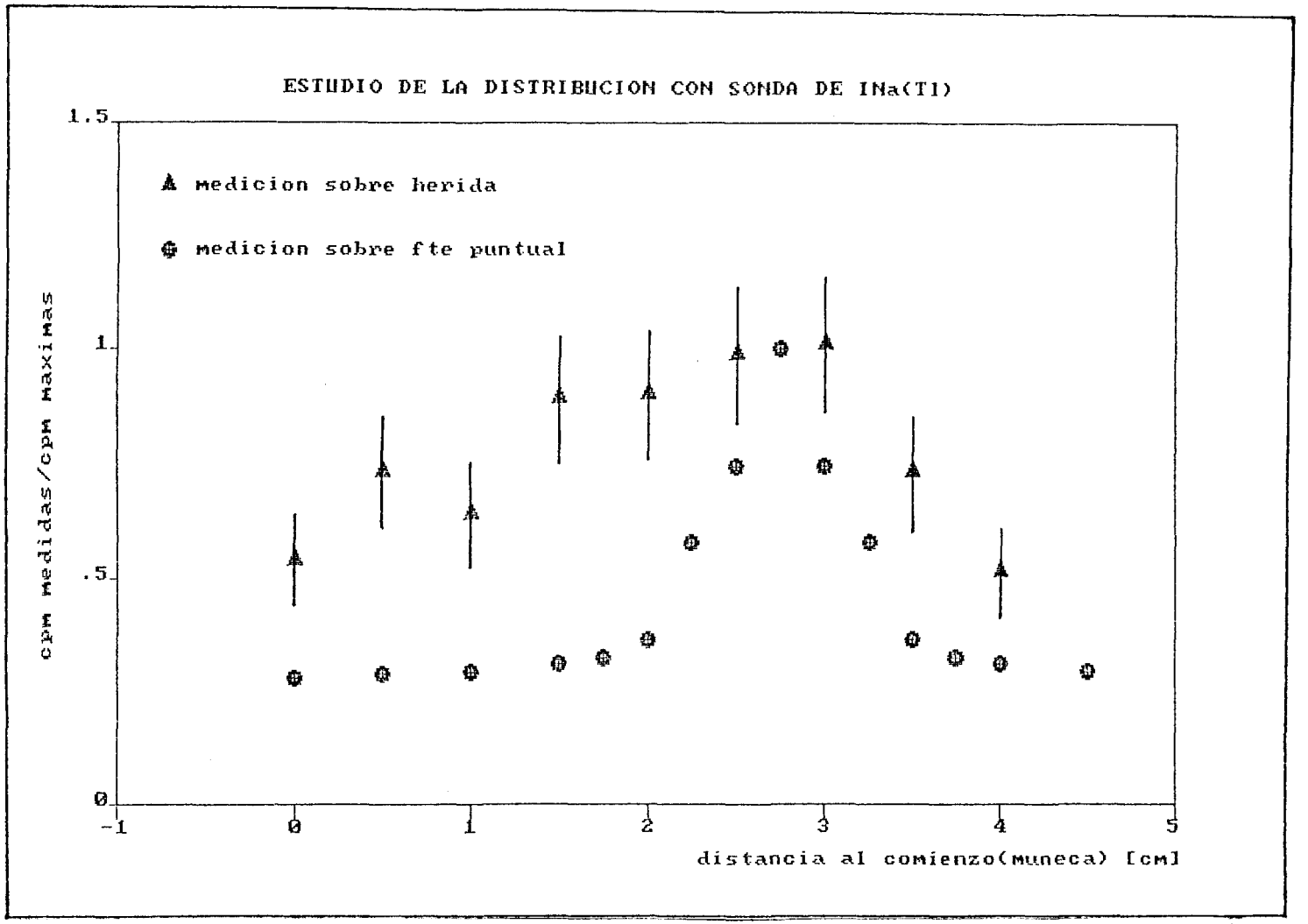
ESTUDIO DE LA DISTRIBUCION CON SONDA DE INa(Tl)
medicion sobre herida
medicion sobre fte puntual
cpm medidas/cpm maximas
1.5
1
.5
0
-1
0
1
2
3
4
5
distancia al comienzo(muneca) [cm]

FIGURA 1

ACCIDENT IN A CRITICAL FACILITY: PHYSICAL DOSIMETRY

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INTRODUCTION

On September 23rd 1983, while a change in core configuration was taking place in an experimental reactor, the RA-2, at the Constituyentes Atomic Center - Buenos Aires - Argentina - an accidental criticality excursion occurred. This accident caused, 48 hours later, the death by overexposure of the operator, and lower doses to the people who were in the commanding room and in the surrounding laboratories.

The absorbed dose to the people involved in the accident and the air absorbed doses in several points outside the facility are shown. An enumeration of the methodologies used for the dosimetric evaluation is done.

DESCRIPTION OF THE FACILITY

The reactor is a critical ensamble of variable configuration with fuel elements of uranium enriched at 90 %; it works using light water as moderator and reflector and it has a 0.1 W nominal power. It is placed into a cylindric aluminium container with a diameter of 2 m and a height of 1.5 m,with a top surface free to reach the core for assembling operations.

The accident occurred just when it was going to be changed the configuration core of the facility. This operation was carried out, as usual, manually by the operator (A), who was leaning on the edge of the container. Figure 1 schematically shows the facility, composed by the reactor, the commanding room and the annexed laboratories to carry out measurements. The figure also shows the location of the 14 persons who were present in the facility when the accident occurred and they are referred to from A to N.

PHYSICAL DOSIMETRY

In order to carry out the dosimetric evaluation it was necessary to use alternative methods because none of the persons involved had their personal dosimeter.

The data were obtained by measuring the induced activity of Na-24 in the body or in blood samples, induced activity of P-32 in hair and activation of personal elements like rings, chains, keys, sweaters, etc., measurements of rate of absorbed dose after the event, and absorbed dose obtained from the personal dosimeters that were hanging on the panel, marked number 3 in fig. 1 and from area dosimeters that are shown in fig. 1.

The methodology used to estimate the absorbed dose, consisted in the use of different evaluation methods based on:

1) using the values of the sodium activation in blood, sulphur in hair and the characteristics of the critical facility (1);
2) knowing the thermic, epithermic and rapid component of the neutrons fluence and applying conversion factors fluence dose in equivalent tissue (2);
3) applying an AERE-R 7487 (3);
4) using the relation between gamma and neutrons absorbed dose (4,5);
5) determining the hardness factor (R) from P-32 measurements in hair and Na-24 in body.

$D = D_{rapid}(1+R)$, considering $R = \frac{D_{thermic} + D_{intermediate}}{D_{rapid}}$

6) measuring the rate of gamma absorbed dose, estimating the components of the gamma dose due to fission products and to prompt radiation, including in this last component, the gamma radiation coming from neutronic captures (6,7);
7) modelling an homogeneous estimated cylindric source based o the core's accidental configuration.
8) defining a transference factor: relation between the gamma absorbed dose measured in a certain place of the facility and in a place of the core.

ESTIMATION OF THE ABSORBED DOSE IN THE PERSONS INVOLVED AND IN THE SURROUNDINGS OF THE FACILITY

In order to evaluate the dose absorbed by the operator (A), all the methodologies mentioned above were applied.

The total absorbed dose in the whole body was obtained adding gamma and neutron components. The weighted neutron fraction in total body was estimated to be 22 Gy, while gamma the fraction in total body was assigned to be equal to the maximum dose in trunk, that is to say, 21 Gy. So, total dose absorbed in whole body was estimated to be 43 Gy.

The dose value obtained belong to the average of the differents methods applied, with a maximun dispersion rate of 30 %

The distribution of the neutron absorbed dose in differen parts of the body are shown in table 1.

The dose in different organs, applying the conversion factors of fluence in free air to organ dose (8) was also estimated. The obtained values are shown in table 2.

The absorbed doses in whole body for the other persons involved are shown in table 3. Errors of measurements are not shown in the tables because they are negligible considering all the other error sources such as: location of each person in the moment of the accident, orientation in the space, time of permanence in each point, etc.

The air absorbed dose in the surroundings of the facility are shown in figure 1.

CONCLUSIONS

From the data of the absorbed dose in the reactor's operator, it was estimated that he was positioned at the time of the critical excursion was leaning on the reactor container with his right arm extended over him, and with his body and head a little turned to his left.

From the measurements of the rate of the gamma absorbed dose taken after the event and from the estimations carried out, we conclude that there is a contribution to the absorbed dose of the persons B and C due to sky effect.

There is a good correlation between the dose obtained by the physic dosimetry and the dose estimated by clinic and biological dosimetry (9).

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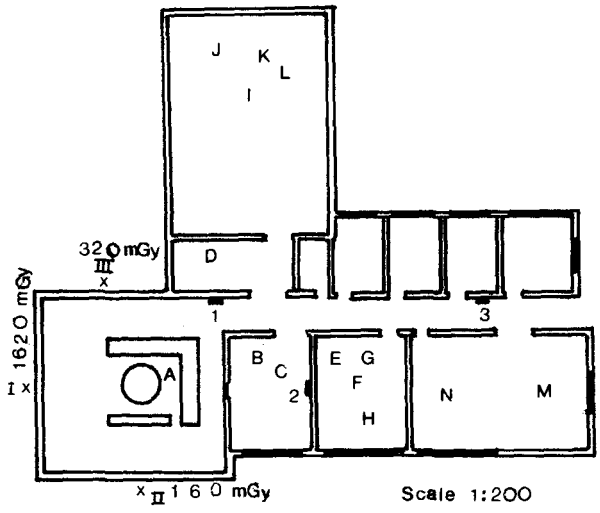

Figure 1
RA-2 Facility

A-N: Location of the persons.

1-2: Area dosimeters.

3 : Panel of personal dosimeters.

I-III: Air absorbed dose.

Table 1: Neutron Absorbed Dose Distribution in body surface

Area	Neutron absorbed dose (Gy)
mustache	22
head,right side	23
head, left side	18
nape	5
trunk	26
right axilla	16
left axilla	8
pubes	8

Table 2: Organ Neutron Absorbed Dose

Organ	Neutron absorbed dose (Gy)
right lung	20
left lung	6
red bone marrow	8
testes	4

Table 3: Total absorbed dose

person	Gamma component (mGy)	Neutron component (mGy)	Total (mGy)
B	130	147	277
C	124	124	248
D	96	83	179
E	100	96	196
F	82	63	145
G	90	81	171
H	90	81	171
I	5	6	11
J	3	3	6
K	3	3	6
M	1	1	2
N	1	1	2

INCIDENT INVOLVING LOSS OF A 37 GBq AMERICIUM 241 SOURCE FROM A RADIATION GAUGE

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This paper describes an incident reported to the Radiation Safety Section on 31 July 1986 wherein a radioactive source was found to be missing from the source housing of a radiation gauge used at a steel mill in Hastings, Victoria.

Description of Gauge and Process

The source was Americium 241 of activity 37 GBq, and was incorporated in a Reuter-Stokes Safety-Ray Type RSS-811 gauge (see Fig.1). It was used in one of six such gauges to detect the presence of steel slabs being rolled into sheet steel in the hot strip mill. Three were operational and three were installed as back-up devices which could be switched in should a fault develop during a rolling period. The source housings were made from stainless steel and were mounted in a comparatively inaccessible position between rollers under the hot strip mill. The radiation beam was directed upwards at an angle (see Fig.2). Below the source housings was a drop of approximately 6 metres to the flume system used to wash iron oxide scale from the mill area to collection pits.

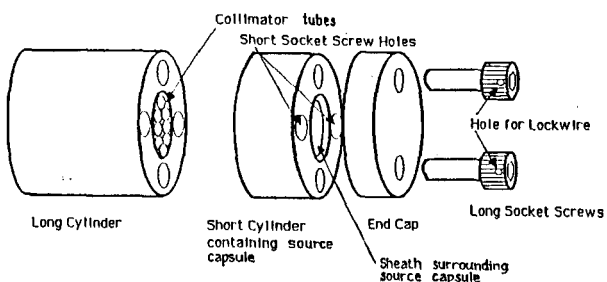

Fig.1 Diagram of Source Housing

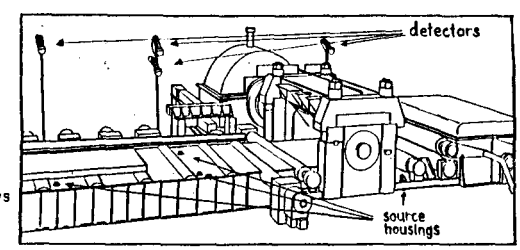

Fig.2 Typical Hot Strip Mill

Sequence of Events

A company operation check of the gauges on 14 July 1986 had shown everything normal prior to a 7 day steel rolling period. On 23 July 1986, a further check prior to the next 7 day rolling period showed that the radiation level in the beam was 8 uGy/h compared with an expected level of 100 uGy/h. As a result the back-up gauge was switched in, the rolling period commenced, and the faulty gauge was listed for maintenance at the next down period on the mill. On 30 July 1986, during the scheduled down period, an examination of the gauge showed that the housing had become disassembled and that the radioactive source and parts of the gauge assembly were missing. A search was commenced by company personnel and one long screw and one short screw were recovered from the flume near the entrance to the finishing mill scale pit.

The following morning, 31 July 1986, the incident was reported to the Radiation Safety Section (RSS) and a full search of the flume area and finishing mill pit was commenced. No further parts of the gauge were recovered from the flume or the finishing mill pit. The search continued in the 200-300 tonnes of scale which had been removed from the pit. The scale was routinely sold to a cement manufacturer with plants at Geelong, Victoria and Traralgon, Victoria for use as a fluxing agent in the manufacturing process. It was advised that approximately 640 tonnes had been transported to Geelong and 40 tonnes to Traralgon since 14 July 1986, when the gauge was last known to be intact. The company was advised that the source may have been transported to one of their factories in the scale, and that they should not process any further scale from Hastings. They were also advised to commence a search of the unprocessed scale and to obtain suitable samples from various stages of the manufacturing process for analysis.

On 1 August 1986, a large industrial earth sifting machine was used to continue the search of the scale at the steel mill and the following parts of the gauge assembly were recovered : the second long screw, the second short screw, the short cylinder, the source sheath, and the collimator tubes. Examination of the scale at Traralgon found no trace of the gauge. RSS staff commenced a search of the scale at the cement works in Geelong. This search concentrated on hoppers which were mixing iron oxide scale with limestone. Advice from the company was that approximately 80% of the scale received since 14 July 1986 had already been processed. A press conference was held by the Health Department giving details of the incident and warning the public of the lost source.

The search of scale processed through the hoppers and conveyor systems at the Geelong plant continued until clinker cement samples examined by RSS staff on 4 August 1986, using a portable multi-channel analyser, showed traces of Am 241. These samples had been collected on a routine basis for chemical analysis. Samples from the relevant dates were forwarded to the Australian Radiation Laboratory (ARL) for assessment. No Am 241 was detected in the sample dated 25 July 1986. A maximum activity of 3-4 Bq/g was found in the sample from 29 July 1986, and lesser activities were detected on subsequent dates. Calculations indicated that, assuming an even distribution of Am 241 in the average daily output of 2500 tonnes of clinker, most of the 37 GBq of Am 241 would be accounted for. The specific activity of Am 241 in the clinker would have been further reduced as it was added to the clinker stockpile of approximately 30,000 tonnes. The cement company advised that some of this material could have been processed into cement and sold either in bulk or in bags prior to any indication of the incident.

The presence of Am 241 in the clinker indicated that the source capsule must have been processed through one of the two ball mills at the plant, and consequently that the remains of the source capsule could be still in a ball mill or in the reject material from the ball mills. Reject material from the ball mills had been used on roads in quarries and farms adjacent to the plant and on

the driveways of a number of employees. Searches of these areas were carried out but no trace of Am 241 was found.

On 5 August 1986, a meeting between the Health Department, Radiation Advisory Committee, ARL, the steel company, and the cement company confirmed that the concentration of Am 241 in the cement did not represent a hazard and a press conference was later held to report the finding of traces of Am 241 in the cement.

It then appeared most likely that the source capsule remains were still in one of the ball mills. Access to the mills was not immediately available. A motor failure prevented unloading of the small mill, and the large mill was therefore necessary for continued operation of the plant. The large mill contained about 2.3 million chromium steel balls ranging from about 2 cm diameter up to 9 cm diameter whilst the small mill contained about 1.5 million similar steel balls.

On 6 August 1986, a meeting was held with the cement company's employees to discuss their concerns regarding the incident. On the same day, a shutdown of the large ball mill was arranged to enable a search of the interior. This proved inconclusive and indicated that the mills would need to be emptied to enable an effective search for the source capsule remains.

On 12 August 1986, the small ball mill was emptied and traces of Am 241 were detected in the slurry, indicating that the source capsule had been processed in this mill. Samples from the slurry were later assessed by ARL and found to contain 22 Bq/g. Thirteen officers of the Health Department searched through the steel balls and the source capsule was recovered the following day. An examination of the capsule by ARL showed that about 1.5% of the original source activity remained, i.e. about 56 MBq. This confirmed that the majority of the activity had been processed into the cement. However, in view of the low concentration of Am 241 in the cement product, it was not considered that any action regarding decontamination of the plant or cement product was warranted.

Review of Incident

As a result of the incident a review was made of the design of the gauges and the reasons for the occurrence. The parts of the source assembly recovered were sent to the Aeronautical Research Laboratories for metallurgical examination to determine whether the gauge had been deliberately interfered with. Their report indicated no signs of interference.

The gauges on the hot strip mill are subject to wide ranges of temperature (the steel slabs on the mill are at 1100°C) and considerable vibration. It was considered that this, along with inadequate maintenance procedures and the design of the gauges, led to the screws loosening over a period of time and the source assembly eventually falling apart.

The design of the gauge was considered inadequate in that the source capsule was held in the source assembly only by the two long screws through the end cap. Whilst these had also been held by a lockwire with a lead seal, this wire had rusted through and disintegrated. It was considered that the design of the device should have been better considering the extremes of temperature and vibration and the configuration of the gauge in the hot strip mill.

Since the incident, the following conditions have been imposed on all RSS-811 gauges registered with the Health Department under the Health (Radiation Safety) Regulations :

a) Each gauge to be removed from installation and inspected in detail in the workshop, with the following work to be carried out:
   i) All source assembly screws examined and tightened to a torque of 8 Nt.m. Locktite 622 or equivalent to be used to ensure that they do not loosen. (It was noted that all screws were able to be tightened by at least a half turn, even a unit which had not been used but was as supplied by Reuter-Stokes).
   ii) Rewire screw heads with MP35N or equivalent wire and seal.
   iii) Inscribe serial numbers into body of assembly.
   iv) Replace scews on beam window with stainless steel screws and MP35N wire or equivalent.

   Source assembly then to be bolted to frame mounting at correct torque setting; bolts to be wired with MP35N wire or equivalent.
b) Prior to each start-up of a rolling process, a physical check of frame and source assembly bolts to be carried out.
c) For each sensor or out of calibration alarm, a physical check to be made to ensure that the source assembly is intact (to be logged).
d) Torque on frame mounting bolts to be checked every 3 months.
e) Torque of source assembly screws to be checked, and screws rewired, every 12 months.
f) Mounting and removal access to the source assembly to be made simpler to allow for ease of removal for thorough inspection.
g) A maintenance-free source holder to be designed and plans to be forwarded to the Health Department for approval; the design to allow source insertion from the top of the assembly and to negate the need for any assembly screws in the lower section.

Other State Health authorities were warned of the incident in order that they could determine whether similar gauges existed in their States and so that appropriate inspections could be carried out.

It was considered that the design of a maintenance-free source housing and the interim conditions imposed on the registrations of similar gauges would prevent any repetition of this incident.

Reference

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SURGICAL TREATMENT OF PLUTONIUM-CONTAMINATED WOUNDS

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ABSTRACT

This Paper reviewed the surgical treatment of plutonium-contaminatcd wounds through 7 cases from 1972 to 1985. All the cases were men. The causes of contamination were mainly carelessness in work. The classification of wound was laceration (2 cases), burn (2 cases) and puncture (3 cases). Original activity of plutonium deposited in wounds were estimated to be from 40.7Bq to 1.739 10^4kBq. After first aids they were performed by early surgical wound excision and DTPA administration.

Only two cases of them, due to the more plutonium still remaining in wound, were admitted to our hospital. We repeatedly monitored the amount of plutonium in wounds and local lymphonodes and also determined the range of contamination. The level of activity was 0.3145kBq, 7.03kBq and 0.555kBq, 1.0915kBq respectively. Under the block anesthesia, according to radio-active operation regulation, wounds and local lymphonodes were excised and to one of them DTPA was injected systemically and locally. Then it was satisfactory that the level of plutonium remained in wounds decreased from 111Bq to 18.5Bq. The committed dose equivalent was 0.885Sv.

The fact as above shows that it is important to pay more attention to plutonium deposited in regional lymphonodes for puo_2-contaminated wounds. Lymphonodectomy were performed for two cases with plutonium in local lymphonodes. We have followed them up to 2-14 years and no unsatisfactoriness was found. It also demonstrated that DTPA local injection not only was effectively to decrease the level of plutonium in wound but also might prevent the increase of the body burden of plutonium. So, we suggested that it should elect the method of anesthesia and abide by regulation in operation and the locally supplying chelating agents was suitable for the cases whose residual amount of plutonium in wound up to a certain level and who was not intended to perform any other surgical procedure.

SPACE AGE RADIATION PROTECTION

by

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ABSTRACT

As we enter the 21st Century, human beings will permanently occupy outer space, including operational space stations in low Earth orbit, work stations at various orbiting facilities throughout cislunar space (e.g. geostationary orbit), and initial bases on the surface of the Moon. Although small in overall number, mankind's permanent extraterrestrial population will be potentially exposed to a very high radiation risk, both from natural (cosmic) radiation sources as well as manmade radiation sources such as space nuclear power plants. This paper describes the projected radiation exposure environments that will be faced by astronauts in a variety of space work places, including extended human expeditions to the planet Mars. Advanced solid state radiation dosimetry devices necessary to provide adequate, real time and long term, radiation protection are described, including test radiation instruments used on the Space Shuttle. The need for advanced radiation protection programs for extended space missions is also discussed. Contemporary U.S. astronaut radiation exposure limits are compared to contemporary "terrestrial" exposure guidelines. Space radiation protection is a critical issue, if we are to permanently inhabit cislunar space and become a space-faring species.

INTERCOMPARISON OF RADON MEASUREMENT TECHNIQUES

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INTRODUCTION

Direct and continuous measurement of radon is of importance e.g. when estimating the effect of various countermeasures against high radon concentrations and locating the possible radon leakage paths. The need for continuous measurements has increased in Finland after the National Board of Health set the recommended limits (an action level of 800 Bq/m^3 and a planning value of 200 Bq/m^3 for new buildings). According to the latest statistics (2) the 800 Bq/m^3 limit is exceeded in approximately 1.4 % of the dwellings. Up till now, few devices for continuous measurement have been available, and are based on different measuring principles. Different radon and radon daughter measuring techniques have been reviewed by Budnitz (1). However, little attention was given to the continuous measurement. The national laboratories in the European Community carry out intercomparison measurements, but these have been limited for the purpose of quality assurance only. Recently, Shimo et al. (15) presented an intercomparison test. With the two continuous instruments included, they obtained comparable results when using indoor and outdoor air as sampling gas.

The object of the present study is to test continuous radon meters available in Finland (representing different measuring principles) to find out their reliability in different circumstances and their suitability for different tasks. Here, we report the results of laboratory measurements showing the effect of humidity and aerosol concentration on the instruments. Preliminary results of the field tests are also included.

METHODS AND INSTRUMENTS UNDER COMPARISON

The techniques and instruments studied were: a scintillation chamber (SC), a continuous flow ionization chamber (IC), an electrostatic monitor (EM), and a new commercial method based on ion concentration measurement (IM).

Scintillation Chamber (SC): Perhaps the most common method in continuous and sample radon measurement is the Scintillation or Lucas chamber (3,12,13,16). We used a commercial ZnS(Ag) device with a build-in pump. The instrument is fabricated by Pylon Electronic Development Co Ltd, Canada. A continuous 0.4 lpm air flow was maintained through the scintillation cylinder and alpha particles were counted for appropriate time periods. The response time of the instrument is approximately 3 hours because of the daughters attached into the scintillator. The detection level is set by statistical limitations: 40 Bq/m^3 is measured with an accuracy of about 20 % using a 30 min counting interval.

Ionization Chamber (IC) is another one of the traditional widely used methods (5,9,10,11,15) We used two continuous flow (8 lpm, chamber volume 20 l) instruments constructed in the Physics laboratory of Tampere University of Technology. The analoc correction circuit presented by Janka and Lehtimäki (10) gives the instrument a fast response time of 5 min. The sensitivity of the instrument is about 20 Bq/m^3.

Electrostatic Monitor (EM) suitable for continuous measurement was presented by Dalu and Dalu (4) and has since been modified (14,17). We used an instrument made

by Studsvik Energiteknik (Sweden) that is to be commercially distributed by Alnor Co. The instrument is based on radon diffusion through a plastic foam into the measurement chamber where the decay products are collected on the surface of a surface-barrier detector by an electric field. Capable of resolving the energies of RaA and RaC', the instrument has potential for fast response. This is somewhat limited by the diffusion process through the foam: the practical response time is over 30 minutes. A built-in pump is used for flushing the chamber but not used during the counting. By using an external pump and a high efficiency filter it is possible to operate the instrument in a continuous flow mode. This mode was also studied. Statistical limitations set the 20 % accuracy concentration at appr. 100 Bq/m^3 using 30 min intervals and RaA counting.

Ion Concentration Measurement (IM): A new commercial instrument, Ilma-Radon, made by Ilmasti Co, Finland is basically an ion meter. The use of indoor small ion concentration as a measure of the radon concentration is based on the fact that radon with the short lived decay products is the major contributor to the ion formation rate indoors. The main advantage of this type of measurement is the fast response, because the life time of small ions is short and practically no daughters are collected. The ion method has, however, inherent causes of inaccuracy. The indoor ion concentration is dependent on the processes which remove small ions: recombination, attachment to surfaces and aerosol particles as well as electric fields (7,8).

EXPERIMENTAL AND RESULTS

The laboratory tests were done in a 26 m^3 test chamber where different concentrations of radon and other airborne substances could be generated. ^{222}Rn was allowed to diffuse into the test chamber from a 1 mCi $RaCl_2$ salt dilution through an injection needle and a dilution volume. The chamber contained filters to clean the chamber air and a humidifier. Also compressed (and filtered) air supply and an exhaust blower were available.

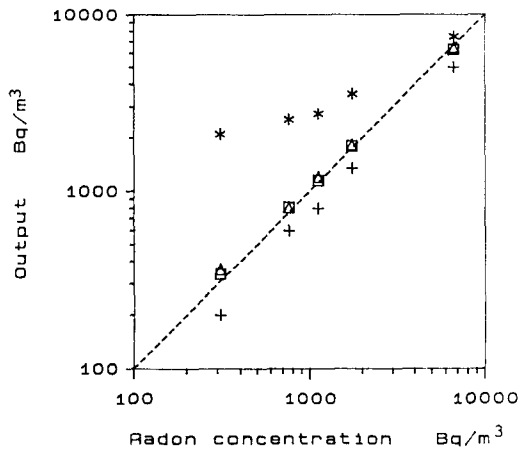

Figure 1. The output of the devices as a function of Rn concentration. △ =SC, □ =IC, + =EM, ⋆ =IM.

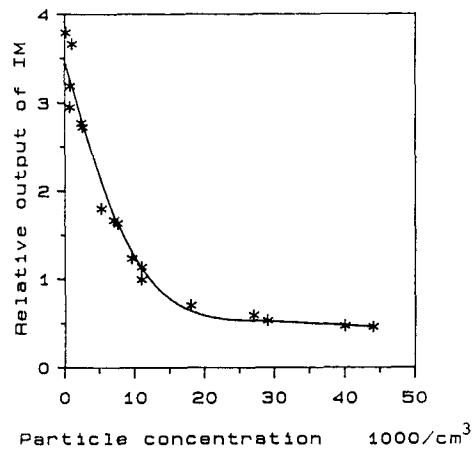

Figure 2. The effect of the aerosol concentration on the IM output.

Calibration: The instruments were calibrated in the facilities of the Finnish Centre for Radiation and Nuclear Safety. Monitors were placed in a 8 m^3 calibration room, where radon was produced by uranium ore. Radon concentration was continuously registered by two Lucas-type detectors, which had been calibrated against standard radon sources. The temperature and the relative humidity remained reasonably constant with values of 25 °C

and 30 % , respectively. The results at five different radon levels are shown in Figure 1: the IC and SC outputs were in good agreement with the radon concentration over the measured range; the EM calibration was too low but linear. The large deviation of the IM output is mainly due to the low aerosol concentration (approximately 250 $1/cm^3$).

Aerosol Concentration: The effect of aerosol concentration on the indicated radon concentration was studied using tobacco smoke. One of the ionization chamber monitors equipped with a HEPA filter served as a concentration reference. The aerosol size distributions and concentrations were measured using a TSI Differential Mobility Particle Sizer. IM was found to be the only instrument affected by the aerosol concentration. Figure 2 shows the relative output of IM as a function of the total number concentration of particles larger than 0.01 μm: the output is approximately inversionally proportional to the aerosol concentration, as would be expected.

Humidity: In order to achieve a low humidity level, compressed air was fed to the chamber. During the test, the humidity was increased stepwise. One of the IC:s was again used as a reference meter taking the sample through a dryer. The temperature in the chamber varied less than 1 °C during the experiment. The relative output of the instruments as a function of R.H. is presented in Figure 3. The most sensitive is EM due to the change in the neutralization of RaA. This has been found to be dependent also on the concentration of various trace gases in the air (e.g. 6). Due to active charcoal filtered air feed, the concentration of these gases in the chamber air was propably lower than in normal indoor air. It is possible that the change in the output is to some extent caused by infiltration of gaseous compounds from the surrounding laboratory during the experiment. The ionization chamber shows a very weak dependency of the same kind. The output of the Scintillation Chamber remained practically unaffected by the changes in humidity. The main reason for the scatter in IM results is probably the changes in the particle size distribution and concentration.

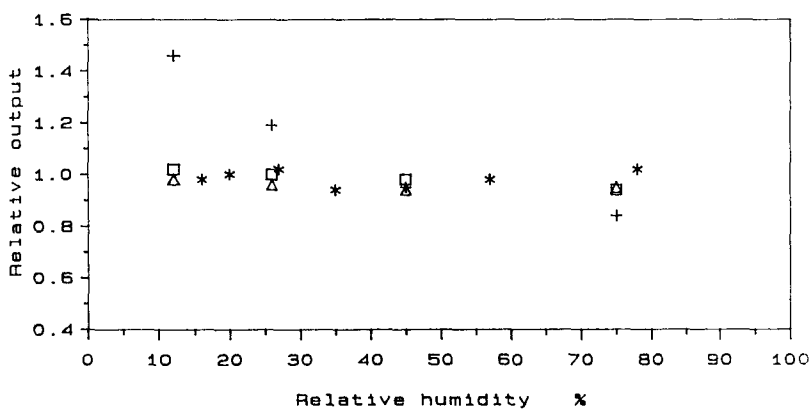

Figure 3. The effect of humidity on the relative output of the instruments. Temperature 25...26 °C. △ =SC, □ =IC, + =EM, ⋆ =IM.

Field Tests: Due to the observed variations of the IM response, the meter was further tested in eleven separate dwellings — ranging from one-storey small houses to four-storey buildings. The results, shown in figure 4, include 4-10 readings in every dwelling. The momentary radon concentration of measurement points were determined by samples of evacuated scintillation chambers. The IM device tended to give high results especially at low radon concentration. Both aerosol generating activities (cooking) and movements near the device were observed to affect the reading.

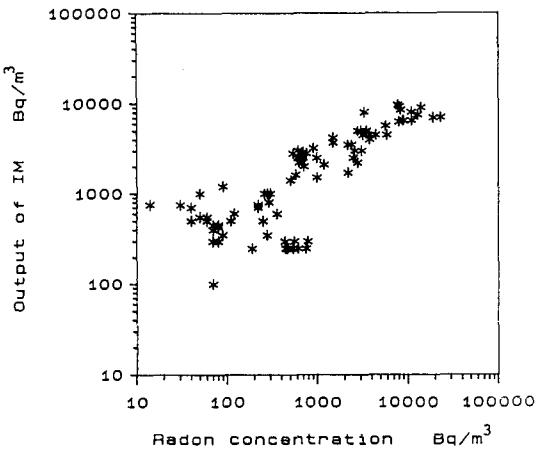

Figure 4. The output of IM as a function of radon concentration. Short time averages in 11 houses.

CONCLUSIONS

In continuous measurement the response of the instrument must be fast enough to be able to follow the concentration variations of interest. IC and IM have response times short enough for practically any application. The use of an external pump and filter gives EM a reasonably fast response, whereas in diffusion mode the response is rather slow. SC is too slow for radon leakage and certain ventilation studies. The most reliable instruments seem to be SC and IC. The IC is somewhat sensitive to changes in the background gamma level, which is not a serious drawback in dwellings. EM is sensitive to the humidity (and the gaseous composition of the air). The dependency of IM output on the aerosol concentration and on electric fields causes the readings to be unreliable, especially in short time measurements. The manufacturer is working on a method to compensate the effect of aerosol particles. The effectivity of this method remains to be studied.

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PACIFIC REGION RADON DAUGHTER INTERCOMPARISON

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ABSTRACT

The Nuclear Energy Agency of the OECD, through its Committee on Radiation Protection and Public Health, has established an international programme for the intercomparison of measurements of radon, thoron and their daughter products. In April 1987 an intercomparison of radon daughter measurements was carried by 9 laboratories from Australia and Japan, using the radon test facility at the Australian Radiation Laboratory (ARL), Melbourne. Measurements of the potential alpha energy concentration (PAEC) between seven of the laboratories showed that 3 were not significantly different from ARL and that 2 were within 5%. Some of the differences can be assigned to systematic differences in activity and flow calibration standards used by each laboratory.

INTRODUCTION

A programme for the intercomparison and intercalibration of measurements of radon, thoron and daughter products, sponsored jointly by the Nuclear Energy Agency (NEA) of the Organisation for Economic Cooperation and Development (OECD) and the Commission of European Communities (CEC), was initiated in November 1983. The first part of this programme, covering the intercalibration and intercomparison of radon measuring equipment was successfully completed in 1986 (1). The intercomparison of measurements of the shorter-lived radon daughters cannot be made in the same manner as radon, using an interchange of sample containers, but requires side-by-side measurements of the same test atmosphere (2,3).

The Australian Radiation Laboratory is the designated Pacific region radon reference facility for this intercomparison programme and has the responsibilty of coordinating intercomparison activities between laboratories of OECD-member countries within the Pacific region. Over the week of March 30 to April 3, 1987, representatives of six Australian and two Japanese laboratories attended ARL in Melbourne to carry out an intercomparison of radon daughter measurements using the ARL radon test chamber.

The measurement programme was divided into four sections,
(a) Intercomparison of flow and activity standards
(b) Intercomparison of radon daughters from grab sampling

(c) Intercomparison of radon daughters from integrating monitors
(d) Intercomparison of unattached radon daughter measurements

A total of 35 grab sample measurements and three days of integrating monitor measurements were carried out by the participating laboratories . This paper briefly descibes the methodology for this series of intercomparisons and reports the some of the results. A more detailed decription of the measurements and results is provided elsewhere (4).

PROCEDURE

All measurements of radon daughter parameters were made using the ARL radon test chamber. The test atmospheres for these measurements were defined at an OECD/NEA and CEC International Radon Workshop, held in Paris during May 1985. Two sets of experimental conditions were produced, with different aerosol concentrations. The chamber was maintained at 20° C, 50% RH and a nominal radon concentration of 4000 Bq/m^3. A small number of measurements were made on the final day at a radon concentration of 650 Bq/m^3. A Carnauba wax aerosol with an activity median aerodynamic diameter of 0.1 micron was generated using a condensation aerosol generator. With aerosol concentrations of 5000 and 30000 per cm^3, the resultant radon daughter equilibrium ratios were 0.3 and 0.6, respectively.

Five of the laboratories provided calibrated alpha activity standards for intercomparison. Where possible, each of these sources was counted in each of the detection systems, and the resultant counting efficencies compared. Dry-test meters were provided by two laboratories for the measurement of sample volume. These meters were intercompared with the ARL wet-test meter. The rotameter flow-meters used by two of the laboratories were also intercompared with the ARL wet-test meter. Seven of the participating laboratories brought their equipment for the sampling and the counting of radon daughters, while two laboratories provided integrating radon daughter monitors for intercomparison.

Four sample ports were available on the radon test chamber. The radon daughter intercomparison was carried out with three measurement groups, with ARL taking part with all three groups. Each group collected concurrent samples from the chamber and the activity on the filter samples was analysed to determine the individual daughter concentrations and the PAEC, using either two-count alpha spectroscopy or three gross alpha count methods. The integrating monitors were operated for three sets of eight hour periods in the radon test chamber, and each exposure period was analysed for integrated PAEC. The results for these monitors are reported elsewhere (4).

RESULTS

The results of the intercomparison of the five calibrated alpha sources are summarised in Table 1. This table shows the derived counter efficencies for each reference source. Three sources were in good agreement (A, C and E), one source gave efficencies approximately 6% higher (B) and one source gave efficencies 24% lower (D). The measured ratios of the flow-rates, relative to the ARL wet-test meter, were 1.03, 1.02 and 1.00 for the flow-meters used by laboratories B,E and H, respectively.

Table 1. Intercomparison of calibrated alpha sources-detection efficency (cps/Bq) for laboratory counting systems

	Laboratory Reference Source				
Counter	(A)	(B)	(C)	(D)	(E)
(A)	.439±.003	.469±.002	.446±.004	.355±.001	.443±.002
(B)	.389±.002	.418±.002	(No measurement)		.401±.002
(C)	.157±.001	.161±.001	.158±.001	.146±.001	.161±.001
(D)	.350±.002	.371±.002	.310±.002	.267±.002	.357±.002
(E)	.127±.001	.138±.001	(No measurement)		.132±.001

Each of the three measurement groups carried out at least nine intercomparisons of radon daughter concentration. The derived values for the PAEC, relative to the respective ARL measured value, are summarised in Table 2, together with the calculated mean, standard deviation and standard error of the mean, for each laboratory.

DISCUSSION

In general, there was excellent agreement between most of the laboratories for the measurement of radon daughter PAEC. The results in Table 2, for the mean ratios of laboratory PAEC to ARL PAEC, show that the values for three of the laboratories (F, G and H) were not significantly different from 1.00. The ratio for laboratory E of 1.02 was different from 1.0 at the 95% confidence level. Laboratories B,C and D , with ratios 0.87, 0.95 and 1.23 respectively, were different from 1.00 at greater than 99% confidence. Some of these difference can be accounted for by the differences in the reference source calibration, evident in Table 1. Examination of the standard deviation of the mean value for each laboratory, in Table 2, shows that some laboratories were more consistent in their measurement procedures.

Table 2. Results of PAEC intercomparison - ratio of PAEC for participant laboratory to corresponding ARL value.

Sample No.	Laboratory Identification: Group 1 (B)	Group 1 (F)	Group 1 (H)	Group 2 (C)	Group 2 (E)	Group 3 (D)	Group 3 (G)
1	0.81	0.92	0.99	0.98	1.07	1.55	0.90
2	0.81	0.96	1.02	0.97	1.03	1.27	0.80
3	0.85	0.98	1.11	0.97	1.05	1.37	0.96
4	1.04	1.11	1.07	0.97	1.01	1.16	1.03
5	0.75	1.11	1.04	0.96	0.97	0.94	----
6	0.91	1.12	1.01	0.95	1.06	1.47	0.97
7	0.90	0.94	1.04	0.96	0.98	1.09	1.37
8	0.88	1.08	0.81	0.94	0.98	0.96	0.95
9	0.82	1.12	1.07	0.88	1.04	1.22	----
Mean	0.87	1.04	1.02	0.95	1.02	1.23	1.00
1 SD	0.09	0.09	0.08	0.03	0.04	0.21	0.18
SEOM	0.03	0.03	0.03	0.01	0.01	0.07	0.07

ACKNOWLEDGEMENTS

The contribution and participation of the following researchers and laboratories is gratefully acknowledged in the successful execution of this stage of the intercomparison programme; Professor Ryuhei Kurosawa, Waseda University, Dr. Keizo Yamasaki, Kyoto University, Dr. Stewart Whittlestone , Australian Nuclear Science and Technology Organisation, Dr. Philip Crouch, South Australian Health Commission, Dr. Frank Cattell, Macquarie University, Mr. Vic Leach, Queensland Department of Health, Mr. David Bromwich, Northern Territory Department of Mines and Energy and Mr. David Woods, Office of the Supervising Scientist.

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IONIZATION CHAMBER RADON MONITOR WITH PULSE COUNTING MODE

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INTRODUCTION

There is a special need for continuous radon meters in studying radon entry and possible countermeasures against high radon concentrations in existing houses. In these studies, a wide concentration range and a fast response are required, as is a movable sampling probe for active sampling. In addition, the meters have to be portable to be suited for field use.

Traditional methods for radon measurement are the two-filter method (9), the electrostatic method (1), the scintillation chamber (5) and the ionization chamber (2). All of these methods can be applied to continuous measurement and relatively fast responses can be achieved using alpha spectroscopy or correction circuitry/computing (3,6,8). The sensitivity of these instruments is in general proportional to the chamber volume. A large measurement chamber, however, is not desirable from the viewpoint of portability. The volume is also limited by demands set by daughter collection and alpha detection in the electrostatic method and the scintillation chamber method, respectively. In the ionization chamber the benefit of using a larger chamber is decreased by the increase in the background current. The background affects also the other methods at a low concentration. A problem common for all active sampling devices is the variation in the diffusion coefficient and the neutralization rate of radon decay products (specially ^{218}Po) caused by changes in the gaseous composition and the humidity of the air. This causes variation in the attachment of the daughters in the measuring chamber and hence affects the instrument output.

To meet the conditions and to solve the problems described we have designed a smaller ionization chamber used both in alpha pulse mode and total current mode. Using an ionization chamber in pulse mode with normal air introduces difficulties: the long and low pulses are hard to process and mechanical vibrations are ready to cause spurious pulses in the frequency range of the pulses. The pulse length causes the coincidence error to become significant at a relatively low pulse rate. Recently, Katase et al. (4) presented a new instrument based on the use of plane multiwire-electrode ionization chamber used in pulse counting mode. A good sensitivity and a fast response time was reported. Shimo et al. (7) reported the performance of an ionization chamber meter used in both pulse and total current mode. Unfortunately, the construction of their instrument is not known by the authors.

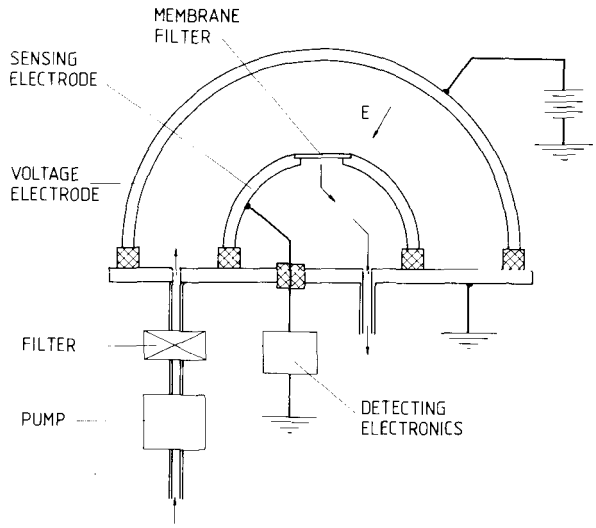

Figure 1. Schematic diagram of the instrument.

CONSTRUCTION

A schematic diagram of the present instrument is shown in figure 1. The hemispherical ionization chamber has a volume of 1.7 litres. The volume is selected as a compromise between portability and sensitivity — the chamber can easily be replaced with a larger one. To achieve an adequate pulse rise time, a 2 kV voltage is used between the electrodes. A switching-generator-module is used with an additional RC-network for reducing the voltage ripple. The edge deformation of the electric field is minimized by a resistive voltage division chain on the supporting plane between the electrodes.

To eliminate the problem caused by the variation in the attachment of the daughters in the chamber, a membrane filter is installed at the air outlet (on the inner electrode) to ensure 100 % attachment of the decay products. The electrodes are constructed using plastic hemispheres coated with highly conductive ($< 3\Omega/\square$) paint in order to minimize the microfonic noise. The outer surface of the outer electrode is painted for electric shielding. The electrode-preamplifier unit is rubber band mounted inside a rigid metal box and connected to the pump unit with a flexible tubing.

The charge sensitive electrometer is constructed using a modern operational amplifier fulfilling the conditions for low noise and high input impedance. The output integrates the input signal without damping, and the electrometer is zeroed externally before saturation. The output signal of the electrometer is further processed and digitalized to satisfy the needs of the pulse counting and the total current mode measurements.

DISCUSSION

Figure 2 shows results of a calibration measurement with different radon concentrations. This preliminary measurement has been performed using a larger prototype chamber. The pulse mode gives more reliable results when the radon concentration is low.

The signal processing in current mode is straightforward, whereas the operation in pulse mode still causes problems. At the moment the signal processing is performed using an external microcomputer. The program has still to be developed to be suited for a microprocessor based system to be built in the instrument. This system is also to deside whether pulse mode is used or the total current measured, as well as the length of the counting period. Alternatively, these can be selected manually.

Although at a prototype stage, the processing system has shown promising features: The pulses are recognized also by their shape rather than their height only. This is important because a large part of the alpha particles hit the electrodes before losing all the kinetic energy. The system is able to cope with some pulse pile-up before changing to total current mode. The spurious pulses caused by vibrations can be recognized and ignored — or the counting halted for a period. In highly vibrating conditions a warning is given and the total current mode used. The background current level can be checked on the field by introducing air with a relatively low radon concentration into the chamber (or rather performing this check before starting the sampling in a new location).

In general, the instrument has a relatively simple and inexpensive construction. On the other hand, the pulse processing and control are somewhat complicated. The micro program is still to be developed. Further development is also needed to have a construction less sensitive to vibrations.

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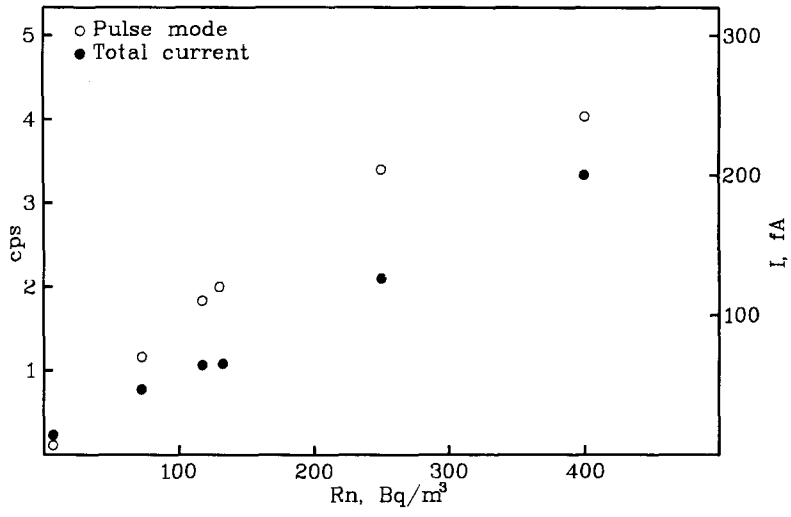

Figure 2. A comparison of the pulse and total current output of the prototype instrument at low radon concentrations.

A METHOD OF MEASURING THE ATTACHMENT RATE EQUIVALENT PARTICLE CONCENTRATION

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INTRODUCTION

When estimating the importance of the aerosol particles on the radon decay products in indoor air, both the concentration and the size of the particles must be taken into account. This can be accomplished by defining an equivalent particle concentration called attachment rate X. Assuming particle size distribution $F(D_p)$, one can give the following equation

$$X = \int \beta(D_p)\, F_N(D_p)\, dD_p \,, \qquad (1)$$

where $\beta(D_p)$ is attachment probability function. Theories dealing with the attachment of free decay products have been discussed by several authors. According to Porstendörfer et al. (1979), the attachment probability is proportioal to D_p in the case of large particles and to D_p^2 for fine particles. The experimental results presented by Porstendörfer et al. (1979) and Porstendörfer and Mercer (1978) are in a good agreement with this theory.

The measurement of the attachment rate or attachment rate equivalent particle concentration is possible if the sensitivity of the measuring instrument has the same particle size dependency as it is the case with attachment probability, eq. (1). The conventional particle measuring instruments, viz. condensation nucleus counters, optical particle counters and particle mass monitors, do not fulfil this requirement. By using the submicrometer particle size analyzers it is, however, possible to determine the particle size distribution and thereafter calculate the attachment rate by multiplying particle concentrations in each size channel by corresponding attachment probability.

In this study, the principles of the electrical aerosol measuring technique can be utilized in developing an alternative method. This measuring technique is based on the measurement of electric current generated by unipolarily charged particles. The diffusion charging of aerosol particles is a process which resembles the attachment process of free decay products. Thus, it is reasonable to assume that the particle size dependencies of these two processes are near to each other.

ELECTRICAL AEROSOL DETECTOR

The electrical aerosol detector, the principle of which is shown in Fig. 1, is a simplified version of the electrical aerosol analyzer (EAA) (Liu and Pui, 1975), a well known instrument used in numerous studies dealing with submicrometer aerosol particles. The basic idea of the present method (Lehtimäki, 1983) is to measure the electric current carried by unipolarily charged particles. Compared to the traditional method, the major difference is that the electric current is measured directly from the aerosol charger without using a filter charge collector. Aerosol charger, high-voltage supply and control electronics form a unit which has been isolated from the ground potential. The electric current escaping from this unit is measured by the electrometer.

Essential features of the present method are the careful shielding of aerosol charger floating at the input of the electrometer and the inductive power transfer to the high voltage supply. Also, careful elimination of the disturbing capacitive currents and prevention of the ions from escaping the charger are of great importance. Despite the corona voltage being 5000 V and the corona current stabilized to 2.3 μA, it is possible to measure aerosol currents down to 0.1 pA.

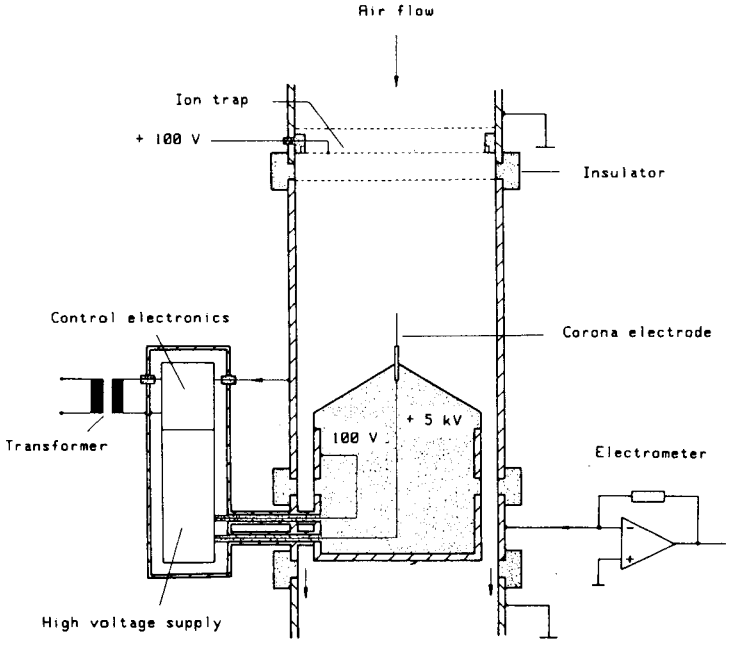

Figure 1. Schematic diagram of EAD. Cylindrical corona charger.

The current I generated by the flow of charged particles is

$$I = \int S_N(D_p)\, F_N(D_p)\, dD_p \,, \qquad (2)$$

where $S_N(D_p)$ is the number sensitivity function of the detector. The sensitivity function of the electrical aerosol detector was determined by using EAA and optical particle size analyzer (PMS Las x) as reference instruments. According to the experimental results, the sensitivity function of EAD has approximately the form

$$S_N(D_p) = K\ D_p^{\,b} \qquad (3)$$

$$K = 1.7\ 10^{-14}\ A\ cm^3$$

$$b = 1.2 \qquad (0.02\ \mu m \leq D_p \leq 1.0\ \mu m)$$

This result is in agood agreement with the results presented by Liu et al. (1975). By comparing the sensitivity function with the attachment probabilty function, it can be concluded that $S_N(D_p)$ can be used as a reasonable approximation for $\beta(D_p)$. Thus, it is reasonable to assume a good correlation between the attachment rate and the current given by the EAD.

EXPERIMENTS WITH RADON DECAY PRODUCTS

The feasibility of the present method in the real-time measurement of the attachment rate equivalent particle concentration was estimated in laboratory conditions. The measuring system consists of a test room (volume 28 m^3), radon source (1 mCi $RaCl_2$ salt dilution), radon measuring instruments, particle generators, and the electrical aerosol detector. The equilibrium factors were measured with a system consisting of a filter sampler for the total activity and a copper screen sampler for the free decay products. The reference radon concentration was measured with an ionization chamber-type radon monitor. In these experiments, the equilibrium factors F_t (total alpha energy) and F_f (alpha energy of free decay products) were determined as a function of EAD's reading for different types of aerosols. During the measurements only the concentration and type of the particles were varied. In these conditions, equilibrium factors can be assumed to be single-valued functions of the attachment rate. If the EAD's reading correlates with the attachment rate, the equilibrium factors should be single-valued functions of EAD's current as well.

Different types of aerosols were used in order to clarify the importance of particle size and composition on the equilibrium factors. The results of these measurements are shown in Fig. 2. In this figure, the equilibrium factors for total activity F_t and the free fraction F_f are presented as a function of EAD's current.

Despite the relative wide range in the size of the particles, only a moderate scattering in the measuring points can be observed. These results can be regarded as a reasonable indication of EAD's suitability to the monitoring of the attachment rate or attachment rate equivalent particle concentration.

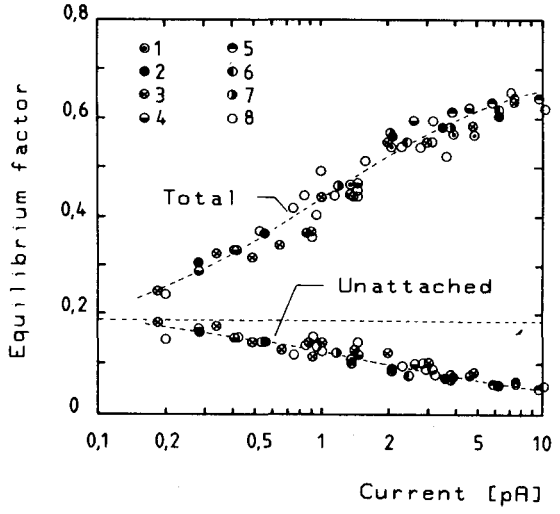

Figure 2. Equilibrium factors F_t and F_f as a function of detector current. Symbols: 1=Arizona road dust (0.3 μm), 2=NaCl particles (0.06 μm), 3=particles generated by the corona discharge (0.02 ... 0.05 μm), 4=NaCl particles (0.035 μm), 5=cigarette smoke (0.1 μm), 6=DOP (0.1 μm), 7=DOP (0.3 μm), 8=laboratory air.

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THE KFK PASSIVE PERSONAL DOSEMETER FOR EXPOSURE TO RADON AND EXTERNAL GAMMA-RADIATION

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INTRODUCTION

The recognition of the role of radon and its decay products in the induction of lung cancer in uranium miners led to exposure limitation guides. The ICRP has recommended limits for the purpose of radiation protection in mines, expressed in terms of Annual Limits on Intake. The efforts required to maintain satisfactory control are quite variable, depending on the mining method, the geological formation, the concentration of uranium and thorium in hostrock and ore, climate etc.General principles of monitoring for radiation protection of workers have been established by the Commission in ICRP Publication 35. The main functions and the various forms of monitoring are analyzed with particular attention given to the design of a monitoring programme and the interpretation of results for external radiation, for surface, air, skin and internal contamination. The requirements for both ambient and staff monitoring in mines should conform with these general principles.

While for the dosimetry of the external γ-radiation the instrumentation is well approved to a high standard, the measurement of internal exposure by radon and its daughters is relative laborious and unprecise. There are several instruments available for grab sampling and continuous monitoring of radon and its daughters in mines and in the environment. Best estimation of exposure, however, can be achieved using continuous or integrating measuring equipment and integrating personal dosemeters. These especially exclude difficulties arising from well known short term variations of radon concentration which can be as much as one order of magnitude or more. There are mainly two techniques used today for radon measurements: active and passive systems. Active techniques need an external power supply for pumps and electronics, which passive devices avoid. They use thermoluminescence detectors or track etch detectors. Detailed information is available from literature.

DOSEMETER DESIGN

For environmental and personal monitoring a small size, integrating dosemeter for the determination of the radon as well as external γ-exposure has been developed at KfK (1). It uses a polycarbonate solid state nuclear track detector (MAKROFOL E) for the registration of α-particles from radon and its short lived decay products. γ-radiation is registered by TLD chips. The design of the dosemeter is shown in Fig. 1, the cross section in Fig. 2. Special attention was given to electrically conductive surfaces for all parts of the dosemeter. Different surface charges occuring on non conductive surfaces will result in inhomogeneous plate out effects of charged radon decay products. This will result in poor reproducibility of the measurement, especially for small size dosemeters, where the distance surface to detector is within the range of the α-particles. The KfK dosemeter is produced from carbon loaded thermoplast by pressure decasting.

CALIBRATION, REPRODUCIBILITY

The quantity measured by the dosemeter is the time integral of the actual radon concentration (2) i. e. the exposure. The radon exposure X is:

$$X_{Rn} = \int_0^T C(t)\ dt$$

where are
X = radon exposure in Bq/m3·d
C(t) = radon concentration at the time t,
T = exposure time

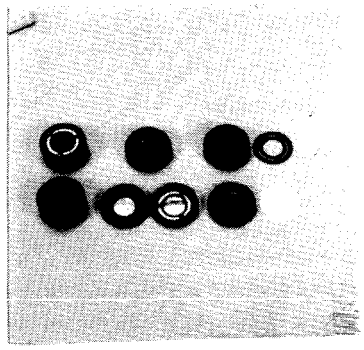

Fig. 1: Design and parts of the KfK Passive, Personal Radon Dosemeter, including TLD for external γ-radiation

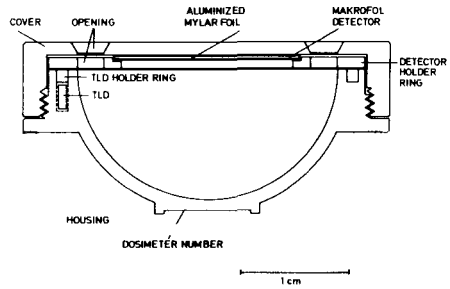

Fig. 2: Cross section of the KfK Passive, Personal Radon Dosemeter, including TLD for external exposure

Assuming linearity between radon exposure and the number of tracks counted with the sensitivity ε of the diffusion chamber:

$$\varepsilon = X_{Rn}^{-1}\left(\frac{N_1}{A_1} - \frac{N_0}{A_0}\right)$$

where are

ε = radon sensitivity in (tracks/cm2)/(kBqm-3d),
N, = number of tracks counted in the field A,
N0 = background tracks in the field A0,
A, = area counted in cm2 for the total number of tracks
A0 = and background tracks,

Several calibrations have been done in the past. Fig. 3 shows as an example the results of the CEC intercalibration test organized by the French CEA in the uranium mine of Lodève in 1985

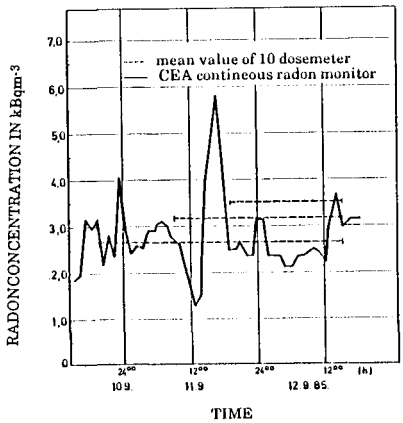

Fig. 3: Comparison of the reading of the CEA continuous radon monitor with the readings of the KfK Passive, Personal Radon Dosemeter

The estimation of the total measuring uncertainty of the radon exposure is done as follows: Taking into account the statistic uncertainty of track counting for the total tracks ($N + N_0$) the relative measuring uncertainty is given for low track numbers by the standard deviation of the background reading S_0 and for the high track numbers by the relative standard deviation s of the exposed set:

$$s(N) = \frac{100}{N} (N + N_0 + S_0^2 + s_0^2 \cdot N^2)^{1/2}$$

where are N, N_0 = number of radiation induced or background tracks,
S_0 = standard deviation of the background reading N_0,
s_i = relative standard deviation caused by systematic uncertainties

The relative standard deviation s(N) vs. number of tracks can be calculated on the basis of the formula and the experimental values S_0 and s_r of an unexposed and an exposed set. The relative standard deviation s_i is given by the measured standard deviation s_r of the exposed set with the number of tracks N_r:

$$s_r^2 = s_i^2 + N_r^{-1}$$

S_i is the total systematic uncertainty resulting mainly from the scatter of the individual detector sensitivity within the set as well as from the etching and counting technique. With the radon sensitivity of the track detector in the diffusion chamber the lowest detectable radon exposure X_L can be calculated

$$X_L = N_L \cdot \varepsilon_{Rn}^{-1}$$

where N_L is the lowest detectable number of tracks at a relative standard deviation of 50%. These calculations have been confirmed by various series of measurements (3).

PERSONAL DOSEMETER FOR MINERS

In rough environments with dust and high moisture loads, active dosemeters are prone to failure. For instance, variations in air flow through the filter can falsify the readings in an unpredictable ways. In addition, the equipment requires a more or less high amount of maintenance.

The passive dosimeters has none of these drawbacks. It is integrated into miner's helmets (Fig. 4) in such a way, that the dosimeter opening, through which gases and aerosols are exchanged, faces the interior of the helmet. This protects the dosimeter mechanically and from direct impacts by dust and water sprays. In environments with mud, high dust or moisture loads, the dosemeter can be covered with a hydrophobic fiberglass filter, which prevents wetting of the dosimeter inside as well as the penetration of dust. This ensures that the measurements are reliable also under rough operation conditions.

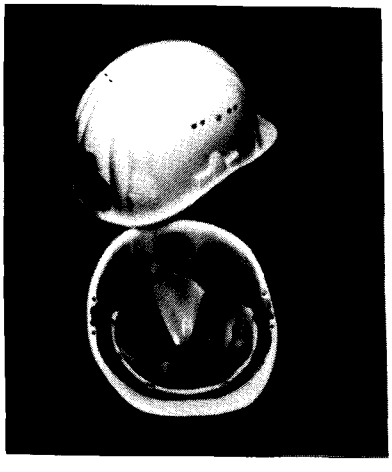

Fig.: 4 KfK Passive Personal Radon Dosemeter integrated in miners helmets

The main disadvantage associated with the use of a filter is, that only the radon gas and not the dose effective short-lived decay products are measured. Therefore, a mean equilibrium factor has to be applied, which is estimated from grab sample measurements. However, these measurements are still necessary also where personal dosimeters are used, i. e. for clearing workstations after changes in ventilation patterns or after a breakdown of the fans. Personal dosimeters only integrate, indicating the monthly exposure received. The working place measurements prescribed by the mining authorities are carried out routinely in uranium mining, and their statistical evaluation results in the necessary accuracy for converting radon concentrations into lung exposures. Another negative characteristic is, that the passive detectors register radiation all the time and not only during working hours. This however can easily be corrected by using some additional dosimeters for the background in the pithead. The time, where the miner was working, is recorded as Ts. The total exposure time of the dosemeter is T. The individual miners exposure $X_{Rn,ind}$ then can be calculated as follows:

$$X_{Rn,ind} = X_{Rn,Pers.dos.} - X_{Rn,controls} \cdot \left(1 - \frac{T_s}{T}\right)$$

where are $X_{Rn,pers.dos}$: radon exposure of the personal dosemeter
$X_{Rn,controlls}$: radon exposure of the control dosemeter

The KfK Personal Dosemeter for miners is used for official exposure measurements in a German uranium mine. Extensive tests side by side with the CEA dosemeter have been performed in German and French mines together with the CEA. In addition the dosimeter is used in Brasilian and Argentinian mines. The main advantage of this dosimeter is, that the individual exposure measurement can be done with ruggedized cheep dosemeters having no moving parts to fail under extreme environmental conditions. Also no power supply and in consequence recharging stations are necessary. An advantage also is, that the dosemeter can be used as it is without any modifications for en vironmental monitoring of mining sites and houses.

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AN INTEGRATING POTENTIAL α-ENERGY MONITOR FOR ENVIRONMENTAL RADON MEASUREMENTS

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INTRODUCTION

The effective lung dose equivalent is calculated from the inhaled potential α-energy of Rn-222 daughters and the unattached fraction f_p of the total potential α-energy (1,2). The unattached fraction f_p can be estimated from the concentrations of Rn-222, Rn-222 daughters and unattached Po-218 by using Jacobi's model (3) and some assumptions. In order to estimate the effective dose equivalent due to inhalation of Rn-222 and its daughters, therefore, it is necessary to measure the long-term average concentrations of Rn-222, potential α-energy of Rn-222 daughters and unattached Po-218 in various environments.

POTENTIAL α-ENERGY MONITOR

The cross-sectional view of the constructed potential α-energy monitor is illustrated in Fig.1. Air is drawn through a wire screen and a membrane filter with a small diaphragm pump. Consequently, unattached and attached Rn-222 daughters are collected on the screen and the filter, respectively. Alpha-particles from the Rn-222 daughters on the screen and the filter are detected with cellulose nitrate (CN) films. Since the geometry between the CN film and the screen or the filter and the thickness of absorbers are determined as shown in Fig.1, only 6.00 MeV α-particles from Po-218 on the screen and only 7.68 MeV α-particles from Po-218 on the filter produce etchable tracks on the CN films.

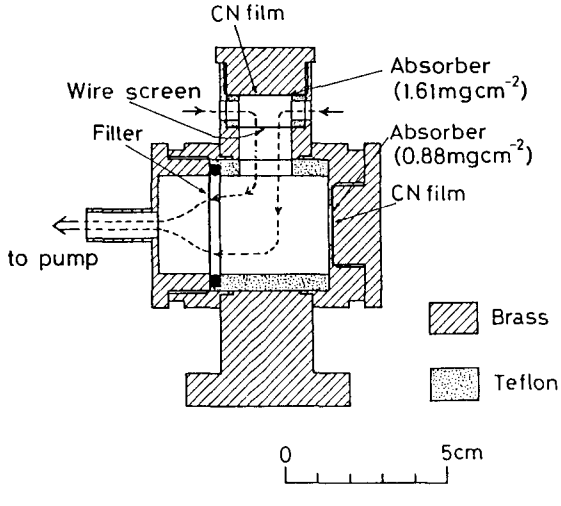

Fig.1 Integrating potential α-energy monitor.

The CN films, the wire screen and the filter are exchanged at intervals of one week. The CN films are etched for 180 min at 60°C in 2.5 N NaOH solution. The etched tracks are counted using a microfiche reader having a magnification of 24 times.

CONCENTRATIONS OF EQUILIBRIUM EQUIVALENT RN-222 AND UNATTACHED PO-218

The relationship between track density on CN film and Rn-222 daughters concentrations in air is given by

$$D = V \cdot \eta \left(\frac{C_A}{\lambda_A} + \frac{C_B}{\lambda_B} + \frac{C_C}{\lambda_C} \right) \quad (1)$$

where D is the track density in tracks cm^{-2}, V is the total volume of the air passing through the filter in m^3, η is the detection efficiency of 7.68 MeV α-particles in tracks $\cdot cm^{-2}$ per disintegration, C_A, C_B, C_C are the activity concentrations of Po-218, Pb-214 and Bi-214 in $Bq \cdot m^{-3}$ and λ_A, λ_B, λ_C are the decay constants of Po-218, Pb-214 and Bi-214 in s^{-1}. The detection efficiency was evaluated to be 0.0273 by using the Monte Carlo calculation.

The potential α-energy concentration can be approximately expressed by the activity concentration of the Rn-222 daughters. Therefore, the equilibrium equivalent Rn-222 concentration $X_{eq,Rn}$ ($Bq \cdot m^{-3}$) is given by

$$X_{eq,Rn} = 8.92 \times 10^{-3} \frac{D}{V} . \quad (2)$$

On the other hand, the relationship between track density D_A (tracks $\cdot cm^{-2}$) and unattached Po-218 concentration C_A^f ($Bq \cdot m^{-3}$) is given by

$$D_A = V \cdot \varepsilon_c \cdot \varepsilon_e \cdot \eta_A \frac{C_A^f}{\lambda_A} \quad (3)$$

where V is the total volume of the air passing through the screen in m^3, ε_c is the collection efficiency of the wire screen for collecting unattached Po-218, ε_e is the fraction of α-particles emerging from the screen, η_A is the detection efficiency of α-particles in tracks $\cdot cm^{-2}$ per disintegration and λ_A is the decay constant of Po-218 in s^{-1}. The collection efficiency ε_c and the fraction emerging ε_e are estimated to be 0.95 and 0.75 from the data of James et al. (4). The detection efficiency was obtained to be 0.052 by the Monte Carlo method. Therefore, eq.(5) can be written as

$$C_A^f = 0.099 \frac{D}{V} . \quad (4)$$

RESULTS

To test the performance of the potential α-energy monitor through the year, the concentrations of equilibrium equivalent Rn-222 and unattached Po-218 have been measured from March 1985 in laboratory air and from February 1986 in outdoor air at intervals of one week. Figure 2 shows the outdoor equilibrium equivalent Rn-222 concentrations and the unattached Po-218 concentrations from May 1986 to July 1987. The weekly mean Rn-222 concentrations observed with an electrostatic continuous Rn-222 monitor is also shown in Fig.2. These concentrations show same seasonal variation of a summer minimum and a winter maximum. The equilibrium equivalent Rn-222 concentrations range from 2.0 to 5.5 $Bq \cdot m^{-3}$ in outdoor air and from 4.4 to 11.0 $Bq \cdot m^{-3}$ in laboratory air. The unattached Po-218 concentrations vary from 0.4 to 1.6 $Bq \cdot m^{-3}$ in outdoor air and from 1.5 to 4.4 $Bq \cdot m^{-3}$ in laboratory air.

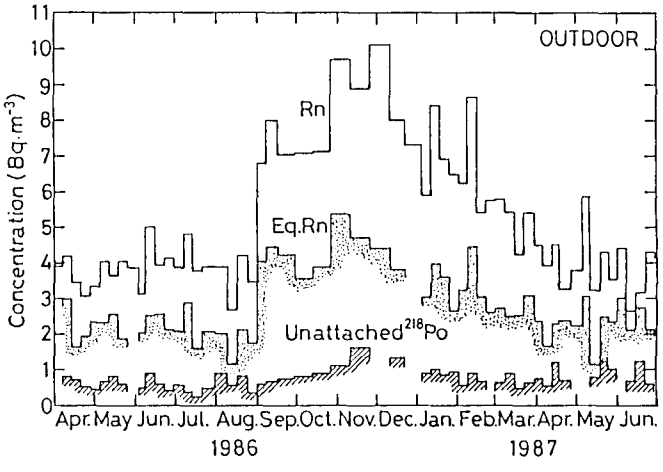

Fig.2 Concentrations of equilibrium equivalent Rn-222 and unattached Po-218 measured with the monitors in outdoor air at intervals of one week.

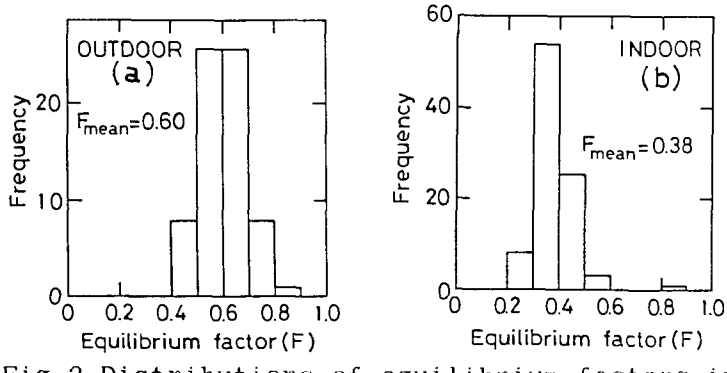

Fig.3 Distributions of equilibrium factors in (a) outdoor (b) indoor air.

The equilibrium factor (F) and the unattached fraction of Po-218 (f) were calculated from the data of the concentrations of Rn-222, equilibrium equivalent Rn-222 and unattached Po-218. Figure 3 shows the distributions of equilibrium factors in

outdoor and indoor air. Annual mean F values were 0.60 in outdoor air and 0.38 in indoor air. The F values are in agreement with the other experimental results (1). The distributions of unattached fractions of Po-218 (f) are shown in Fig.4. Annual mean f values were 0.17 in outdoor air and 0.22 in indoor air. Since the aerosol concentration in laboratory air is smaller than that in outdoor air, the f values are compatible with the aerosol concentrations.

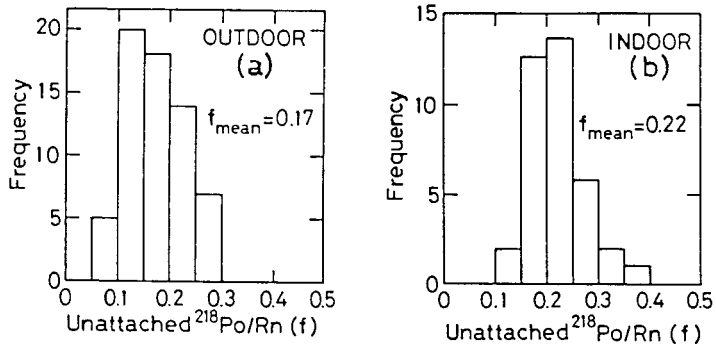

Fig.4 Distributions of unattached fractions of Po-218 in (a) outdoor (b) indoor air.

SUMMARY

In the present study, a new type of integrating monitor has been developed for measuring simultaneously the equilibrium equivalent Rn-222 concentration and the unattached Po-218 concentration. The CN film over the wire screen distinguishes α-particles of Po-218 from Po-214 and the CN film facing the filter distinguishes α-particles of Po-214 from Po-218. The geometrical efficiencies of the α-particles impinging on the CN films were calculated by the Monte Carlo method.

In a small survey, annual mean equilibrium factors were 0.60 in outdoor air and 0.38 in indoor air. on the other hand, annual mean unattached fractions of of Po-218 were 0.17 in outdoor air and 0.22 in indoor air.

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RADIOLOGICAL PROBLEMS IN PHOSPHORIC ACID PLANTS

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ABSTRACT

The Uranium and Radium contents in phosphate may cause some radioprotection problems in phosphoric-acid production plants. In this work authors take into account the results of some different researches carried out in two Italian plants.

The measurements carried out have considered:

- external dose ratio
- Radon and daughters concentration in air and their contribute to internal dose
- Radium and Uranium intake by inalation.

The radiological consequences come from a significant presence of Radium scales in tubings and vessels which may cause controlled areas.

In order to prevent this situation of operational disease solutions are analysed.

CONSIDERATIONS ABOUT TREATMENT AND HANDLING OF RADIOACTIVE SCALES FROM PETROLEUM AND FORMATION WATERS

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ABSTRACT

The phenomenon of "scale deposition" in oil well tubing may generate some radioprotection problems because of the presence of Uranium and Radium concentrations.

In this work the authors have carried out a program of environmental radioactive measures around oil and gas field.

The results take into account spectrometric and integrated dose values which are discussed and analysed in function of:

- age of oil/gas well
- geological configuration
- type of exploitation

In any case health physics problems exist, not relevant but significant with particular care in handling and treatment of piping and tubing.

Radiological conclusions are discussed.

CHEMICAL AND RADIOMETRIC DETERMINATION OF URANIUM AND THORIUM IN ZIRCON SANDS AND PHOSPHORITES AFTER SEPARATION BY EXTRACTION CHROMATOGRAPHY

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INTRODUCTION

Zircon sands, essentially composed of zircon, are materials widely used in zirconium extraction industry, in foundry sands, in refractories and ceramics [1]; phosphorites are the starting material for the industrial production of phosphoric acid, fertilisers and plasters [2].

Zircon sands and phosphorites contain sensible concentration of uranium, thorium and radium representing a potential radiation risk for the professionally exposed staff [1, 3].

The measurements by gamma spectrometry give often too high uranium contents if compared with those obtained by other methods, because the members of uranium series are not always in radioactive equilibrium with the parents [4]. Consequently it is useful to operate both by analytical and radiochemical techniques which permit the determination of the elementary thorium and uranium and also of their isotopic composition.

The aim of the present work was to find out two different methods by Extraction Chromatography [5] for the separation of uranium, thorium and radium before their radiometric determination by ZnS(Ag) alpha counting (Ra-226) and by alpha spectrometry (U, Th).

ZIRCON SANDS

100 mg of powder were dissolved as described by Sill [6]; Zr, U and Th were precipitated as hydroxides adding concentrated NH_4OH up to pH 9. The precipitate was washed twice by centrifuging with diluted NH_4OH and dissolved in 5 ml 6M HNO_3; then 10 ml water and solid NH_4NO_3 up to saturation were added. The solution was percolated through a chromatographic column consisting of 5 g 50÷100 mesh microporous polyethylene (Microthene-710) supporting 4 ml 0.5 M tri-n-octylamine (TNOA). After washing with 40 ml 10M NH_4NO_3, uranium was eluted with 60 ml 6M HNO_3 and thorium with 60 ml 8M HCl, (Fig. 1). From the fraction containing uranium, 1 ml was taken for the fluorimetric analysis [7], and from that containing thorium 10 ml were withdrawn for colorimetric analysis with Arsenazo III [8]. The remainder of the two solutions was reserved for U and Th

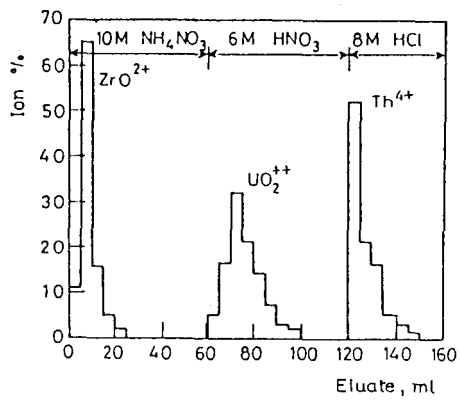

Fig. 1 - Elution diagram of zirconium, uranium and thorium by a Microthene-TNOA chromatographic column. Flow-rate: 1 ml·min^{-1}.

electrodeposition according to Talvitie [9]. Then the disks were counted by alpha spectrometry: Fig. 2 shows the alpha spectra of the isolated thorium and uranium.

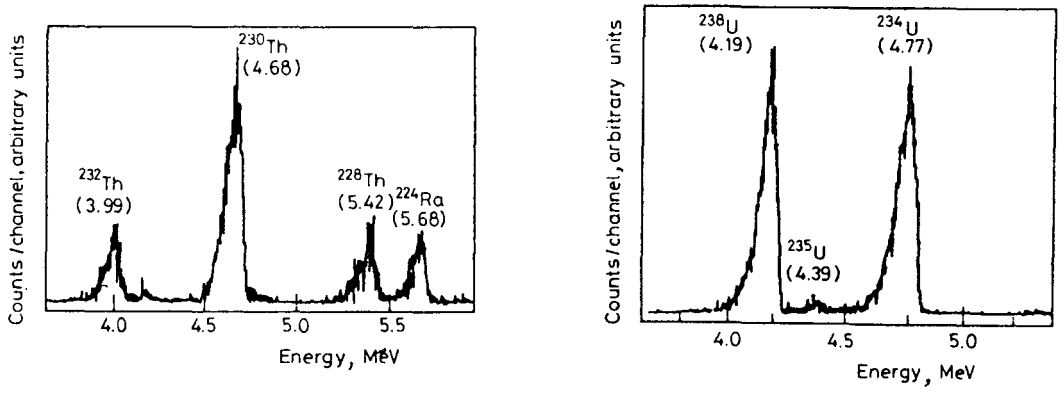

Fig. 2 - Alpha spectra of thorium and uranium isolated from a zircon sand sample.

Table I shows the results obtained for a zircon sand sample; a good agreement between the radiometric and the chemical-physical method can be derived.

Moreover these values are in good accordance with those obtained by other authors [10, 11] by radiometric techniques (0.023% for uranium and 0.017% for thorium). The alpha spectra also show that U-238 and U-234 are in secular equilibrium.

The analysis of the spectra also shows a complete decontamination of uranium from thorium and of thorium from uranium.

TABLE 1

Uranium and thorium contents obtained with radiometric and chemical-physical methods

Sample number	Th % by colorimetry	Th % by α spectrometry	U % by fluorimetry	U % by α spectrometry
1	0.022	0.018	0.021	0.025
2	0.021	0.022	0.023	0.024
3	0.016	0.016	0.023	0.025
4	0.019	0.019	0.020	0.023
5	0.016	0.017	0.024	0.022
6	0.022	0.019	0.029	0.023
Mean±σ	0.019±0.003	0.018±0.002	0.023±0.003	0.024±0.001

PHOSPHORITES AND THEIR DERIVATIVES

100 mg of phosphorite, plaster ($CaSO_4.2H_2O$), or H_3PO_4 were dissolved in 6M HNO_3; some fluorimetric measurement were carried out direclty [7]. As for as the separation of Ra-226 and uranium is concerned a chromatographic column was prepared consisting of 3 g Microthene-710 supporting 2.5 ml 0.2M Tri-n-octyl-phosphine-oxide (TOPO). Fig. 3 shows the elution diagrams for the two radio-nuclides; radium was coprecipitated [12] as $Ba(Ra)SO_4$ and counted by a ZnS(Ag) detector up to equilibrium (20 days); uranium was electroplated [9] and counted by alpha spectrometry.

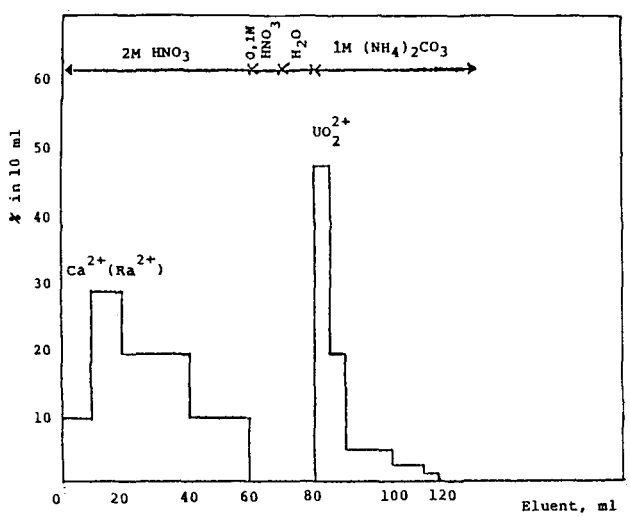

Fig. 3 - Separation Ra^{++}-UO_2^{++} in a phosphorite by a Microthene-TOPO column.

The uranium concentrations found in four different phosphorites (Morocco, Togo, Giordania, Boucraa) was respectively 0.010, 0.009, 0.006 and 0.009%; in 30% H_3PO_4 (Boucraa) the concentration was found to be 0.0045%.

The uranium concentration found by alpha spectrometry (Fig. 4) and calculated by adding a known activity of U-232 as on yield internal standard, was in good agreement with the fluorimetric results. Also in this case the uranium isotopes were found to be in a secular radioactive equilibrium.

Radium-226 concentration were found 1.22 KBq/Kg for phosphorite Morocco (σ = 0.23), 0.69 KBq/Kg for plaster Togo (σ = 0.03) and 0.35 KBq/Kg for plaster Boucraa (σ = 0,06); Radium-226 in H_3PO_4 was not detectable.

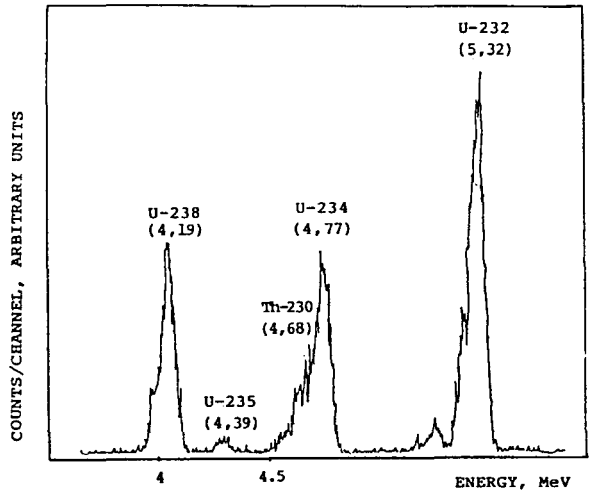

Fig. 4 - Alpha spectrum uranium isolated from a phosphorite sample.

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A COMPARISON OF RADON EMANATION RATES FROM FLY ASH AND FROM CONCRETE MADE WITH FLY ASH

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ABSTRACT

Although radium in the earth is often the primary source term for radon entering dwellings, building materials may also contribute significantly to indoor radon levels in certain cases. Fly ash, a byproduct of coal combustion, has been used for many years as a pozzolanic admixture in the production of cement in many countries. Because fly ash has been shown to have radium concentration in the range of 10 pCi/g (0.37 Bq/g) in some cases, the extent to which radon can emanate from fly ash and from products prepared with such ash is of interest.

This study was designed to determine radon emanation coefficients for fly ash as a function of particle size and moisture content, and to determine radon exhalation (radon released per unit surface area) for concrete prepared from fly ash of known particle sizes and radium content. Radon emanation measurements were made for fly ash and concrete samples by means of specially designed emanation chambers. The radon exhaled by the samples was sampled at appropriate time intervals by a standard scintillation cell technique.

For dry fly ash, emanation coefficients were found to vary with particle size, ranging from 9.6×10^{-4} for large particle sizes (421-891 mμ) to 3×10^{-3} for small particle sizes (less than 75 mμ). For ash with 20% moisture content the emanation coefficients varied from 2.0×10^{-2} for large particles to 1.6×10^{-1} for small particles. However, when such fly ash was incorporated into concrete, the particle size dependence of the emanation could no longer be observed. For concrete prepared with fly ash, only the total radium content effected the radon exhalation.

RADIOACTIVE SCALE IN OFF-SHORE OIL INSTALLATIONS

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INTRODUCTION

The deposition of naturally occurring radionuclides from the Uranium-238 and Thorium-232 series onto pipes and valves in oil-wells has been known about for many years. (Campbell, 1953; Gott & Hill, 1953). The only problems it was considered to pose were those associated with the difficulties it introduced into well logging operations, making any logs using nuclear tools impossible. (One type of log uses a Caesium-137 and an American-241/Beryllium source to evaluate the oil bearing formation). The potential health risks of the deposits to anyone working on contaminated downwell pipes (tubulars), or process line valves, pumps and vessels were ignored or not considered. Despite this historical background it came as a surprise to UK North Sea Oil producers to find that they had deposits of radioactive material on board their oil producing platforms. They did, however, realize the health hazard associated with these deposits on account of the legislation associated with the other off-shore uses of ionising radiations and the control of the hazard now costs the oil industry several millions of pounds each year.

SCALE PRODUCTION

Experience has shown that the problem is generally, but not always, associated with injected water breakthrough into the oil production zone. Seawater injection, a secondary production mechanism, is used to maintain downhole pressure and was not intensively used until North Sea oil development. Hence, the problem is more widespread in North Seal oil production than it has been elsewhere. The Jurassic sandstones which are the host to many North Sea reservoirs typically have low (1-5ppm) U-238 and/or Th-232, but the oil formations are capped with 'hot' black Kimmeridgian shales. However, formation water (oil-associated water) is the carrier of the radioactivity and this is rarely associated with these active shales. Analysis shows that Radium-226 and Radium-228 and their daughters are responsible for this activity and are essentially unsupported by their U and Th parents. Theories as to these Ra anomalies centre on alpha particle recoil reactions, either directly ejecting Ra into solution or making the decay site more susceptible to leaching; and the continual process of silica dissolution (Kraemer and Reid, 1984).

The formation waters also contain many other cations and anions the one of particular importance being the barium ion. When the injection water breaks through and is mixed with the formation water then the cation/anion ratios are drastically changed. Sea water is high in sulphate, and barium sulphate is sparingly soluble, mixing results in precipitation. Turbulent mixing is expected once it enters the perforated tubulars and beyond. As the barium sulphate precipitates it co-precipitates radium sulphate. The barium sulphate generally forms a hard scale on metal surfaces which can be extremely difficult to remove. The activity of the scale is normally in the

region of 37 Bq/g or less but may rise as high as 3.7 kBq/g. Injection water need not always be present for active deposits to occur. On at least one platform without water injection, drawing oil from two different oil from two different oilfields, quite active deposits have been found associated with rust sites. In other areas active silt deposits are found.

SITES OF SCALE BUILD UP

The scale is first deposited on the tubulars in the perforated zone and can then be found anywhere from there through the oil/water separation vessels and through the produced water discharge system. The actual sites vary from platform to platform with active areas being separated, for no immediately obvious reason, by lengths of clean plant. It does tend to always be present in separation vessels and is associated with areas of turbulence such as just beyond a bend in a pipe. The problem is not limited to the off-shore oil production platforms and radioactive deposits have been found, albeit at much lower levels, in the on-shore oil receiving facilities.

HANDLING PROCEDURES

The hazard from the scale is essentially one following ingestion, particularly by inhalation of scale particles. The external hazard is small, the dose rates from scaled materials rarely rising above 7.5 μSv/h, although this can be exceeded inside a separator or if tubulars are piled up together. Most of the protection thus relies on wearing suitable protective clothing, including masks, stopping the scale from drying out to perhaps produce a dust, and cleaning up after any operation where scale was present. The introduction of protective clothing was not as difficult as anticipated. The drilling crews did not feel their macho image suffered by having to wear it when pulling tubulars from down a well . (Scale has been found in both oil production and water injection wells). It is suspected they were secretly glad to have to wear extra protective clothing as the brine used to hold the well during these operations is corrosive and traditionally produced "brine burns". Although wet when pulled the tubulars rapidly dry out and are immediately end capped to keep any scale inside. During these operations contamination monitors have to be used to identify active tubulars. On off-shore platforms in general but on the drilling floor in particular, contamination monitors have a short half life. Problems can be experienced when tubulars are externally scaled as they then have to be wrapped in polythene. Even when wrapped it can be difficult to ensure they are subsequently lifted in a completion skip not using the normal wire slings. The latter are remarkably efficient at stripping off polythene. The cleaning out of large vessels is relatively easy to control as the much more severe hydro carbon hazard automatically controls the scale hazard. (Contamination measurements can be difficult as intrinsically safe contamination monitors are not available). Care must be taken that the material taken from these vessels is not spread about the platform and controlled areas with shower facilities have to be established. Because of the exposed nature of the environment this often entails erecting plastic screens. The disposal of this material is directly into

the sea subject to a maximum particle size of 1 mm. This involves using grinding equipment and discharge pipes below sea level. Records of quantities and activities must also be kept. During the discharge the platform water makers are generally switched off although it is debatable how necessary this is. Any debris left after pulling tubulars or cleaning a vessel is generally sluiced into the platform drains. These must be kept clear (difficult on most platforms) and monitored for a build up of debris. More problematic tends to be the unexpected repairs on pipe work or valves that have to be carried out. Ensuring that these operations are always monitored is difficult. They can often involve grinding before repair welding takes place thus producing a very fine scale dust which can be ingested if masks are not worn.

The equipment sent to the shore can at the moment only be cleaned at one centre in Aberdeen. Here the scale is water jetted off, ground down to a particle size of 0.2 mm and discharged into Aberdeen Bay. Surveys of the bay using grab samples are undertaken several times a year and all samples have proved negative. The cleaning is expensive and has to be carried out on all scaled items, even those going for scrap, as well as those being refurbished to be used again. Most operators are now spending several hundreds of thousands of pounds a year cleaning items with a very low scrap value. Alternative operations which propose to send the waste to a land fill site are currently being assessed. It is thought these will be cheaper.

FUTURE DEVELOPMENTS

Chemical scale inhibitors are now being injected into most wells and better inhibition is likely to be developed. These inhibit the scale formation but there is some evidence which suggests that occasionally almost pure radium sulphate will plate out when they are being used. This is not visually obvious and can be very tenacious and very active. Downwell scale dissolvers have been tried without much success. An immense effort is being put into obtaining a suitable dissolver but although many work well in the laboratory, in a well under high pressure and temperature they are nowhere nearly as successful.

Basic research into the problem is difficult. Obtaining truly representative samples of formation water from several thousands of feet under the ground is fraught with problems and much time can be wasted working at them. Likewise oil companies can be very reluctant to release core samples from known wells making constructive analysis difficult. Work in Aberdeen has centred upon defining more accurately the exact mode of formation of radium anomalies in the formation waters, i.e. the responsible minerals and the relevant contribution of alpha recoil and silicate dissolution. Also the role of the brine chemistry is being investigated.

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Some Radioecological Aspects from Coal Power Plants

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Many types of lignite burned in Coal Power Plants (CPP) in several countries of Europe as also in USA contain radioactive nuclides, particularly of the uranium series. An enrichment of these nuclides is observed in produced fly ash by the loss of carbon during the combustion process. A large part of the above nuclides escapes from the stack of CPPs and discharges into the atmosphere as gases (vapor phase) or fine particles. Tracers of radium-226 as well as uranium-238 has been determined in soils,rainwaters and snow of CPP regions sometimes a little higher than normal levels. The Valley Model of atmospheric dispersion was applied to simulate our experimental results.

The contamination of soil in CPP region in which the refining of radium and uranium has been carried out for about 30 years of CPP activity has raised the question of the maximum permissible concentration of radium in soil on its uptake by cerials, vegetables, potatoes, sugar beets and various fruits.

OBSERVATION OF THE DIURNAL VARIATION OF THE SHORT-LIVED ^{222}Rn DAUGHTERS CONCENTRATIONS IN THE ATMOSPHERE BY α-RAY SPECTROMETRY

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I. INTRODUCTION

Methods have developed for measuring the air concentrations of the short-lived ^{222}Rn daughters using a surface-barrier detector in uranium mine and house.[1-2] It is difficult for these methods to measure their diurnal variation in the open air because of their lower limit of detection. Kojima and Abe[3] developed a monitor for measuring individual concentrations of the short-lived ^{222}Rn daughters in outdoor air using ZnS(Ag) scintillation detector. The count of α-rays obtained using this method involves that from the ^{220}Rn daughters. When the short-lived ^{222}Rn daughters concentrations are low, this increases their errors.

On the other hand, ^{222}Rn and its short-lived daughters have been used for studying large scale transport of air mass and vertical diffusivity in the lower atmosphere. The correlations between the concentrations of ^{222}Rn and/or its short-lived daughters and individual meteorological elements were given earlier by Ikebe[4] and Shimo.[5] However, there are very few works on the correlations between the short-lived ^{222}Rn daughters concentrations and some meteorological elements that are simultaneously observed.

In this paper, we describe the simultaneous observations of the short-lived ^{222}Rn daughters concentrations in the outdoor air by α-spectrometry and some meteorological elements at two sites, Okayama-shi (34° 35′ N, 133° 52′ E) and Kamisaibara-mura (35° 18′ N, 133° 35′ E) in Japan. Furthermore, the correlation between the short-lived ^{222}Rn daughters concentrations and each meteorological element is discussed.

II. MEASUREMENTS

Air was drown through a 2"-diameter millipore filter (type HA) at the rate of 4.0×10^{-4} $m^3 \cdot s^{-1}$ by low volume air sampler (Staplex, type LV-1) at 1.5 m elevation above the air-ground interface. The short-lived ^{222}Rn daughters were sampled on this filter for about 1 h. About 5 min after sampling, they were counted with a 19.6-mm-diameter surface-barrier detector and multi-channel analyzer. The counting time was 2,000 s. Many reserchers have reported on the departure from radioactive equilibrium among the short-lived ^{222}Rn daughters in the outdoor air. It is desirable that the diurnal variation of their activity ratios are measured with that of their concentrations. However, it is difficult to measure the diurnal variations of their activity ratios, since their concentrations in the outdoor air vary to low level (under 0.37 $Bq \cdot m^{-3}$). It is assumed, therefore, that their

activities are in the ratios 1:1:1 in calculating their concentrations. The relation between the short-lived ^{222}Rn daughters concentrations, A_4, and the net peak area count of 7.69 MeV ^{214}Po α-rays, C_4, is represented by

$$C_4 = \varepsilon\zeta\eta q\frac{1}{\lambda_1}\left[\frac{\lambda_2\lambda_3\lambda_4}{\lambda_1(\lambda_2-\lambda_1)(\lambda_3-\lambda_1)(\lambda_4-\lambda_1)}(1-e^{-\lambda_1 T_c})(e^{-\lambda_1 T_1}-e^{-\lambda_1 T_2})\right.$$
$$-\frac{\lambda_1\lambda_3\lambda_4}{\lambda_2(\lambda_2-\lambda_1)(\lambda_3-\lambda_2)(\lambda_4-\lambda_2)}\left(1+\frac{\lambda_1-\lambda_2}{\lambda_2}R_1\right)(1-e^{-\lambda_2 T_c})(e^{-\lambda_2 T_1}-e^{-\lambda_2 T_2})$$
$$+\frac{\lambda_1\lambda_2\lambda_4}{\lambda_3(\lambda_3-\lambda_1)(\lambda_3-\lambda_2)(\lambda_4-\lambda_3)}\left\{1+\frac{\lambda_1-\lambda_3}{\lambda_2}R_1+\frac{(\lambda_3-\lambda_1)(\lambda_3-\lambda_2)}{\lambda_2\lambda_3}R_2\right\}$$
$$\times(1-e^{-\lambda_3 T_c})(e^{-\lambda_3 T_1}-e^{-\lambda_3 T_2})-\frac{\lambda_1\lambda_2\lambda_3}{\lambda_4(\lambda_4-\lambda_1)(\lambda_4-\lambda_2)(\lambda_4-\lambda_3)}$$
$$\times\left\{1+\frac{\lambda_1-\lambda_4}{\lambda_2}R_1+\frac{(\lambda_4-\lambda_1)(\lambda_4-\lambda_2)}{\lambda_2\lambda_3}R_2+\frac{(\lambda_1-\lambda_4)(\lambda_4-\lambda_2)(\lambda_4-\lambda_3)}{\lambda_2\lambda_3\lambda_4}R_3\right\}$$
$$\left.\times(1-e^{-\lambda_4 T_c})(e^{-\lambda_4 T_1}-e^{-\lambda_4 T_2})\right]\frac{A_4}{R_3}, \qquad (1)$$

where ε is the detection efficiency for the α-ray, ζ is the emerging efficiency, η is the collection efficiency, q is the flow rate of air passing through the filter ($m^3 \cdot s^{-1}$), λ_1, λ_2, λ_3 and λ_4 are the respective decay constants of ^{218}Po, ^{214}Pb, ^{214}Bi and ^{214}Po (s^{-1}), R_1, R_2 and R_3 are the respective ratios of the ^{214}Pb, ^{214}Bi and ^{214}Po activities to the ^{218}Po activity, T_c is the collecting time (s), T_1 is the time interval from the end of the sampling to the start of the measurement (s) and T_2 is the time interval from the end of the sampling to the end of the measurement (s).

The meteorological elements such as flux of solar radiation, wind velocity and temperature were simultaneously observed. The flux of solar radiation was 1-h-average. The wind velocity was 10-min-average. The temperature was momentary value. The flux of solar radiation and the temperature at Kamisaibara-mura were observed about 2 km west from the measuring point at the same elevation above sea level.

III. RESULTS AND DISCUSSION

Okayama-shi is near the coast of Setonaikai and about 1 m above sea level. Kamisaibara-mura is located inland about 81 km north from Okayama-shi and is about 710 m above sea level. From 1985 to 1986, three runs of the observations of the diurnal variations of the short-lived ^{222}Rn daughters concentrations were made at Okayama-shi and Kamisaibara-mura, respectively. The results are shown in Fig. 1. The short-lived ^{222}Rn daughters concentrations range from 0.3 to 6.3 $Bq \cdot m^{-3}$ at Okayama-shi and from 0.4 to 5.7 $Bq \cdot m^{-3}$ at Kamisaibara-mura during the observations. Except one run

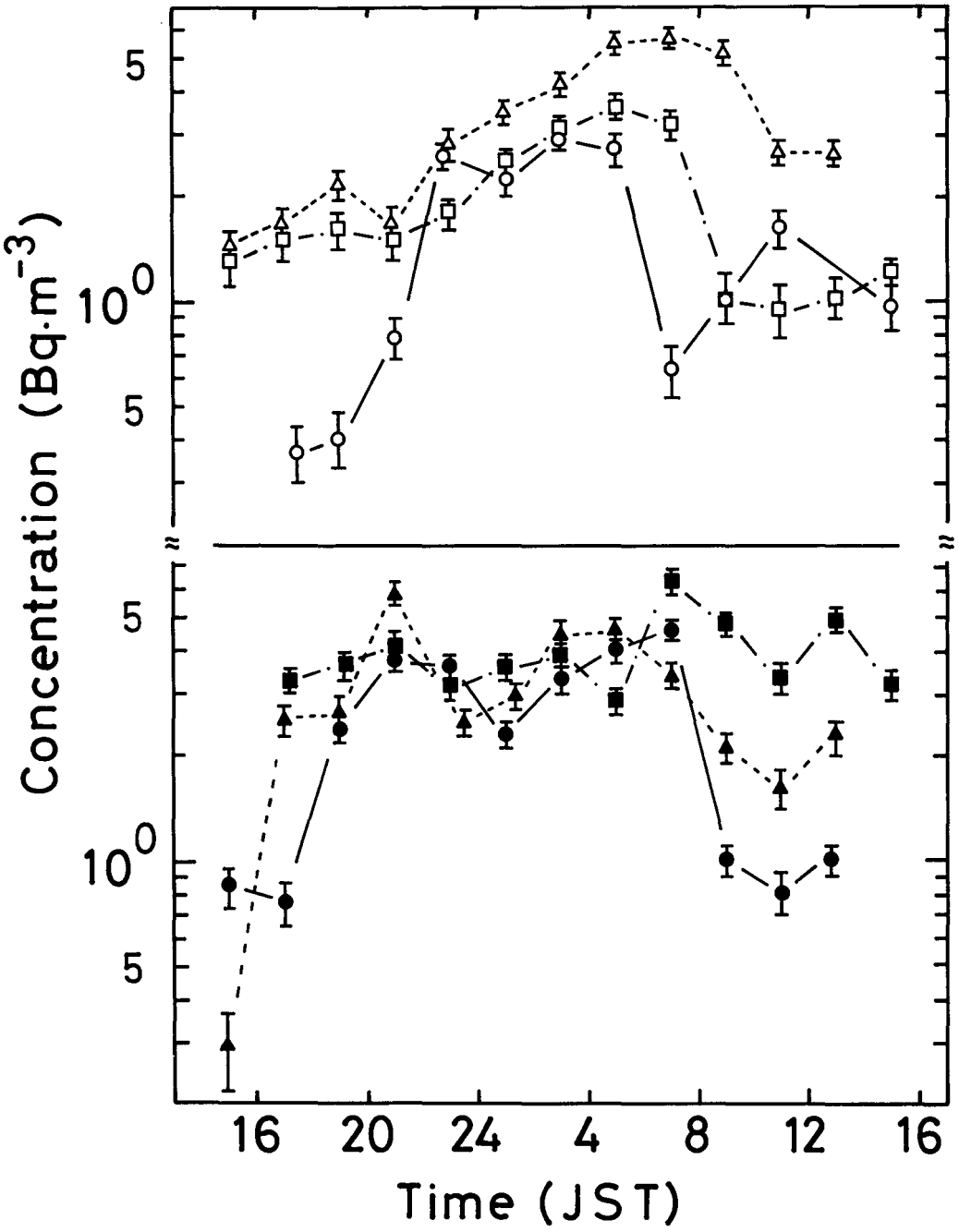

Fig. 1. Diurnal variations of short-lived ^{222}Rn daughters concentrations in the outdoor air.
○ △ □ : Okayama-shi; ● ▲ ■ : Kamisaibara-mura

they are low in the daytime and high in the nighttime. These values and patterns observed elsewhere on land for ^{222}Rn and/or its short-lived daughters.[5)]

From July 7 to 8 in 1986 at Kamisaibara-mura, a distinct diurnal variation of the short-lived ^{222}Rn daughters concentrations was not observed. The total flux of solar radiation is less than those of the other five runs. The wind velocities in the daytime are lower than those of the other five runs. These are thought to be the main cause that the concentrations did not vary so much during this run.

The inverse correlation between the short-lived ^{222}Rn daughters and the air temperature are obtained at both sites excepting one run where there is not very much change in the short-lived ^{222}Rn daughters concentrations. Their patterns differ from one another.

The inverse correlation between the wind velocity and the short-lived ^{222}Rn daughters concentrations is found in each run at Okayama-shi. The inverse correlation is not foune at Kamisaibara-mura, since the wind velocities are low.

IV. CONCLUSIONS

The method developed in this paper make it possible to measure the diurnal variation of the short-lived ^{222}Rn daughters concentrations in the outdoor air using a surface-barrier detector. Their level and variation pattern are independent of the height of the ground above sea level. Furthermore, the concentrations of the short-lived ^{222}Rn daughters are correlated with the wind velocity and the diurnal temperature difference.

Acknowledgements: The authors are grateful to Drs. S. Kawamura, J. Inaba, M. Kashima and Y. Katoh of National Institute of Radiological Sciences for their helpful discussions throughout this work. They are also grateful to Dr. M. Shimo for the derivation of the equation for the calculation of the short-lived ^{222}Rn daughters concentrations. And, they are indebted to the members of the staff of Health and Safety Section of Ningyoh-Tohge Works of PNC for providing the data of the solar radiation and the temperature at Kamisaibara-mura.

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THE VERTICAL DISTRIBUTION OF RADON 222 IN THE ATMOSPHERE

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INTRODUCTION

Atmospheric vertical diffusion is important for studying the vertical transport of heat, momentum, moisture and air pollutants. Radon-222(Rn) and its daughters have been frequently used as tracers for studying the atmospheric transport phenomenon because Rn is radioactive noble gas.

The present paper describes the vertical distribution of Rn in the atmosphere over the Japan Islands and its surrounding sea. In order to study the vertical distribution of Rn, atmospheric samples are collected in the bags aboard an aircraft(helicopter V-107) in flights at different altitudes as high as 3000 m above the sea level, and later Rn concentration in the collected samples is measured in the laboratory.

SAMPLING AND LOCALITIES

Sampling Localities

The sampling localities are shown in Fig. 1. The sampling has been performed at seven different altitudes of 200 m, 500 m, 1000 m, 1500 m, 2000 m, 2500 m and 3000 m above the sea level.

The sampling flights are classified into two groups; one is circular flight over the sampling point and the other is straight flight around the sampling point. In Fig. 1, the former is shown by the circle and the latter is shown by the arrow, respectively.

Sampling Method

Atmospheric samples have been collected in 33-μm saran film bags of 75 cm diameter by using the blower of 8 m^3 air per min. The amount of the sample is about 1.5 m^3. Permeability of Rn from the saran film is experimentally tested by the following method. The atmospheric sample of about 4 m^3 at the ground level was collected in the saran film bag and the decay of Rn concentra-

tion in this bag was measured. The results obtained are shown in Fig. 2. In this figure, the measured points are shown by the circle and the solid line indicates values calculated by the Rn half life of 3.82 day. From this figure, it can be seen that the measured points are in good agreement with the calculated values. Because it takes about 4 days to measure the Rn concentration in the atmospheric samples at seven different altitudes above the land and the sea, it can be said that the permeability of Rn from the saran film is neglected.

MEASUREMENTS

Radon-222 gas in the sample air is adsorbed on the 5 gram activated charcoal in U-shape glass tubes which are cooled by acetone and dry ice. The flowing rate of the sample air is 4 liter per min. The adsorbed Rn is transfered to an ionization chamber by heating the charcoal trap up to 400°C and the pressure in the chamber is made to 1 atm by adding nitrogen gas. The volume of the chamber is 1.5 liter and 3 liter and the applied voltage is 500 volt.

RESULTS

The results for measurements of Rn concentration are classified into two groups; one is maritime atmospheric sample and the other is land atmospheric sample. The former is shown in Fig. 3 and the latter is shown in Fig. 4.

In maritime atmospheric samples, except two samples of Jan. 27, 1987 Sagami bay and June 30, 1987 Kashimanada, Rn concentration decreases with increasing altitude and the value of Rn concentration at 3000 m is around 10 pCi/m^3 or below. On the contrary, in land atmospheric samples, Rn concentration seems not to decrease above 1000 m and the value of Rn concentration seems to become around 20 pCi/m^3.

From these measurements, some difference in the vertical profile of Rn is found between maritime and land atmosphere, but the seasonal variation in the vertical profile is not obvious.

The cause of high Rn concentration in maritime samples of Jan. 27, 1987 Sagami bay and June 30, 1987 Kashimanada is now under consideration.

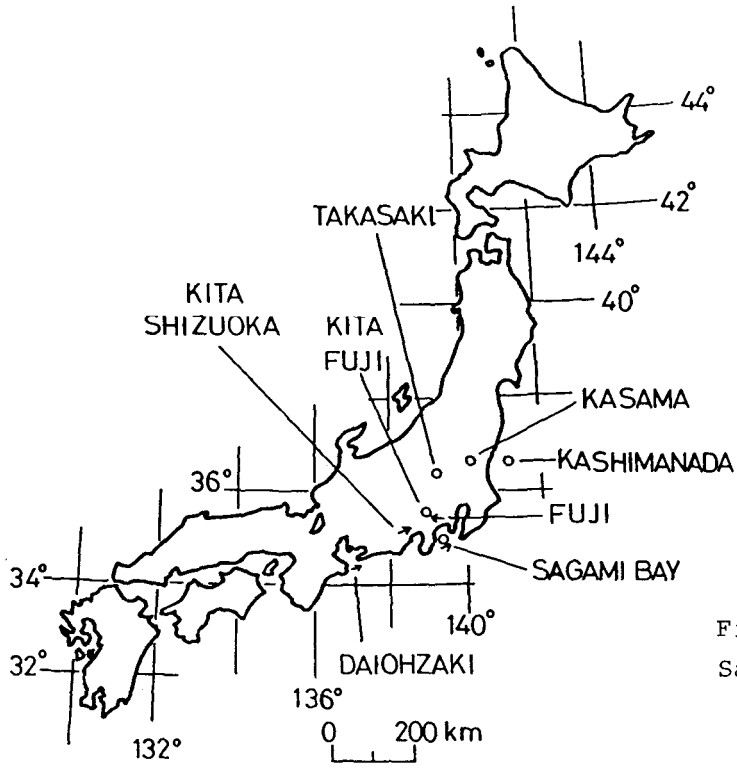

Fig. 1
Sampling localities

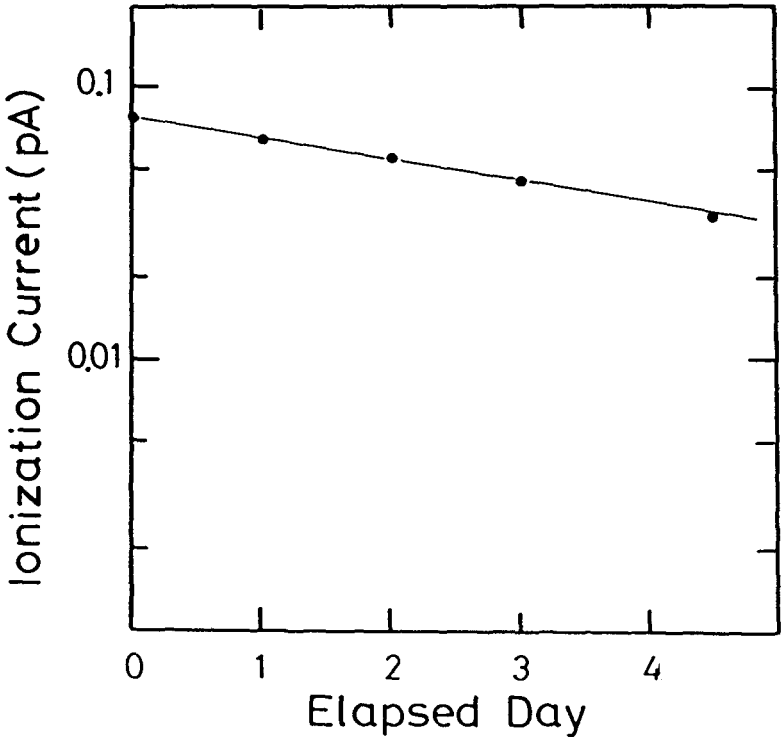

Fig. 2
Decay of Rn concentration in the atmospheric sample collected in the saran film bag

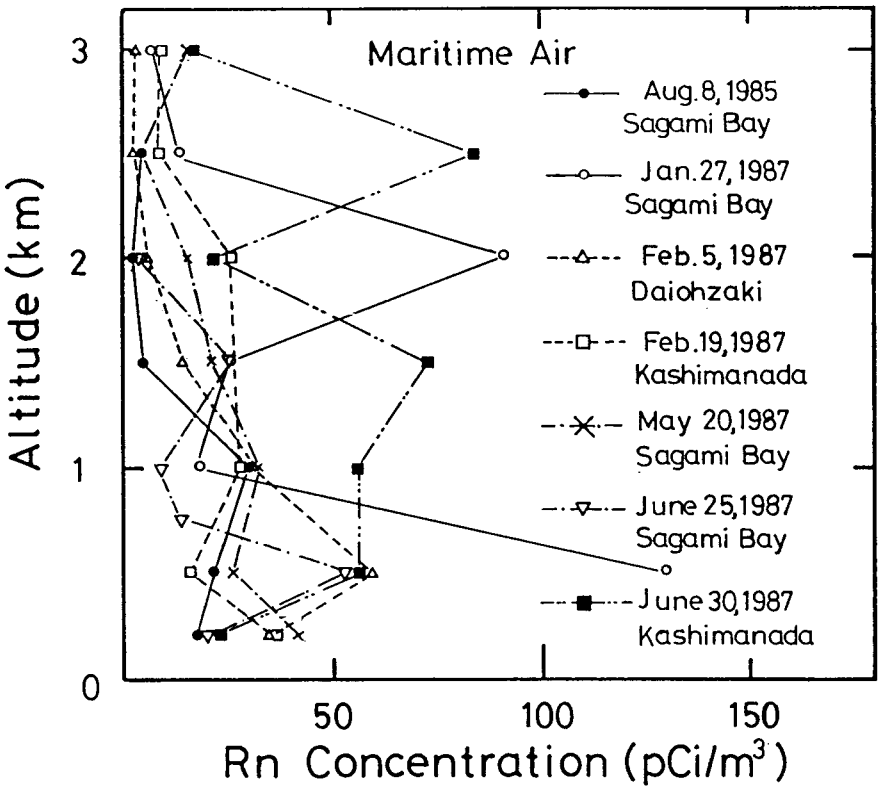

Fig. 3 Vertical profiles of Rn concentration in maritime atmospheric samples

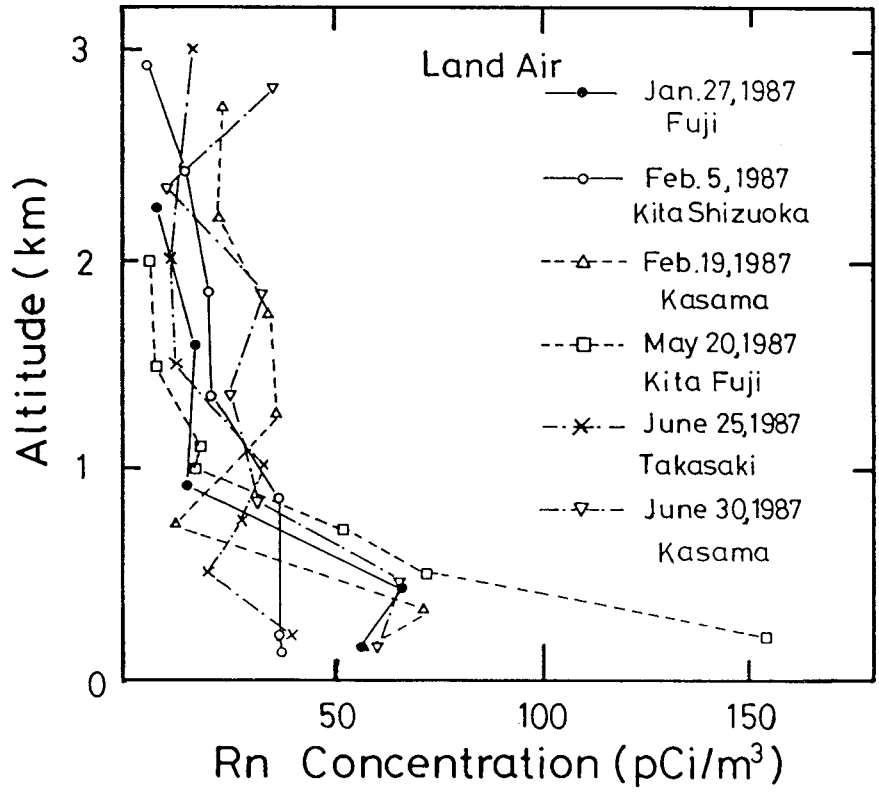

Fig. 4 Vertical profiles of Rn concentration in land atmospheric samples

COMPARISON OF ATMOSPHERIC RADON DAUGHTERS CONTENT OBSERVED AT MOUNTAIN PEAK WITH NEARBY PLAIN

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INTRODUCTION

It is well known that the atmospheric radon daughters content at a plain shows the diurnal variation whose high values appear in the night and low values in the afternoon (1-3). These high values are said to be caused by the inversion of the vertical atmospheric temperature gradient.

The variations of the atmospheric radon daughters content at some height have not been studied. Only a small number of data on the instantaneous content observed with an airplane were reported. Much information is necessary about the differences of the atmospheric radon daughters content at several nearby points where geographical and meteorological conditions are different.

Continuous measurements of the atmospheric radon daughters content were carried out at an isolated mountain peak to study its variation at high altitude. Simultaneous measurements were carried out at the center of a nearby plain and a seacoast in order to compare the diurnal variations of the content under different geographical and meteorological conditions. When the top of the mountain was covered with rainy cloud, the radon daughters contents in precipitation were also observed there. These data will contribute to some extent to the rainout and washout model of the radon daughters content in precipitation.

EXPERIMENTAL

The observation at the center of a plain was carried out at Fukui located 20 km interior from the coast of the Sea of Japan in the Central Japan. The observatory Mikuni located at a seacoast was 22 km distant to the north-northwest from Fukui. The observatory at a mountain peak, which was placed on the Kunimi-Dake mountain of 656 m height, was 11 km distant to the west from Fukui. This mountain stands alone near the Sea of Japan.

The atmospheric radon daughters contents at these three observatories were measured by using the instruments equipped with an aerosol filtration and a ZnS(Ag) scintillation counting system. During the filtration at Kunimi-Dake, filter-papers were exposed to the open air directly in order to catch the cloud droplets simultaneously with the atmospheric radon daughters, without any absorption by the inlet pipe wall.

The radon daughters contents in the precipitation were measured with a well-type NaI(Tl) scintillation counter. The details of the measuring methods were described in the previous papers (4,5).

RESULTS AND DISCUSSION

Figure 1 shows an example of the time variations of the atmospheric radon daughters contents observed at Fukui, Mikuni and Kunimi-Dake at the same time. The atmospheric radon daughters content at Fukui showed the typical diurnal variations in the calm days as mentioned before. The time variations of their content at Mikuni were fairly similar to that at Fukui. Sometimes, the contents at Mikuni and Fukui were slightly different, but generally both values and variations were the same. On the other hand, the atmospheric radon daughters content at Kunimi-Dake showed no diurnal variation as observed at Fukui and Mikuni. In the calm days, it showed sometimes the time variation whose low values appear in the night and high values in the daytime. These are the remarkable phenomena observed at the mountain peak.

From our observation of the inversion layers at Fukui using the sodar, an instrument to measure the height of the atmospheric inversion layers of the temperature with pulsed sound echoes, their heights were ordinarily 200-300 m above the ground surface. Kunimi-Dake is higher than these heights, and, from the top of Kunimi-Dake in the calm night, the inversion fog were often observed below the top and spreading over the nearby plain and valley. When the sun rising, the inversion layer vanishes gradually and the radon daughters accumulated under the layer

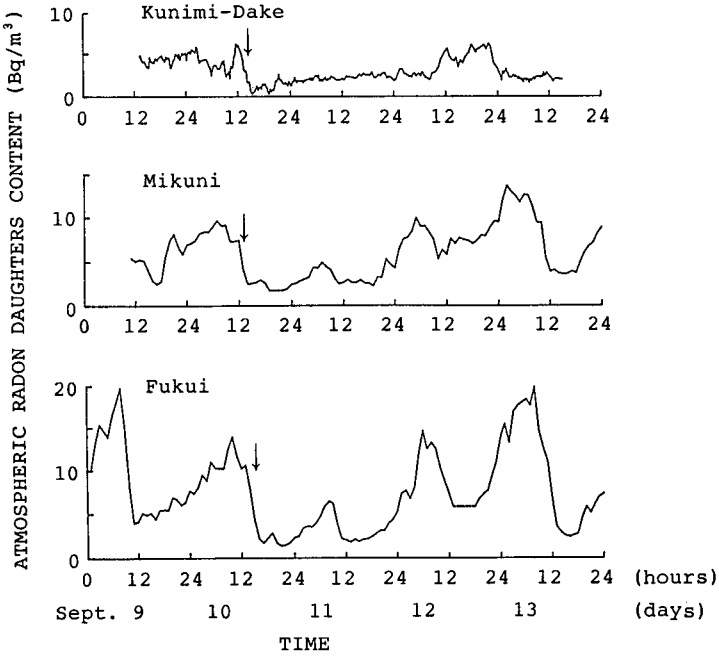

Fig.1 Time variations of the atmospheric radon daughters content observed at a mountain peak (Kunimi-Dake of 656 m height), a seacoast (Mikuni) and a center of nearby plain (Fukui).
(September 9-13, 1986)

diffuse into the upper layer. Then, when the wind blows from the nearby plain to Kunimi-Dake, somewhat high contents of the atmospheric radon daughters were observed there occasionally.

There were some cases that similar sudden decreases of the atmospheric radon daughters content were observed almost simultaneously at the three observatories as indicated by the arrows in Fig. 1. In this case, the changes of the temperature and the wind direction suggest that the cool air mass from the Sea of Japan moved into the area. It is concluded that such large scale meteorological phenomena as the passage of the boundary of the air masses of different qualities can vary the atmospheric radon daughters contents all over the area even if there are some differences of the geographical conditions.

It is said that, generally, the air mass movement and the passage of the air mass boundary were uncertain from a weather map. The atmospheric radon daughters contents are more reliable than the many kinds of the meteorological data for the indication of the passage of the air mass boundary.

Figure 2 shows an example of the time variation of the radon daughters content in precipitation at Kunimi-Dake. In this figure,

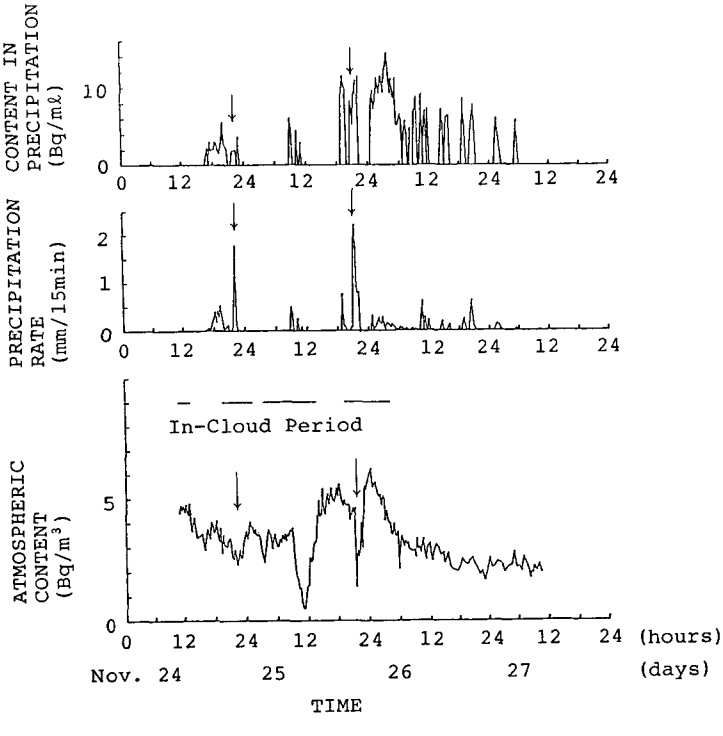

Fig.2 Time variations of the various factors measured at a mountain peak (Kunimi-Dake). The atmospheric radon daughters content, the precipitation rate and the radon daughters content in precipitation are shown. (November 24-27, 1986)

the precipitation rate and the content in atmosphere there are also shown for comparison. At present, no clear correlation was found between the atmospheric radon daughters content in the cloud and the radon daughters content in precipitation observed at Kunimi-Dake. From our data, it seems that the variations of the quantities are dependent on the characters of individual air masses. As shown by the arrows in Fig. 2, when the precipitation rate was high, the atmospheric radon daughters content was low. This phenomenon is considered as the result of the scavenging effect of heavy rain.

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THE SIZE DISTRIBUTION OF RADON DAUGHTER AEROSOL PARTICLES IN INDOOR AND OUTDOOR AIR AND THEIR DEPOSITION TO RESPIRATORY TRACT

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INTRODUCTION

Since the effective dose equivalent due to radon daughter products has estimated to be more than 30 percent of total one exposed to natural and artificial radiation in the UNSCEAR 1982 Report, the human lung dose to radon daughter products has attached special interest. For precisely estimating the lung dose, the size-distribution is needed because it gives precisely evaluation of the amount of radon daughter products deposited to the respiratory tract. It is therefore important to obtain the knowledge of many particle size-distributions of radon daughter products in various indoor and outdoor airs. In this work, measurement of the size-distribution of radon daughter products in indoor and outdoor air was carried out and the deposition of those to respiratory tract was calculated.

MEASURING METHOD

The diffusion battery method was used in sampling; The sampling device separates into two parts, a screen-type diffusion battery SDB and a filter holder. When the sampling air passes through the SDB, a part of radon daughter products in the air is settled on the surface of wire-screen and remains is sampled on a backup membrane filter. Alpha particles emitted from the surface of backup filter were counted with a ZnA(Ag) sintillation counter.

The SDB is constracted with a cylindrical tube and many pieses of wire-screen(165 mesh), which was made of stainless steel and mounted on a brass ring of 40 mm inner diameter and 3 mm thickness, one by one. Utmost 43 pieces of wire-screen ring can be put into the battery tube. In measurement, the five SDBs, in which adequate pieces of the wire-screen ring n is respectively set, were simultaneously used with the flow rate of 4 l/min (measured value was to be Ai). Another one in which is set a piece of wire-screen ring for settling unattached fraction, is prepared at this time (measured value; Ao). Measurement was carried out two times for obtaining a penetration curve (relation

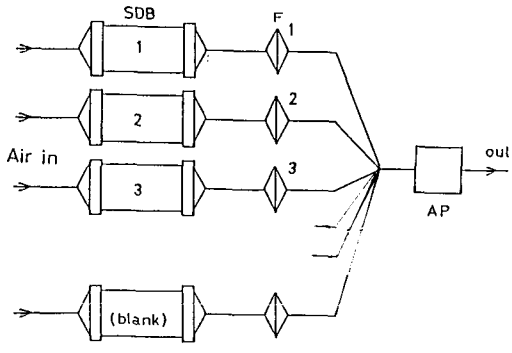

Fig.1. Schematic diagram of the experimental apparatus. SDB; Screen-type diffusion battery, F; Filter holder, AP; Air pump.

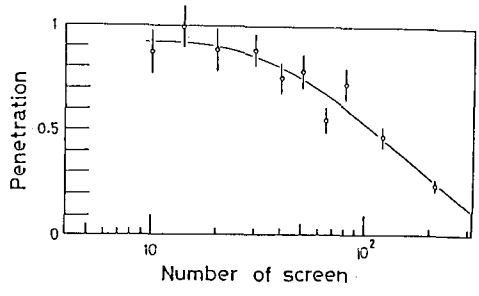

Fig.2. An example of the penetration curve.

between n and Ai/Ao, i=1 ~ 10) (see Figs. 1 and 2).

One series of measurement is planned as follows; Total measuring time (i.e. sum of sampling and counting times) for obtaining ten measurements is 2 hours; The first sampling time is 40 min, the waiting time from end of sampling to start of counting is 3 min, and the counting time 30 min. The second sampling is started under the same measurement condition of the first sampling within 10 min of the end of first sampling.

A computor program with the iterative method was applied to calculating the size-distribution from the penetratrion curve.

MEASUREMENTS

Measured sites Measurements were performed in a dwelling which is a tipical Japanese house, in two rooms in a building of Nagoya University and in outdoor air on the campus. One of the rooms was a meeting room and the other was a laboratory room.

Example of measurement Some examples of measurement are shown in Fig.3. These example are not always representative of the size-distribution of radon daughter products in each measured site and time but also rare one. We could rather think that similar results are often taken in various sites and time.

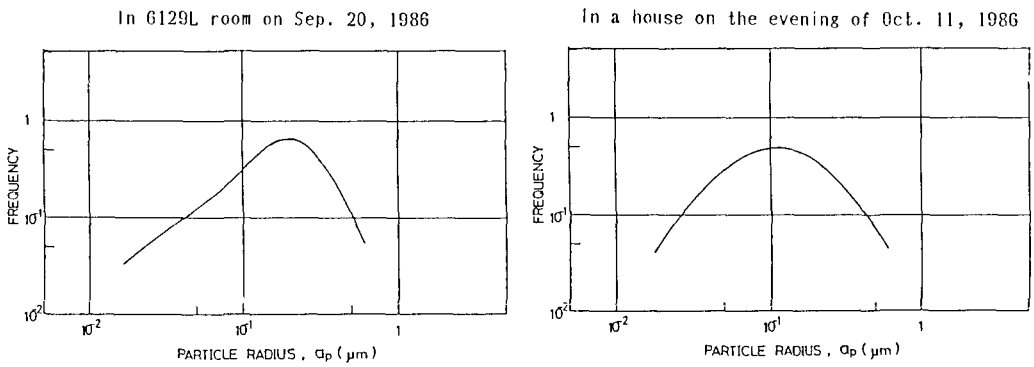

Fig.3. Examples of the size-distribution of radon daughter products.

DEPOSITION OF RADIOACTIVE PARTICLES TO RESPIRATORY TRACT

The radioactive aerosol particles are classified into two groups; One is the unattached atom (atomic size order) and the other is the attached atom in the range of 0.02 - 0.2μm radius.

In the calculation of deposition for radioactive aerosol particle to the respiratory tract, the Gormley and Kennedy formula was applied to the unattached atom. On the other hand, for the attached particle, the deposition which was calculated by Takahashi and Kawamura for monodisperse aerosol particles, by using a Yeh and Schum lung model modified, was used together with the observed size-distribution data. The parameters considered in the calculation were the tidal volume (1,000 and 1,500 cm^3), the rate of respiratory (7.5 and 15 min^{-1}), and the unattached fraction (0.1, 0.2 and 0.4) taking into account the human life-stile.

An example of deposition calculated on the left and right lobes is shown in Fig.4. The results are as follows; (1) Unattached atoms deposit entirely in the upper respiratory tract (before generation 17) and almost 80 % of attached atoms is expired and remains deposits mainly in the lower respiratory tract. (2) Deposition is more in lower lobe than upper lobe. (3) Difference of deposition among the obtained size-distribution is relatively small.

CONCLUSION

The present work is summerlized as follows: (1) The suitablly-measuring condition of the screen-type diffusion battery for obtaining

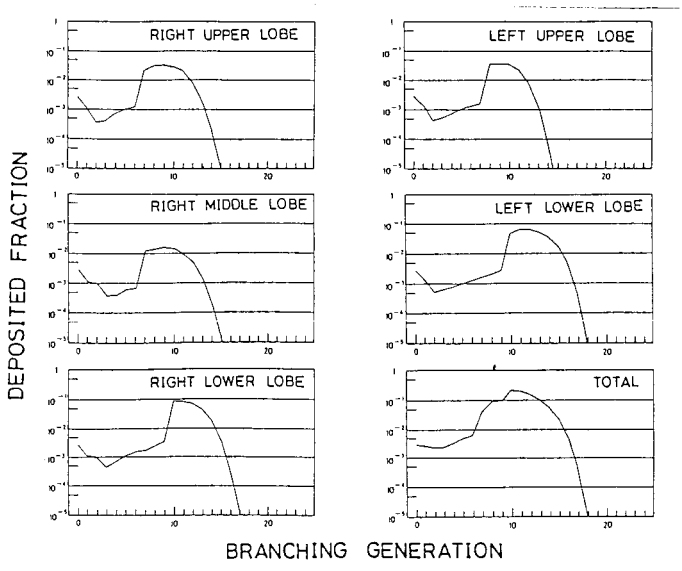

Fig.4. An example of deposition of radioactive aerosol particles to the respiratory tract.

the radioactive aerosol size-distribution were decided. (2)The measurements of the size-distribution of radon daughter products were carried out in outdoor and several indoor air; The aerosol particles were dispersed with log-normal mode and/or bi-modal distribution, and their radii were mainly in the range of 0.1 -0.2μm. (3) The deposition of radon daughter aerosol particles to the human respiratory tract was estimated to be 100 % for unattached atom and about 20 % for attached atom.

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DOSE ASSESSMENT OF MAINTENANCE WORKERS AND SIGNIFICANCE OF RADIUM-IMPREGNATED USED FILTERS IN THE PHOSPHATE FERTILIZER INDUSTRY

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ABSTRACT

Phosphate rock from Florida or the western United States is used for making fertilizer in three large plants in Alberta, Canada. With a uranium content of about 150 ppm the raw material presents some radiological problems in its processing, especially with the build-up of radium in filtration units and in the gypsum waste.

As part of a continuing investigation into the occupational hazards associated with the process surveys were conducted in the vicinity of the filtration units of the plants to assess the external doses received by maintenance workers when changing filters. The method involved shadowing the workers and measuring background exposure rates using a sensitive environmental monitor. The study showed that personnel were exposed to annual dose equivalent of from 0.2 to 0.5 mSv (20 to 50 mrem), the range relating to frequency of filter cloth change and how many shifts each change took. The results did not indicate the need for concern, although verification of the doses, by limited use of TLDs was recommended.

During use the filter cloths are impregnated with radium and part of the investigation focused on assessing the concentration. The results from all three plants showed close correlation, with specific activities ranging from 420-1220 Bq/g (11,400-33,000 pCi/g).

It was recommended that all used filters be handled with caution, and only in the damp state, and that they be disposed of within the gypsum tailings retention ponds on site, rather than in domestic landfill sites.

NEUTRALIZATION, CLUSTERING AND ATTACHMENT OF RADON PROGENY IN AIR

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INTRODUCTION

The decay products of radon (^{222}Rn) are formed in a unique process in which an inert gas becomes a positive ion of a metal, namely ^{218}Po. The diffusivity and electrical mobility of the progeny are important physical properties, and are obviously determined by the size and charge of the species in question. Until fairly recently (1), it was assumed that the diffusivity of radon progeny had a unique value. Simple evidence (1) that this is not so was the dependence of the penetration fraction of freshly formed ^{218}Po (less than 30 s old) on the ratio of the diffusion battery length, ℓ, to the volumetric flow rate, Q, for separate measurements under different relative humidity conditions. There was clearly an effect of humidity, although the measurements were made on different days and therefore with different concentrations of air pollution trace gases. Of great significance is the fact that the lines were curved, not straight as predicted by the penetration equation for diffusion batteries. Apparently the penetration factor is greater than predicted at large ℓ/Q; with increasing residence time of this very young ^{218}Po in the tube, the diffusivity decreases. The interpretation placed on this observation is that the effective mass of the freshly-formed ^{218}Po is changing during the measurement process. These effects required analysis of what processes may affect the newly-formed ^{218}Po positive ion.

Positive ions formed on decay of radon gas are potentially subject to a number of effects, specifically, neutralization, chemical reaction (in particular, oxidation), and attachment to aerosol particles. Neutralization may occur by one of two processes, combination with negative small ions, or stripping of electrons from atoms or molecules of lower ionization potential. The ionization potentials of Po and PoO_2 are ~10-10.3 eV so that certain cyclic or aromatic hydrocarbons (e.g. butyl benzene, 8.68 eV; naphthalene, 8.12 eV), NO (9.25 eV), and NO_2 (9.79 eV) have sufficiently low ionization potentials to serve as electron donors. Polluted air might be expected to be rich in electron donors for neutralization of ^{218}Po. The physical and chemical properties of ^{218}Po after formation are examined here in light of various experimental evidence.

MECHANISMS OF NEUTRALIZATION

An early test for the mechanism of neutralization was to determine the relative electrostatic collection efficiency of ^{218}Po in gases of different ionization potential. The result (Fig. 1) shows that environmental gases appear to act as electron donors in accordance with their ionization potential (1). At relative humidities (R.H.) greater than about 15%, it appeared

that the mechanism of neutralization might be due to combination with negative ions produced by the presence of radon (2). More sophisticated work on the effects of radon concentration and relative humidity on the neutralization rate constant of ^{218}Po lead to results such as shown in Fig. 2 (3). A theoretical model developed earlier (2) predicted K to depend on the 0.5 power of the radon concentration. The exponent found here (3) was b = 0.6 in the equation $K = aC_{Rn}^{b}$. The value of a increased with R.H. Another measure of neutralization is the fraction, f, of ^{218}Po having a positive charge at the end of the recoil path. Figure 3 (4) shows that f decreases with radon concentration according to the relationship $f = a'\exp(b'C_{Rn})$.

DIFFUSIVITY, MOBILITY AND NEUTRALIZATION CONSTANT

An experimental procedure was devised in which radon gas was introduced into a closed region between two parallel plates into one of which was embedded an alpha spectrometer solid state detector (2). The plate containing the detector could be held at either positive, negative or zero potential relative to the other plate. Under such circumstances, the count rate at the detector is the flux. When the detector is at zero potential, the flux is due to diffusion only; when the detector is at negative potential, the flux is due to diffusion and electrostatic attraction. Finally, when the detector is at positive potential, the flux is due to diffusion only, but the process of neutralization must also be considered, that is, the governing partial differential equation, equation (1), includes the term in K:

$$\frac{\partial C_{A+}}{\partial t} = f\lambda_{Rn}C_{Rn} - \lambda_{A}C_{A+} - KC_{A+} + D_{A+}\left(\frac{\partial^2 C_{A+}}{\partial x^2}\right) - BE\left(\frac{\partial C_{A+}}{\partial x}\right) \qquad (1)$$

where D = diffusivity; B = mobility; E = electric field strength; K = first order neutralization rate constant; λ = decay constant; t,x = time, x coordinate respectively; A+, A, Rn = $^{218}Po^{+}$, ^{218}Po and radon respectively.

Early experiments in which the data were fitted to the above equation (with appropriate boundary conditions) showed that the fraction born charged decreased with relative humidity (2). More recent work using refined experimental procedures including a pulse time of flight method (4) examined the effect of trace NO_2 on the neutralization rate constant (Fig. 4) (4). At a NO_2 concentration of 4.5 ppm in argon, K was found to be nearly independent of C_{Rn}, suggesting that charge transfer from trace NO_2 is the dominant process. At lower NO_2 concentrations, both charge transfer and negative ion combination appear to occur. In air (Fig. 5), K was found to be nearly independent of C_{Rn} and to increase slightly with R.H., suggesting that charge transfer is the dominant process (4).

ATTACHMENT OF RADON PROGENY TO AEROSOLS

In estimating attachment of radon progeny to aerosols, it has been demonstrated that large and unacceptable errors arise in calculating the unattached fraction of a polydisperse aerosol by characterizing the aerosol as monodisperse (5). Significant

errors in estimation of the attachment coefficient may arise if the kinetic theory of attachment is used at large particle sizes or the diffusion theory at small particle sizes (Fig. 6). Hybrid theories are to be preferred (Fig. 6).

With respect to the effect of charge on attachment, it has been estimated from a theoretical model that the charge would have a maximum enhancement effect on the attachment coefficient of about 25% (6).

CONCLUSIONS

The diffusivity, mobility and neutralization rate constant of unattached radon progeny depend on the nature of the gaseous environment. Neutralization appears to occur by two mechanisms, the dominance of which depends on the radon concentration.

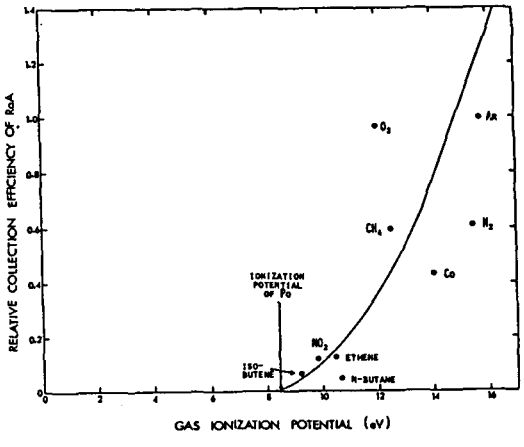

Fig. 1. The dependence of the relative electrostatic collection efficiency of newly formed RaA (^{218}Po) on the ionization potential of the gaseous environment (1).

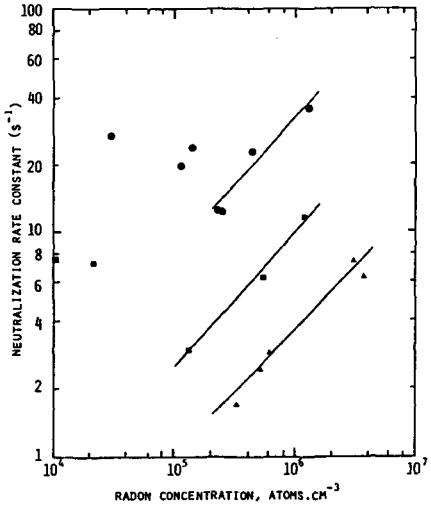

Fig. 2. Effect of radon concentration and relative humidity on neutralization rate constant of ^{218}Po in argon. Relative humidity: ▲ = 8-15%, ■ = 17-25%, ● = 95-100% (3).

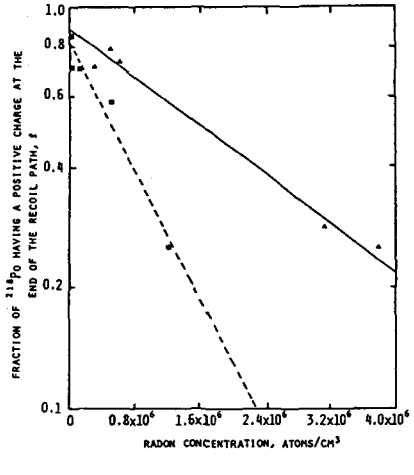

Fig. 3. Effect of radon concentration and relative humidity on the fraction of ^{218}Po having a positive charge at the end of the recoil path, f, in argon gas. Relative humidity: ▲ = 8-15%, ■ = 17-25% (4).

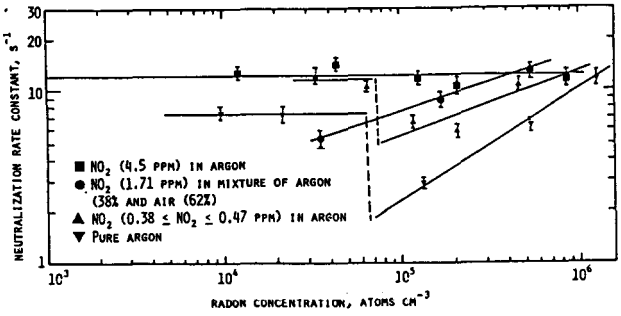

Fig. 4. Effect of radon concentration and NO_2 concentration on neutralization rate constant of ^{218}Po (4).

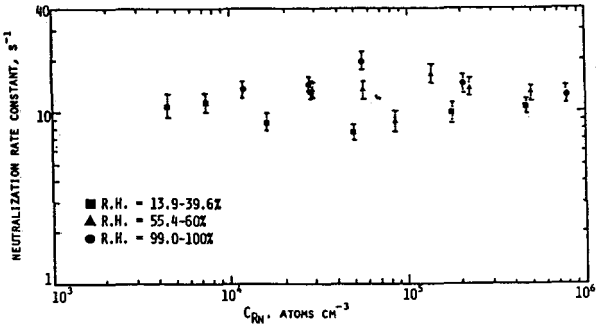

Fig. 5. Effect of radon concentration and relative humidity on neutralization rate constant of ^{218}Po in air (4).

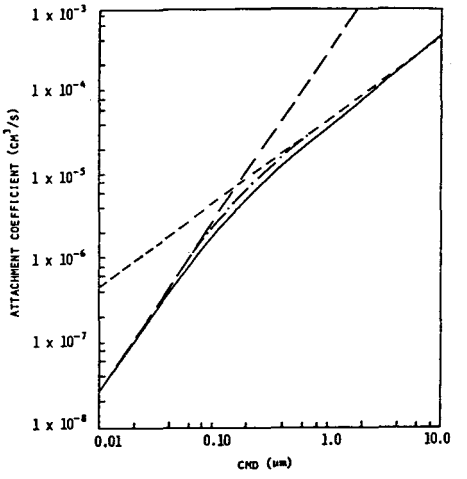

Fig. 6. Attachment coefficient vs count median diameter CMD (σ_g = 2).
----- diffusion theory
— — kinetic theory
—·—· kinetic-diffusion theory
——— hybrid theory

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THE IMPLICATION OF THE TIME-DEPENDENT DIFFUSION THEORY ON RADON-222 EXHALATION MEASUREMENTS

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By collecting the radon (Rn-222) emanating from a sample enclosed in an accumulation can, the mean exhalation rate (Bq s^{-1}) can easily be deduced by extracting gas from the can and analysing it for radon. This so-called closed-can method has been used in various forms since the beginning of this century. The radon exhalation rate from a sample free in air and in equilibrium with its surroundings (i.e. the radon concentration gradient inside the porous sample is constant in time) is referred to as the free exhalation rate of the sample. The natural aim of most exhalation measurement techniques is to determine this free exhalation rate. Unfortunately, the radon accumulating variant of the closed-can technique fails in this (except for samples that are very thick compared with the diffusion length), and what is really remarkable is that this failure can pass unnoticed, even under carefully controlled laboratory conditions. The objective of this paper is to illustrate, by means of theoretical calculations using diffusion theory and Fick's law, why it is so difficult to experimentally determine the free exhalation rate with closed-can methods. The temporal variation of the radon exhalation rate and the corresponding radon gas concentration in the sample enclosure (outer volume) for 'one-dimensional' samples will be presented. The conclusions drawn are, in principle, also valid for more realistic sample geometries, as long as the dimensions of the sample relative to the diffusion length are kept the same.

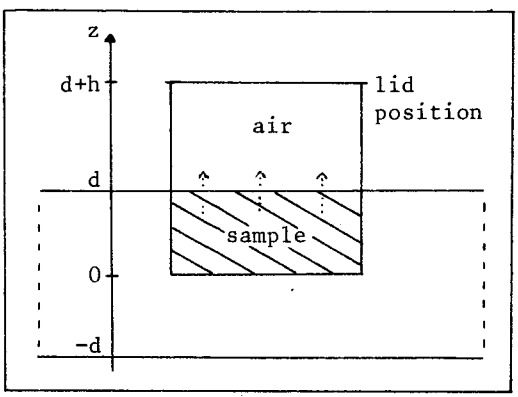

Fig. 1. A sample of thickness d placed in the bottom of an can is closed at time t=0, the enclosure exhibits the same areal exhalation as a semiinfinite slab of thickness 2d. The outer volume height with lid in place is denoted h.

In order to facilitate theoretical calculations, the simple sample geometry displayed in Fig. 1 is assumed. Furthermore, it is presupposed that before closing the sample is freely exhaling with an outer volume radon concentration equal to zero. In the absence of pressure and temperature gradients across the sample, the transport of radon from the sample pores into the outer volume is purely diffusive, i.e. is governed solely by a radon concentration gradient. If the can is closed at time t=0, the differential equation governing the pore concentration, C(z,t), of radon at time t and depth d-z in the sample is given by equation 1 (Sa84).

Eq. 1 $$p\,\partial C(z,t)/\partial t = p\lambda\{C_m - C(z,t)\} + D_e\,\partial^2 C(z,t)/\partial z^2$$

where D_e is the effective diffusion coefficient, p the sample porosity, C_m the maximum pore concentration in an infinite sample, and λ the decay constant of radon-222. C_m is related to the specific radium concentration, C_{Ra}, of the sample by the relation $pC_m = C_{Ra}\varepsilon\rho$, where ε is the emanation fraction (the fraction of the radon produced in the sample reaching the pore space), and ρ the bulk density of the sample. The initial condition is that C(z,t) is zero for all z>d (i.e. in the outer volume). The boundary conditions for the bottom and the open surface of the sample for t>0 are given by Eqs (2) and (3), respectively (instantaneous mixing of the air in the outer volume is assumed)

Eq. 2 $\quad \partial C(z,t)/\partial z = 0 \qquad$ for z=0 (bottom)

Eq. 3 $\quad -pD_e\, \partial C(z,t)/\partial z - h(\lambda+\nu)C(z,t) = h\, \partial C(z,t)/\partial t \qquad$ for z=d

where ν is the leakage rate constant of the can. Equation 3 is the mathematical expression for the balance between the exhalation according to Fick's law (first term) and the loss of activity (second term) due to decay and leakage. The relative leakage factor, μ, is equal to $(\lambda+\nu)/\lambda$ and the diffusion length, L, is defined as $L=(D_e/p\lambda)^{1/2}$. Analytical solutions to Equation 1 for $\mu=1$ (i.e. no gas leakage out of the can) have been published (Kr71, Sa84).

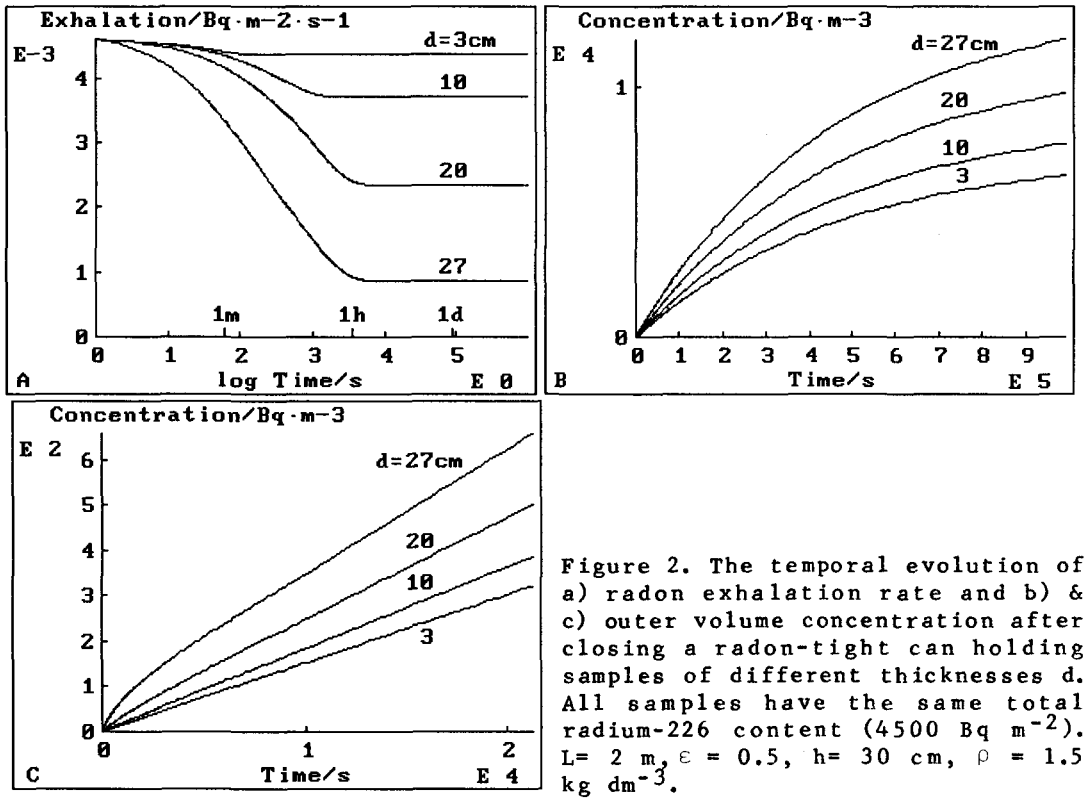

Figure 2. The temporal evolution of a) radon exhalation rate and b) & c) outer volume concentration after closing a radon-tight can holding samples of different thicknesses d. All samples have the same total radium-226 content (4500 Bq m^{-2}). L= 2 m, ε = 0.5, h= 30 cm, ρ = 1.5 kg dm^{-3}.

In this paper, numerical solutions for C(d,t) (i.e. the outer volume concentration as a function of time) and the corresponding exhalation rate (Bq m^{-2} s^{-1}) are presented in Figures 2-4. The radium-226 activity of all samples, irrespective of sample thickness, is normalized to 4500 Bq m^{-2}. The time scale of the exhalation part of the figures is

logarithmic in order to emphasize the initially rapid change in exhalation rate after closure. The concentration versus time curves are drawn on linear scales. All samples in figures 2-3 are thin compared with the diffusion length, L= 2m. Nearly all radon atoms available to the pore volume of the samples will therefore escape to the outer volume initially. In other words, all samples will exhibit about the same free exhalation rate. Let us, with this in mind, examine the radon-tight (μ=1) enclosure illustrated in figure 2. As soon as the lid is put on the increase in radon activity in the outer volume will lead to an increase in the radon concentration in the outermost pores, thereby decreasing the exhalation rate. Within a few hours a final exhalation rate is reached. Thereafter, the increase in radon concentration is uniform throughout the whole pore volume and outer volume, a state corresponding to an exhalation rate which is constant in time. The described behaviour is typical for all samples that are thin in comparison with the effective diffusion length L.

For d$\leqslant$20 cm the rapid initial change in exhalation rate is well hidden in the concentration curves even if the time scale is expanded (Fig. 2c). In the past, experimentalists have judged the initial part of activity growth as truely linear and corresponding to the free exhalation rate. This is in conflict with the results of the time-dependent diffusion theory predicting that the mean exhalation rate during the first few hours is much closer to the final steady-state exhalation rate than the free exhalation rate.

The influence of leakage is exemplified in Figure 3. The initial slope of all concentration curves is approximately the same, corresponding roughly to the final steady-state exhalation of the radon-tight can (μ=1). A few hours after closure the exhalation rate starts to increase again for all leaking cans since the increase in radon concentration in the outer volume is not fast enough due to radon atoms leaking out. This unexpected behaviour has been explained in detail elsewhere (Sa87).

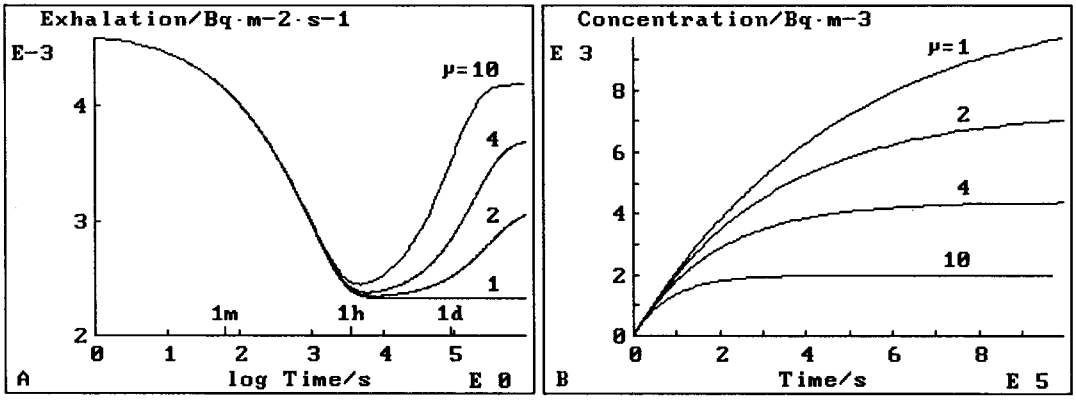

Figure 3. Exhalation and outer volume concentration for the 20 cm thick sample in Fig.2 but in leaking cans. The relative leakage $\mu = (\lambda+\nu)/\lambda$.

The diffusion theory predictions for samples with different diffusion lenghts are displayed in Figure 4. If the thickness of the sample is much larger than the diffusion length, the free exhalation rate prevails for a substantial time and thus can be easily measured with the closed-can technique. The prolonged period of free exhalation is due to the slow increase in radon concentration in the outer volume. The radon

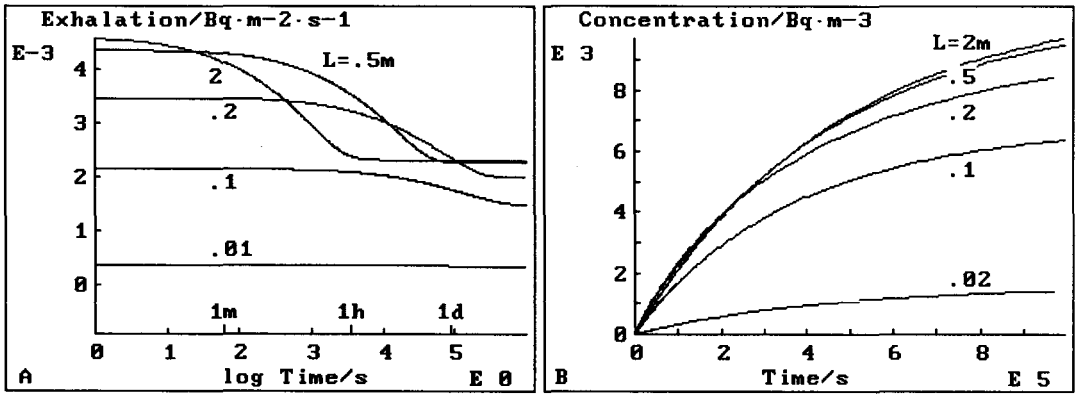

Fig. 4. The exhalation and concentration caused by the d= 20 cm sample in Fig. 2 but with different diffusion lengths L.

concentration in the pore volume has, in this case, enough time initially to 'follow' the concentration increase in the outer volume. The conclusion from Figure 4 is that the thicker the sample (relative to the diffusion length) the more the disturbance of free exhalation is delayed, and for very thick samples free exhalation prevails for such a long time that an experimental determination is feasible.

The results in this paper are based on time-dependent diffusion theory and the following main assumptions:

1) Fick's law is valid for porous samples.
2) Prior to time t=0 the sample is in equilibrium with a surrounding radon concentration equal to zero.
3) There is an instantaneous mixing of the gases in the outer volume.

None of these assumptions is neccessarily true to 100% in the practical use of the closed-can method. Despite these reservations it should be stressed that the exhalation school, which identifies the free exhalation rate with the mean exhalation rate during the first few hours after closure, has no support from the time-dependent diffusion theory, except for samples which are very thick compared with the diffusion length.

Acknowledgement
Financial support by the Swedish Natural Science Research Council is greatly appreciated.

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EFFECTS OF ATMOSPHERIC STABILITY ON RADON DISPERSION IN THE LOWER BOUNDARY LAYER

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INTRODUCTION

Inhalation of the short-lived decay products of radon (Rn-222) is presently considered (UNSCEAR, 1985) to account for a significant fraction of the radiological risk from natural sources of radiation. Extensive monitoring programs carried out during mining operations in Australia have shown that, under normal atmospheric conditions, airborne radon and radon daughter concentrations are very low (Auty et al., 1984; Marshman, 1983). However, measurements at Nabarlek in the Northern Territory showed that radon and radon daughter concentrations increased markedly during periods of atmospheric stability (Leach et al., 1982a).

An instrumented 20 metre tower has been erected over an extended uranium ore deposit at Yeelirrie (lat. 27° 10' S, long. 127° 55' E) in Western Australia, as part of a long term study to improve our understanding of the processes associated with radon dispersion, particularly under temperature inversion conditions and to develop models for this dispersion.

DATA ACQUISITION AND ANALYSIS

Mean temperature, wind speed and radon concentration were measured at a site near the centre of the ore body in an area characterized by low, scattered scrub. The meteorological variables were measured using commercially available sensors, while the radon sensors were designed and built at ARL. The meteorological sensors were sampled every 20 seconds and a running mean and variance accumulated for each quantity. The radon sensors were polled every 10 minutes. A data acquisition cycle length of 20 minutes was chosen as being long enough to include low frequency micrometorological changes in wind and temperature, but not so long as to exclude the effects of the major high frequency fluctuations on the measured means and variances.

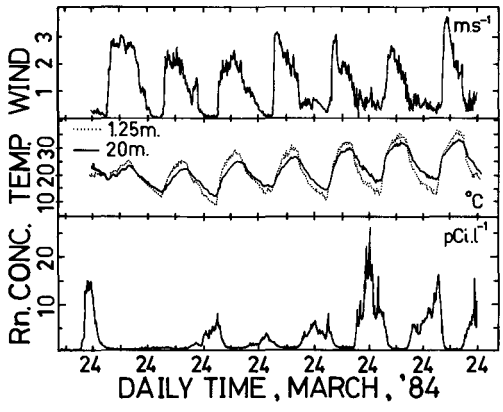
The raw meteorological data were combined with calibration results for each sensor to produce 20 minute average profiles of wind speed and temperature as a function of time. The raw radon data was analysed using the method of Stranden (1981), and converted to radon concentrations using the measured calibration factors.

RESULTS

At the tower site the radon emanation rate, averaged over 24 hours, as measured by the charcoal canister method (Countess 1977) was found to be 1.1±0.15 Bq/sq m/sec.

Figure 1 shows the radon concentration and horizontal wind speed at 1.25m and temperature at 1.25m and 20m as functions of time for a one week period in mid-autumn. During this period, temperature inversions accompanied by low horizontal wind speeds (near the ground) occurred every night. However, the nocturnal increase in radon concentration showed considerable variability from night to night.

FIGURE 1



The vertical radon flux was calculated using the formula

$$\text{flux} = -\left(D + \frac{ku_{*}z}{\phi_H(z/L)}\right)\frac{dC}{dz}$$

where z is height, L is the Monin-Obukhov length, which is a measure of the depth of the turbulent boundary layer; u_* is the friction velocity, which is related to the vertical turbulent momentum flux; C is the radon concentration at height z, D is the molecular diffusion constant for radon (1.2×10^{-5} sq m/sec), and $\phi_H(z/L)$ is a universal functional representation of the temperature gradient in the atmospheric boundary layer (Hanna et al., 1982). u_* and z/L were calculated by fitting the measured wind speed and temperature profiles to the universal functions for temperature gradient and vertical wind shear (Hanna et al., 1982) using a least squares method.

Figure 2 shows the variation of the calculated radon flux (normalized to the local radon emanation rate) as a function of atmospheric stability (characterized by the parameter z/L), for stable atmospheric conditions. A value of z/L = 0 corresponds to neutral stability. An increase in z/L corresponds to increasing stability of the atmosphere and suppression of vertical motion. The radon flux is close to the measured surface flux (emanation rate) at neutral

stability, and decreases systematically with increasing atmospheric stability. Also shown is the bulk Richardson number, defined by $B = (g.\Delta T.\Delta z)/(T.(\Delta U)^2)$, where g is the gravitational acceleration, T is the mean temperature in the height interval Δz, and ΔT and ΔU are the changes in temperature and wind speed across the height interval Δz. For stable conditions B and z/L are related by
$B = (z/L)/(1.0 - 5.0z/L)$.

FIGURE 2

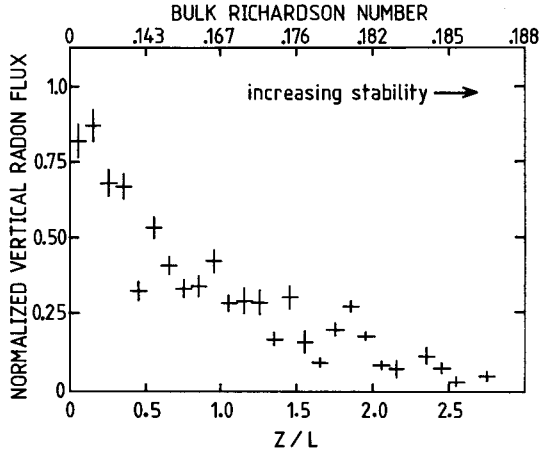


FIGURE 3

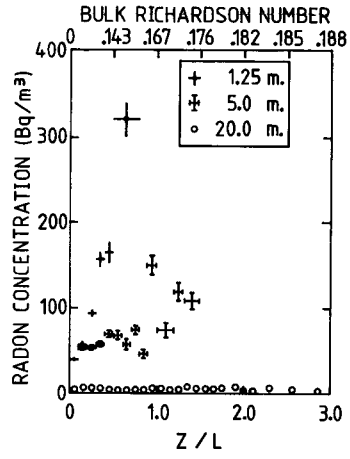


Figure 3 shows the variation of the radon concentrations at the three measurement heights as a function of z/L. The variation with stability decreases rapidly with height and is negligible at 20 metres.

DISCUSSION AND CONCLUSIONS

It is well established (Wallace et al, 1977) that, during the day when the vertical temperature gradient is negative (i.e. unstable conditions), the dominant mechanism for removing pollutants near the ground is eddy diffusion, while under temperature inversion (i.e. stable) conditions the vertical temperature gradient is positive and eddy diffusion is strongly suppressed. Consequently the wind speed near the ground should strongly influence pollutant concentration under inversion conditions. Binkowski (1983) has summarized the behaviour of the boundary layer during the day in terms of a turbulent mixed layer which is at least 0.5km deep, is capped by an inversion layer and maintained by convective heating from below. During the night this mixed layer collapses as the underlying heat source is removed, but there is still a shallow mixed layer present, even under conditions of extreme stability. This turbulent nocturnal layer is driven by wind shear from above. The transition from one state to the other seems to be very rapid. This is certainly consistent with the rapid build-up and decay of elevated radon levels observed during the current experiment.

The results presented here suggest that the variation of radon concentration with atmospheric stability is confined to the lowest 20 metres of the atmosphere. The radon concentration seems to vary rapidly across this layer, especially in stable conditions. This is consistent with the findings of other workers that the vertical diffusion of heat and momentum show their greatest variation near the ground. The results also show that the vertical radon flux in the lower boundary layer is equal to the measured emanation rate at the ground surface at the tower site for neutral atmospheric conditions and decreases with increasing atmospheric stability.

ACKNOWLEDGEMENTS

The authors would like to acknowledge the contributions of Drs K.H. Lokan, S.B. Solomon, and L.J. Martin, and Messrs. J.D. Osborne and F. Forgione of ARL, and Mr and Mrs R.A. Biggs at Yeelirrie, to the data acquisition phase of this project.

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PROBABILITY OF BACKGROUND RADIATION ENHANCEMENT ACCOMPANIED WITH RAIN

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INTRODUCTION

It is well known that the background radiation level often enhances when a rain starts. However, this is well known only on its qualitative nature. Quantitative information has been poor. In order to obtain clues for making a numerical model of this correlation, we have concentrated efforts on analysing the relationship between the radiation enhancement at onset of a rain and the timelength of pre-rain dry period.

INSTRUMENTATION AND DATA

Most rain measuremnts in routine monitorings have been done by conventional "seesaw" rainmeters whose minimum detectable precipitation is 0.5 mm, and most rain analyses have been done on hourly data basis. However,such low-sensitivity rainmeters cannot detect start of a rain unless it is a considerably heavy shower. In addition, hourly data cannot reveal detail time variation.

Accordingly, we initiated a continuous rain measurement by use of a "raindrop counting" rainmeter which features much higher sensitivity(the minimum detectable precipitation of 0.00426 mm or precipitation rate of 0.2556 mm/h.) The measurement has been done simultaneously with that of gamma radiation level in an open field of NIRS since 1985. The latter has been measured by a scintillation monitor with a cylindrical NaI(Tl) detector(2"ϕ). The rainmeter was set near the ground surface while the scintillation monitor was set at 1.5 m above the ground. These detectors were installed about 3 m apart horizontally. Rain data have been printed out digitally every 1 minute, and radiation data have been plotted on a dot-printer chart. Although NIRS accommodates various accelerators and RI facilities, data have been collected only in periods with no artificial contribution. Data analyses were done for periods below:Mar.2,1985-Mar.28,1985; May30,1985- Sep.9,1985; Sep.10,1985-Nov.6,1985; Nov.7,1985-Feb.3, 1986;Feb.4,1986-Apr.1,1986;May20,1986-Jun.26,1986;Sep.10,1986-Nov. 6,1986;Nov.7,1986-Feb.3,1987.

ANALYSIS

We wanted to know how a pre-rain dry period should continue so that the radiation enhancement is accompanied with onset of a rain. Therefore, we have measured the timelength of no-rain period between the onset of a rain and the endpoint of its

previous rain. If the enhancement was not accompanied with the onset,or if the enhancement was extremely indiscernible, such was excluded from our data. However, we had met a probelm that there was sometimes aftereffect of an ending rain because our rainmeter was highly sensitive to count even waterdrops falling from a funnel to which the waterdrops had been stuck after the rain had stopped. Such "pseudo" rains were found only among the rain data of 0.2556 mm/h(lowest recording level.) Therefore, we set a threshold of 0.5112 mm/h(second lowest recording level) to ignore all data below that level. In order to check sensitivity on the threshold, however, we have also analysed data corresponding to five other thresholds of 1.2780, 2.0, 5.0, 10.0 and 20.0 mm/h.

This "threshold method" has, however, the problem that there remains a possibility that small detached rains may be involved in the concerned no-rain period because "no rain" here depends on a given threshold. If there is such an internal rain, its onset may cause another radiation enhancement. In order to avoid such inconvenience, we set two conditions for a starting rain: (1) it should clear a given threshold, (2) it should be the first rain after the end of the previous one(ending rain of any magnitude.) However,internal rains below 0.5112 mm/h were ignored in any case. The radiation enhancement at onset of a rain will be referred as "event" hereafter. We got 115 events under the above conditions.

DISCUSSION

Fig.1 shows the ordering of the timelength of the no-rain period for six thresholds separately. It can be seen that every distribution is almost monotonous in which more events occur as the no-rain period becomes shorter. This figure implies that the enhancement is possible even just after the previous rain.

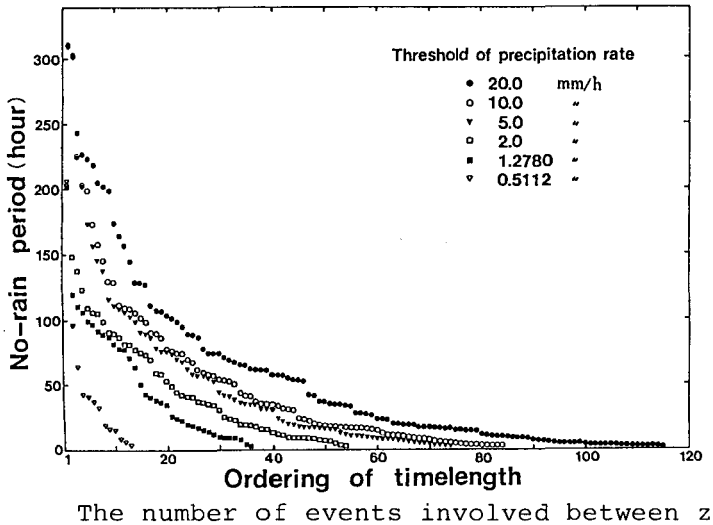

Fig.1 Occurrence of events in order of timelength of no-rain period. Seasonal difference is ignored, and data involve all seasons. Ordinate: timelength of no-rain period; Abscissa: ordering of the timelength.

The number of events involved between zero and an arbitrary time in this figure was divided by the number of all events which covered the whole period. Such a quotient can be considered to be

the "Probability of the event which occurs by the given time after the end of the previous rain." It is essentially a time-integrated probability, and the time in this case does not mean the no-rain period but means, in effect, the maximum waiting time for the event. As the denominator of the quatient does not involve rains with indiscernible events, though such are rare, the quotient may commit a slight overestimation.

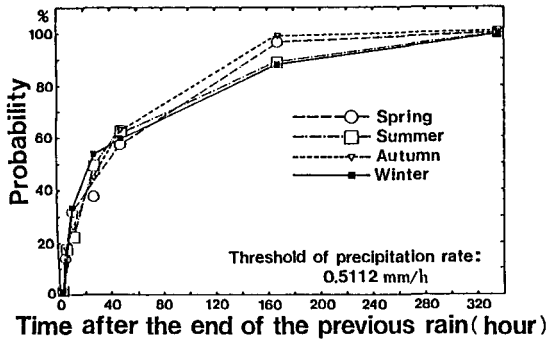

Fig.2 Seasonal dependence of the time-integrated probability for the threshold of 0.5112 mm/h. The probability of each season is plotted against the time after the end of the previous rain.

Fig.2 exemplifies how the time-integrated probability depends on seasons. As no significant difference can be seen among the four groups, this figure implies that we need not separate data according to seasons. Fig.3 demonstrates how the time-integrated probability is sensitive to a value of threshold. We can see that the probability does not depend on the threshold seriously though there is a systematic dependence on the time after the end of the previous rain. Therefore, we can consider that the time is the only significant variable among parameters considered.

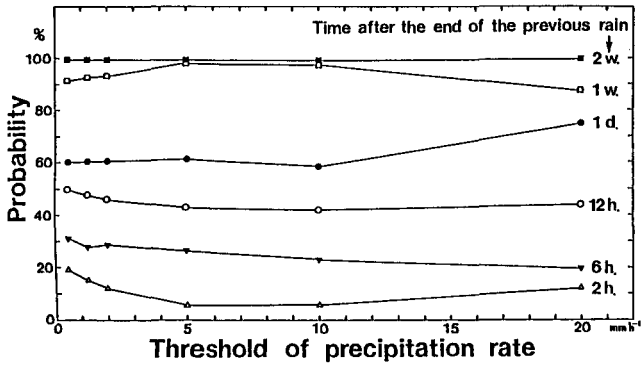

Fig.3 Sensitivity check of the time-integrated probability on the threshold. The probability is plotted for six time after the end of each previous rain separately. Seasonal difference is ignored, and data involve all seasons.

The number of events in all seasons was 115. However, owing to the feature of Fig.3, there will be no inconvenience if we combine data of the six thresholds. We could get 375 events by the combination, and we could smooth out the data effectively.

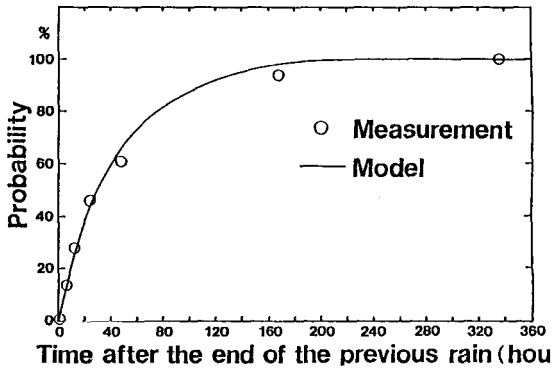

Fig.4 Comparison between the time-integrated probability derived from data(circles) and that of a model(solid line.) Ordinate: time-integrated probability of radiation enhancement accompanied with onset of a rain; Abscissa: time after the end of the previous rain.

Fig.4 shows the time-integrated probability derived from the 375 cases. The solid line represents a model expressed by a simple formula as

$$P = 1 - \exp(- kt) \qquad (1)$$

where P is the time-integrated probability, t is time after the end of the previous rain in unit of hour, and k is a constant. The k was estimated to be 0.023 by best-curve-fitting. It is apparent that the model well approximates the data.

The enhancement of the background radiation with a rain will be due to an increased deposition of natural radon daughters on the ground. When a rain occurs, it will scavenge more or less the airborne radioactivity leading to the radiation enhancement, only the degree of its prominence will depend on the amount of the deposition. Therefore, it will be natural to consider that the time-integrated amount of the deposition is in proportion to the formula(1). As the formula(1) is similar to the form of radiation strength which comes from a self-absorbing source, we can derive, by analogy, "effective half life" of the airborne radioactivity. The half life was calculated to be about 1.3 days.

Such a half life characterises the loss of radioactivity from air column above a local ground. It will depend on radon emanation from the ground, nuclear disintegration of radon and its daughters, vertical and horizontal air transport, and rain frequency. The half life will vary from one to another locality. More detail analysis will be done later.

CONCLUSION

The occurrence of radiation enhancement at onset of a rain is a function of timelength of no-rain period prior to the rain. It can be expressed well in terms of time-integrated probability as the formula(1). The seasonal dependence will be of no significance. The effective half life of natural airborne radioactivity in air column will be about 1.3 days though it may depend on locality.

SHIELDING EFFECT OF SNOW COVER ON INDOOR EXPOSURE DUE TO TERRESTRIAL GAMMA RADIATION

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1. INTRODUCTION

Many people in the world live in high latitude region where it snows frequently in winter. When snow covers the ground, it considerably reduces the external exposure from the radiation sources in the ground. Therefore, the evaluation of snow effect on exposure due to terrestrial gamma radiation is necessary to obtain the population dose as well as the absorbed dose in air in snowy regions. Especially the shielding effect on indoor exposure is essentially important in the assessment of population dose since most individuals spend a large portion of their time indoors. The snow effect, however, has been rather neglected or assumed to be the same both indoors and outdoors in the population dose calculation. Snow has been recognized only as a cause of temporal variation of outdoor exposure rate due firstly to radon daughters deposition with snow fall and secondly to the shielding effect of snow cover. This paper describes an approach to the evaluation of shielding effect of snow cover on exposure and introduces population dose calculation as numerical example for the people who live in wooden houses in Japan.

2. ASSUMPTIONS

Representative conditions were assumed as follows. Snow covers the smoothed flat interface of infinite half space of ground, where an isolated house stands. Snow has a variety of density from 0.1 g/cm^3 of freshly fallen snow to more than 0.4 g/cm^3 of wet snow. Two typical values of 0.2 and 0.4 g/cm^3 were chosen for light and heavy snow, respectively. Soil moisture content is assumed to be constant regardless the depth of snow cover. The terrestrial radiation sources of uranium and thorium series, and potassium are assumed to be uniformly distributed in the ground. The radon daughters floating in the air are neglected as they contribute little to the total dose from natural background[1]. In addition, the dose from the radon daughters deposited on the ground surface by precipitation in not taken into account since it is negligibly small as far as the annual dose is concerned. The house is assumed to be made of wood. Based on the indoor and outdoor exposure measurements in Japanese houses, the indoor exposure rates in wooden houses appear to be equal to outdoors[2]. It is consequently assumed that wooden structures have little shielding effect on the radiation penetrating from the outside and no dose contribution from the building materials. The representative building area is assumed to be 60 m^2. The effect of the area size is also considered by changing the parameter. Occupancy factor of 0.8 and 0.2 for indoors and outdoors, respectively, is used to evaluate the population dose.

3. CALCULATIONS

Indoor gamma exposure sources can be divided into three

components; (1) the building materials, (2) the soil under the house and (3) the soil outside the house. The first component is neglected in this paper, since it is assumed that building materials in wooden houses have no dose contribution to the indoor exposure rate. The other two are evaluated separately with a computer code based on the Adjoint Monte Carlo Method. The exposure resulting from the soil under the house can be estimated from the calculation of dose due to the radiation originating in a limited area. In this case it is not necessary to consider the shielding effect of snow even when snow covers the ground, since snow will not cover the area under the house. Whereas, the radiation emitted from the third source must penetrate the layer of snow to reach the house when the ground is covered by snow. The reduction of exposure by snow cover can be roughly estimated using the results[3] for the shielding effect of snow cover on the outdoor exposure. Further specification is necessary to obtain a better estimate of indoor exposure due to outdoor radiation. When there is a house, snow covers the ground only outside the house thus making a difference in the configuration of the snow cover. For this reason another series of shielding calculations was performed taking into account the presence of a house on the ground.

4. RESULTS AND DISCUSSION

Fig. 1 shows the dose contribution from radiation originating in the circular area of radius R m. It is shown as a fraction of the total exposure rate from the sources in the infinite area. The receptor is located 1 m above the center of the circle. The dotted line in the figure shows the solid angle at the receptor subtended by the area of radius R m. The exposure rate are lower than the subtended solid angle. The difference between the two lines reveals the fact that the dose contribution from a distance is larger than expected by the solid angle. Moreover, it shows the necessity of simulation calculation of exposure from the limited area to obtain dose from the source under the house. These results permit estimation of the dose contribution from the building area. When the building area is 60 m^2, the area provides 68% of the total exposure. This exposure fraction varies slowly with the size of the area. If this area decrease half or increase double, the fraction would change a little to 61 or 74%, respectively. It thus proves that the simplification in shape of the building area introduces no significant error. The residual of the fraction represents the dose contribution from the outside with no snow cover on the ground. The outdoor radiation contribution in the house is 32% for the building area of 60 m^2. This dose contribution is reduced further when snow covers the ground.

Fig. 2 shows the fractions of absorbed dose rates in air at 1 m above the ground surface as a function of snow cover. The solid line shows the shielding effect on outdoor exposure. Snow cover introduces an effective shielding of gamma radiation from ground sources. This calculation provides rough estimation of the exposure reduction due to snow cover. The simulation calculation considering the presence of a house shows that the exposure reduction inside the house is 10 to 30% larger than outdoors.

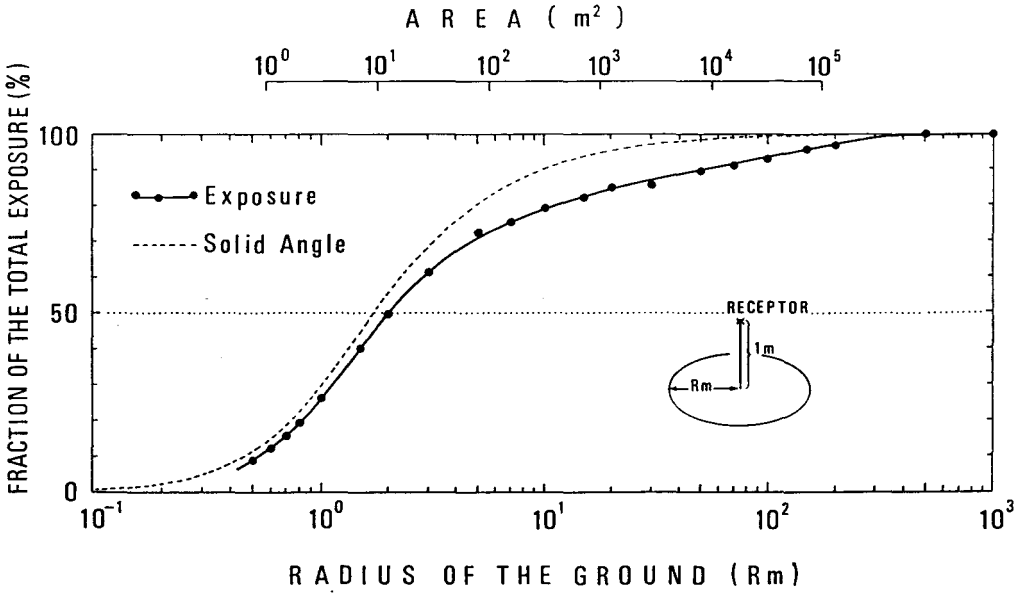

Fig. 1 Dose contribution from the circular area of radius R m to the receptor

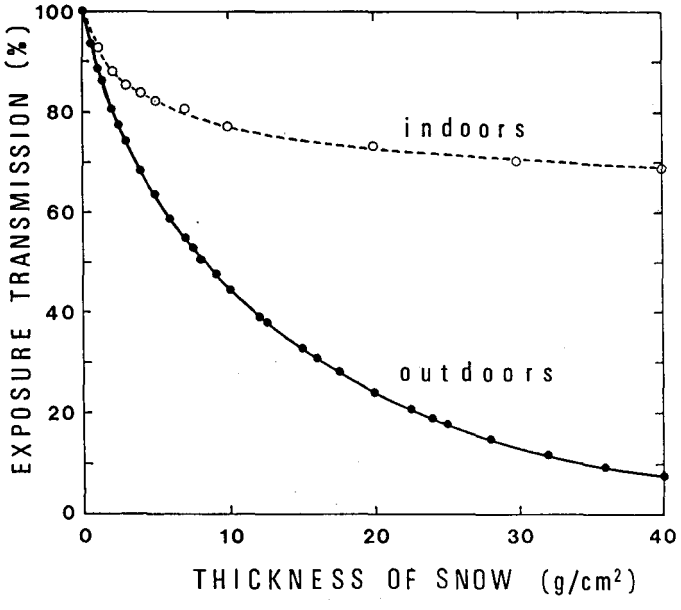

Fig. 2 Exposure transmission of terrestrial gamma ray
(**indoors:** Exposure transmission in the model house in this paper. **outdoors:** Exposure transmission in the open field)

The results for the representative house with a floor area of 60 m^2 is shown with the dotted line in Fig. 2 taking into account the outdoor dose contribution of 32% to the indoor exposure. Large shielding effect on outdoor exposure turns out to be small in the case of indoors due to the large contribution to the indoor exposure from the underlying soil. Fig.2 permits calculation of the reduction of exposure from the soil outside the house with snow cover on the ground surface. Verification of this calculation is not easy, since the ideal situation is hardly satisfied in actual

condition. However, the present evaluation by computer simulation is in relatively good agreement with the results obtained by actual measurements.

5. POPULATION DOSE ESTIMATION

Population dose is estimated considering the snow effect on the indoor exposure in wooden houses. The amount of snow precipitation has varied substantially from place to place and year by year. The published values of average depth of snow cover over past 30 years in each city were used to evaluate the snow effect on population dose. The depth of snow cover is classified into a category of 5 groups; less than 10, 10 to 20, 20 to 50, 50 to 100 and more than 100 cm. The average number of days in each group has been reported. Since no data is available about snow density, two densities are chosen to estimate the range of the effect of snow cover. The annual outdoor absorbed dose in air was found to be reduced down to 43 or 57%, respectively for the density of 0.4 or 0.2 g/cm^3 in the northern part of Japan. Whereas, the corresponding values for indoor absorbed dose in air are 79 and 83%, respectively. When the occupancy factors are taken into account to obtain the annual dose, the corresponding values are 71 and 78%.

In the case of heavy structures, one has to take into account the walls as attenuators of outdoor radiation and also as sources of radiation. Hence the evaluation of snow shielding effect on the indoor exposure in concrete houses becomes rather difficult. The effect depends on the place in the house. At a position near the window, larger shielding effect might be found, since about half of the radiation comes from the outside. On the other hand, the center of the house has little snow effect due to both the predominant dose contribution and the large shielding effect by the wall.

6. SUMMARY AND CONCLUSION

The simulation calculation has shown that snow cover causes large reduction in the outdoor exposure due to terrestrial gamma radiation, but that the degree of the reduction decreases in indoors. When the shielding value in the outside is applied to estimate the population dose, it should cause a significant underestimation. However, one cannot neglect the snow shielding effect on population dose. Neglecting snow effect will cause more than 20% of overestimation for the people who live in wooden houses in snowy regions. Therefore, it is necessary to employ the approach described in the present paper to obtain a better estimate of population dose in wooden houses.

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VERTICAL DISTRIBUTION OF NATURAL RADIOACTIVE NUCLIDES IN SOIL (DEPTH OF 0 TO 30 CM)

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INTRODUCTION

Knowledge of vertical distribution of natural radioactive nuclides in soil is very important in health physics. Works concerning the vertical distribution of the artificial nuclides in soil have been widely studied. Few studies were reported on the vertical distribution of a several natural radionuclides in soil and ,however, these distribution are considered to be important in investigation of their mobility in soil. The study reported in this paper presents data on the distribution of U-238, Ra-226, Pb-210, Ra-228 and K-40 in soil core samples.

SAMPLING AND MEASUREMENT

Soil core samples of 30 cm depth in length were taken on the ground in Fukui Pref.and Osaka Pref.,which is located in the central part of Japan, and the former is on the Japan Sea side and the latter is on the side of the Pacific Ocean. The area collected the samples is composed of various geologic properties. Core samples of soil were collected by using an iron sampler. Each core sample of 30 cm long was sliced into 5 cm section and dried. The part of the soil sample of larger than 10 mesh is excluded by sieving and 120 g of dried sample are packed in a disk-shaped plastic case.

The soil sample are examined in the concentration of U-238,Ra-226,Pb-210,Ra-228 and K-40 by gamma-ray spectrometry with a coaxial and a planer type of Ge detectors. To reduce the background counting rate, the detectors were surrounded with the shielding materials[1]. The activities of U-238 and Pb-210 are determined by using the planer type Ge detector from photopeakes due to 63.3 keV of Th-234(daughter nuclide of U-238) and its 46.5 keV, respectively. The activities of Ra-226, Ra-228 and K-40 are determined by using the coaxial type Ge detector from the photopeakes due to 609 keV of Bi-214, 911 of Ac-228 and 1.46 MeV,respectively. A reference sample by NBL(the New Brunswick Laboratory of the Atomic Energy Commission) is used as the standard for the activity determination.

The Fe, Mn and Ca contents in samples were determined by atomic absorption spectrometry according to Bruland et al.[2] and Terashima[3]. The soil sample of 1.0 g was decomposed with a mixture of HF and $HClO_4$. After having been converted to the chlorides by HCl, the molecular absorption interferences were suppressed by addition of strontium.

RESULT AND DISCUSSION

As shown in Fig. 1, no distinct vertical difference was found in the concentration of U-238, Ra-226 and Ra-228 in the soil samples except a single case described afterward. The concentrations of K-40 in the soil samples were found to decrease with increase in ignition loss of soil samples, as shown in Figs. 1 and 2. The slightly low concentrations of K-40 in the top layer were usually found compared with those in the deeper layers of soil. This may be caused by the fact that the alkali metal and those of the alkaline earths in the soil sample of the top layer easily carried away in solution by erosion.

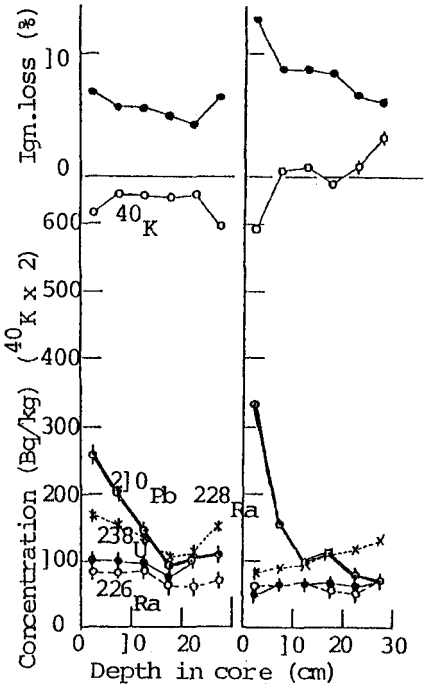

Fig.1 Concentrations of U-238, Ra-226, Pb-210, Ra-228 and K-40 and ignition loss as a function of depth in core for the first case.

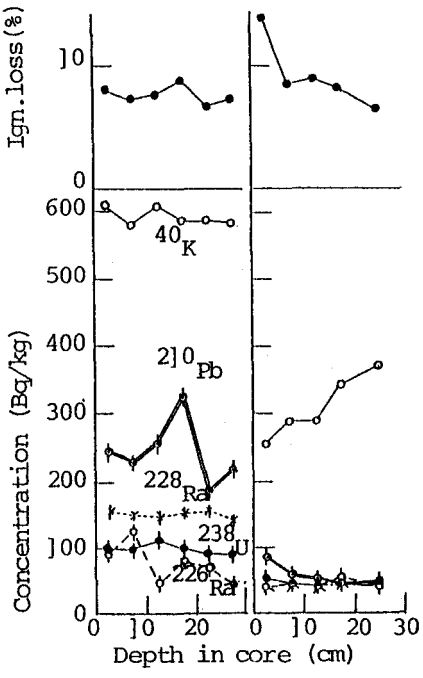

Fig.2 Concentrations of U-238, Ra-226, Pb-210, Ra-228 and K-40 and ignition loss as a function of depth in core for the third case.

The vertical variations of the Pb-210 contents in soil might be classified into four cases. For the first and the second cases of vertical distributions of Pb-210, the Pb-210 contents were the highest at the top layers(0-5 cm) and decrease with the depth. As shown in Fig. 1 for the first case, the concentration of Pb-210 in the 25-30 cm layers was established radioactive equilibrium with the concentrations of U-238 and Ra-226. For the second case,the Pb-210 content in the 25-30 cm layer was higher than the contents of U-238 and Ra-226 in the layers. The liquid phase contained Pb-210 permeates into the deeper part of 30 cm depth. For the third case,the Pb-210 contents were almost uniformly, as shown in Fig. 2, because of artificial mixing of soil layers. For these three cases described above a rather good relationship was found between the Pb-210 contents and the organic contents.

Fig.3 shows the vertical distribution of U-238, Ra-226, Pb-210, Ra-228 and K-40 on the the last case which is singular. The concentrations of U-238, Pb-210 and Ra-228 in the soil layer show the highest values in the 10-15 cm layer and the Mn and Fe contents are also high in the same layer. Fig. 4 shows the

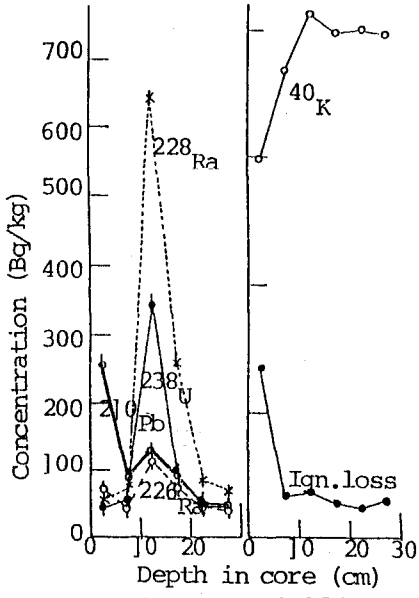

Fig. 3 Concentration of U-238, Ra-226, Pb-210, Ra-228 and K-40 and ignition loss as a function of depth in core for the last case.

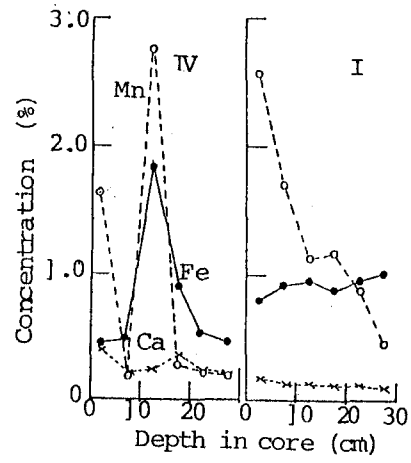

Fig. 4 Concentration of Fe, Mn and Ca as as a function of depth in core.

the variations of these contents on the last case and the first case which is the most general case. In the general case the Mn contents are the highest in the 0-5 cm layer and gradually decrease with the depth, and the Fe and Ca profiles in both cores show relative uniformity with depth. This is due to the fact that manganese dissolve in reduced layer, then slowly migrate and accumulates in the oxidized top layer. Iron forms insoluble compounds in the oxidized and the reduced layers. No appreciable change was found in the K contents. A possible explanation of relatively high concentration of U-238 and Ra-228 is as follows. The organic compounds in the top layer decompose and form humic and fulvic acids in the oxdized conditions. Metal elements are complexed with fluvic acid and transported downward in solution. And then, the metal complexes accumulate when the ratio of metal and fulvic acid attain to a constant value. Thus, uranium and thorium show relatively large accumulations in 10-15 cm deep.

The migration rate of Pb could be determined by evaluating vertical decrement with depth in the Pb-210 contents and its half-life for the first case described above, as shown in Fig. 5. The migration rate of Pb-210 were lead to the values of 0.2 in an usual area and 0.36 mm/yr in a moisture area.

Table 1 is shown the total accumulation of Pb-210 on the ground and yearly rain fall at Wakasa and Sakai,

which the former is on the Japan Sea side, near the Asian Continent and the latter is on the side of the Pacific Ocean. Then, the total accumulation at Japan Sea side is several times higher than that at the side of the Pacific Ocean due to influence of the Asian Continent.

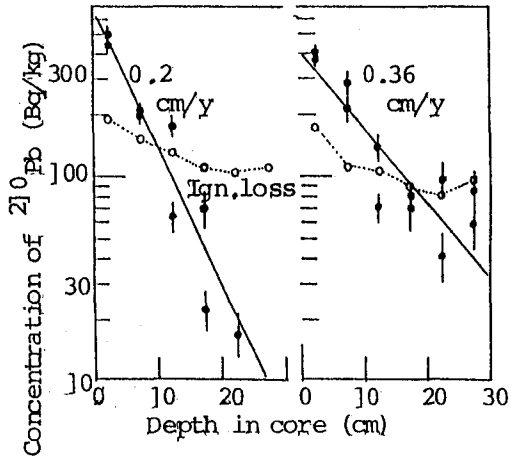

Fig.5 Migration rate of the surface soil layer collected in Wakasa.

Table 1 Total accumulation of Pb-210 and yearly rainfall in Waksa and Sakai

Location	Total accumulation of Pb-210 (kBq/km^2)	Yearly rain fall (mm/y)
Wakasa	20.2 21.3 29.1 21.4	2560
Sakai	2.6 2.3	1390

CONCLUSION

The vertical distribution of the Pb-210 contents in the surface soil layer might be classified into four cases. The migration rate of Pb-210 could be determined by evaluating vertical decrement with depth in the Pb-210 contents and were lead to values of 0.2 in an usual area and 0.36 mm/yr in a moisture area. The concentration of U-238, Pb-210 and Ra-228 in the soil layers showed the highest values in the 10-15 cm layer and the Mn and Fe contents were also high in the same layer and, however, this is singular.

ACKNOWLEDGMENT The authors wish to express their gratitude to Dr. S. Abe of National Institute of Radiological Sciences, Dr. Y. Nakashima of Nagoya University and T. Ishiyama of Radiation Center of Osaka Pref. for their useful advice. The authors are also indebted to Dr. K. Yamazaki, I. Urabe, K. Okamoto and T. Yoshimoto for useful discussions and collecting samples.

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BACKGROUND RADIATION IN THE SYDNEY METROPOLITAN AREA SCIENTIFIC REALITY AND PUBLIC PERCEPTION

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1. SUMMARY

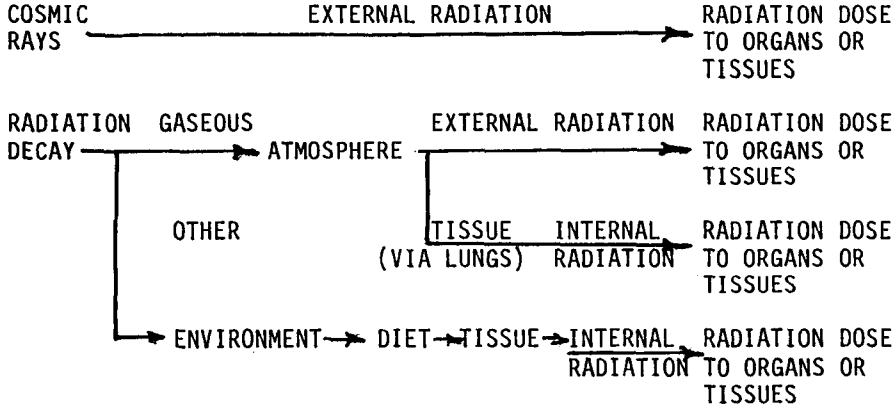
Very little information if any, is available to the public about the ubiquitous nature of low-level background radition -both from terrestrial and cosmic sources - in our highly urbanised or predominantly agrarnian societies, Public perception has naively accepted that the combustion of precious oxygen by air breathing engines in both "natural" and "societally acceptable". Little though has been given to the fact that on the one planet known to be able to sustain life based on oxygen, this is a recipe for disaster. One the other hand, since the evolution or creation of our universe, billions of tonnes of radioactive material within the crust of planet earth together with radiation from the thermonuclear powers station of the solar system, the sun, have sustained life, assisted in its evolution and, through photosynthesis and energy deposition, have ensured its propagation.

In this on-going student project, employing wide-band, sensitive scintillation radiation detectors, careful statistical analysis and an understanding of the physics of radum-uranium decay chain and the production of radon gas and its daughter products, fluctuation in the radiation fields of large cities is clearly demonstrated.

The work is usually carried out by the author and his final year thesis students and has great educational value for the public, civic authorities, school teachers, the media and political decision makers.

2. PATHWAYS OF EXPOSURE AND TYPICAL DOSES RECEIVED

Exposure to external or internal radiation from the above sources leades eventually to radiation doses to organs or tissues. The different pathways by which this occurs are summarised below:



The above information leads to the fact that not everyone will receive the same radiation dose. Also the proportion received from each source will differ from person to person and place to place.

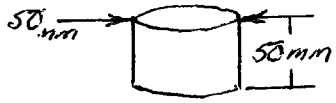
TABLE 1

ESTIMATED PER CAPITA ANNUAL EFFECTIVE DOSE EQUIVALENT FROM NATURAL SOURCES IN AREAS OF NORMAL BACKGROUND (UNSCEAR, 1982)

SOURCE	AIR DOSE (MSV) EXTERNAL RADIATION	INTERNAL RADIATION	TOTAL
COSMIC RAYS			
-IONISING COMPONENT	280	-	280
-NEUTRON COMPONENT	21	-	21
COSMAGENIC		180	300
-RADIONUCLIDES K40	120	6	6
Rb87			
TERRESTIAL/INTERNAL			
238 U SERIES			
238 -- 234	)	10)	
230 Th	)	7)	1044
226 Ra	) 90	7)	
224 Ra -- 214 Po	)	800)	
210 Pb -- 210 Po	)	130)	
232 Th SERIES			
232 Th	)	3)	
228 Ra -- 224 RA	) 140	13)	326
220 Ra -- 208 Tl	)	170)	
	---	---	---
(ROUNDED)	650	1340	2000

3. EQUIPMENT USED

The detector used was SPA-3, a scintillation detector. It incorporates a sodium iodide crystal of the dimensions shown below.



It is capable of detecting radiation in the range 100 keV to 1.5 MeV. The sensitivity of the detector is 1200 counts/minute/milli Roentegen/hr.

TABLE 2

COMPARISON OF SOME FEATURES OF THE ESP-2 AND THE EKCO N529A

		EKCO N529	ESP-2
SIZE	Length	53 cm	26.7 cm
	Height	29 cm	12.7 cm
	Width	24 cm	13.2 cm
WEIGHT		15 kg	1.73 kg
POWER SUPPLY		mains 110-120v	6 'C' cell batteries
		200-120v	(5.8-10.ov dc)

4. EXPERIMENTAL RESULTS

Background Radiation in Different Locations

LOCATION	DESCRIPTION	DATE/TIME	COUNT/5 MINUTES
MECHANICAL ENG. LABS. UNSW	On Concrete, by wall	7/10 8.15 am	60 400
COMMONWEALTH BANK MARTIN PLACE	On the ground floor	8/9 2.40 pm	59 900
MECHANICAL ENG. BUILDING, UNSW	Third floor, nr. wall	7/10 8.30 am	59 200
MARTIN PLACE	In the sun, breeze	8/9 2.45 pm	54 500
CENTRAL STATION	Country train area	8/9 1.40 pm	53 100
CORNER BOURKE ST O'RIORDAN ST & BOTANY	Industrial area	8/10 7.40 am	52 000
ABOVE GORDON BAY, COOGEE	On rocks, sunny, breeze	8/9 9.25 am	48 000
		9.30 am	48 400
		9.35 am	48 200
CORNER BOTANY ST 7 EPSON RD.	Industrial area	8/10 7.50 am	46 900
OPERA HOUSE	In the foyer	8/9 3.25 pm	46 600
MECHANICAL ENG. BUILDING, UNSW	On grass, nr. wall	7/10 8.00 am	45 800

UNSW LIBRARY	7th floor, nr. wall	20/9	2.00	pm	45 300
QUEEN VICTORIA BUILDING	On the second floor	8/9	2.15	pm	44 600
STARK ST, COOGEE POST OFFICE	On the soil	8/10	2.00		43 200
MARTIN PALCE		8/9	2.30	pm	42 000
MOSMAN WHARF	On a wooden seat	12/9	11.55	am	37 100
			12.05	pm	36 700
	On the ground		12.10	pm	38 700
COOGEE ESPLANADE	Sunny, breeze	8/9	10.10	am	30 000
BUS (373 ROUTE)	City	12/9	1.30	pm	35 100
	City	12/9	1.40	pm	36 000
	Moore Park to Church Street	12/9	1.55	pm	18 400
	Church Street to The Spot	12/9	1.55	pm	25 700
TRAIN	Around Lidcombe	25/9	10.00	am	20 700
COOGEE BEACH	Sunny, small breeze	8/9	9.10	am	17 300
			9.15	am	13 800
CENTENNIAL PARK	Cnr. Parkes Drive and Dickens Drive	15/9	1.25	pm	11 600
HARBOUR BRIDGE	Centre of Bridge	12/9	1.15	pm	6 070
FERRY (STEEL)	Opera House to Cremorne Point	12/9	11.30	am	3 180
	Cremorne Point to Mosman Wharf	12/9	11.40	am	6 390
FERRY (WOOD)	Mosman Wharf to Cremorne Point	12/9	12.25	pm	8 190
	Cremorne Point to	12/9	12.35	pm	3 740

5. SUMMARY

Twenty-fold differences in background may come as a surprise to all except the specalist in radiation technology. The fact that a ploughed field may have a background radom daughter emmission nearly as great as a well ventilated open cut uranium mine is of equal interest. Finally as has already been demonstrated, the probability that background radiation in the Sutherland Shire Council Chambers is greater than that in most buildigns of the Australian Nuclear Science and Technology Organisation may restore an atmosphere of informed realism and logic in the sustained attacks of the anti-nuclear lobby on a vital and highly disciplined industry which within two decades will make great contributions to sustain and make pleasureable life on plant earth. Nuclear free zones? Such a possibility was never meant to exist!

ACKNOWLEDGEMENTS

The author is greatful to Ms. N. Foote whose B.E. Thesis "Background Radiation in Sydney" was the first study in a forthcoming series and which contains some fifty references for which room cannot be found in such a short note.

RESULTS OF 30 YEARS OF ENVIRONMENTAL SURVEILLANCE IN ITALY

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ENEA-DISP, Rome - Italy

INTRODUCTION

The surveillance of the artificial radioactivity began in the middle fifties when the nuclear tests carried out in the atmosphere drew the attention of the radiation protection Community and the public concern on the problem.

Several laboratories distributed over the countries join together in carrying out measurements on various environmental and food components according to common procedures of sample collection and analysis of data.

The ENEA (formerly CNEN) National Committee for Nuclear Energy and Alternative Energy acts as coordinator with the task to promote the creation of new laboratories, to stimulate environmental studies and to collect, analyze and publish the data (1). Moreover the results of the measurements and the development of the surveillance activity are being discussed once a year in a general assembly.

The ENEA was also responsible to maintain liasons in the field of the environmental monitoring with the European Community according to the requirements of the EURATOM Treaty.

Since from the beginning the monitoring was set up as "person related" environmental monitoring with the aim to evaluate the global impact of all the radioactive sources on the popolation and "source related" monitoring around nuclear installations.

SURVEILLANCE PROGRAMME

The surveillance programme evolved during the years according to the development of the detection techniques and to the scientific interests connected with the widespread of radioactive contamination due to the fall-out of nuclear tests (evaluation of the fate of radionuclides in the environment vs climate characteristics, agriculture, use of the territory, etc.) and finally to the evolution of the dietary habits of the population.

In table 1 the temporal evolution of the programme is shown; it reflects the primary aim of the surveillance i.e. the health control.

TAB. 1
SURVEILLANCE PROGRAMM

	YEAR	DETECTION
AIR	1956	Gross beta
FALLOUT	1956	Gross beta
FALLOUT	1958	Gamma spectroscopy + Sr-90
MILK	1958	Gross beta
MILK	1960	Gamma spectroscopy + Sr-90
GEOGRAPHICAL WATER	1958	Gross beta
GEOGRAPHICAL WATER	1960	Gamma spectroscopy + Sr-90
VEGETABLES, CEREALS AND LEGUMES	1961	Gamma spectroscopy + Sr-90
ALIMENTARY PRODUCTS	1961	Gamma spectroscopy + Sr-90

Air was sampled and analyzed from the beginning considering its importance to detect large scale contamination. Some of the sampling points were located at high altitude being of particular interest acting as primary indicators of the radioactive contamination of the troposphere. The average trend of the gross ß activity concentration during the years is shown in fig. 1. The peaks which appear during the sixties are due to the nuclear explosions; the effect of the nuclear moratorium is visible after 1972. The small peaks in the following years are due to the French and Chinese explosions and the finally the "Chernobyl effect" is visible in the second quarter of 1986. γ-ray spectrometry, Sr-90 and Pu-299 measurements began to be performed after 1962 by some laboratories.

Fig.1 - Gross beta activity in air

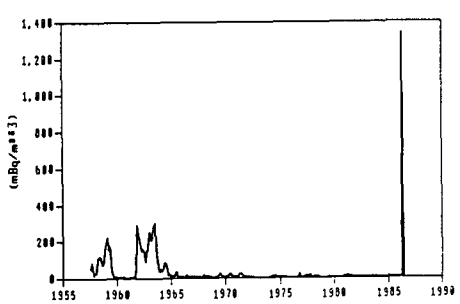
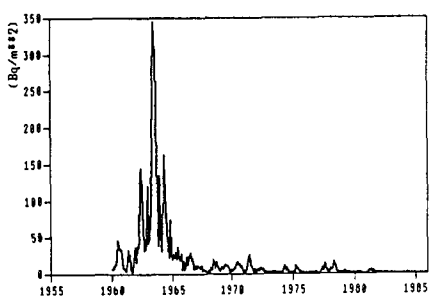


Fig.2 - Deposition of Cs-137 in Italy



Deposition was also collected throughout the country beginning from 1960. In fig. 2 the monthly deposition of Cs-137 averaged for the Italy, is shown. Sea waters and geographical waters have been also monitored. Also the studies carried out on the contamination of small lakes in the North Italy were of particular interest.

As far as the content of artificial radionuclides in human diet is concerned, different components have been analyzed during the years, some of which have been dismissed along the time as the activity of the fall-out decreased below the minimum detectable activity. However the analysis of milk samples, meat and industrially produced food was performed regularly during the years.

The radioactive content of Sr-90 in milk is shown in fig. 3 averaged for the Italy. The data for 1986 are not reported as following the Chernobyl accident the radioactive contamination was rather different on the territory and it was deemed incorrect to give a single average value for all the country. During the sixties bread samples were analysed regularly; moreover, taking into account the importance that this type of food has in the Italian diet, samples of "spaghetti" and wheat flour were also analyzed during the sixties. As an example in fig. 4 Cs-137 and Sr-90 detected in "spaghetti" during 1964 and 1986 are reported. It is worthwhile to underline the different ratios Sr-90/Cs-137 detected in 1964 and in 1986.

Fig. 3 - Sr-90 content in milk

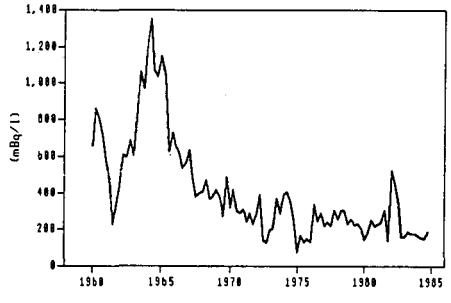
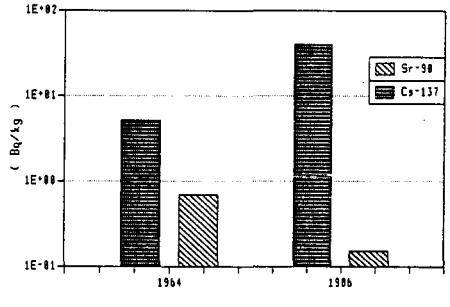


Fig. 4 - Radionuclides activity in spaghetti



Beginning from 1983 samples of complete diets of three large communities were also analysed to detect -emitters, tritium and transuranics. These analyses are of great interest after the Chernobyl accident as allow to evaluate the average increase of radioactivity in the total intake.

In the intake of the radioactivity.

Finally the results of the analysis of irrigation waters used for the rice cultivation in North Italy deserves attention as they represent hystorical set of data on the environment-water-cultivation-rice-man patway. Measurements of Cs-137 and Sr-90 are available, as well as the calcium content of these waters. It is possible therefore to express the results in terms of Sr-90/g Ca.

FINAL REMARKS

The importance of maintainingg active a monitoring network throughout the country was underlined by the Chernobyl accident. The presence on the territory of laboratories capable to carry out sampling and measurements according to standardized procedures is a garantee to obtain relable results. Moreover measurements carried out over the year allow the evaluation of the trend of artificial radioactivity in the environment and represent a suitable mean to control the impact on the population and on the environment of the radioactive artificial sources and to prove the adequacy of the radiation protection systems put into force.

Finally it has to be underlined that the control of the radioactivity in the environment as it has been carried out during the years in our country and in other countries, could serve as an example to set up similar monitoring systems for other conventional pollutants existing in the outdoor environment. Taking also into account that our experience showed that the economic cost of maintenance of the system just described was very limited.

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NATURAL RADIATION EXTERNAL EXPOSURES LEVELS IN CHILEAN SUB-ANTARTIC AND COUNTRY STATIONS

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Since 1983 gamma exposures levels, at 1,5 m above the soil are being investigated using TLD detectors. A network of 12 stations have been sta blished from ARICA (18°20' S latitude) in Northern Chile to YELCHO (64°52' S latitude) in Antartic territory. One year monitoring period was used at Antartic stations and a four months pe - riod in the country. The main subject of this study is to assess the average background radiation levels along the territory which is relevant to get a reference level and specially an esti mate of the average natural radiation contribution to the population dose in Southern Chile. Fig. 1 shows the TLD chilean network.

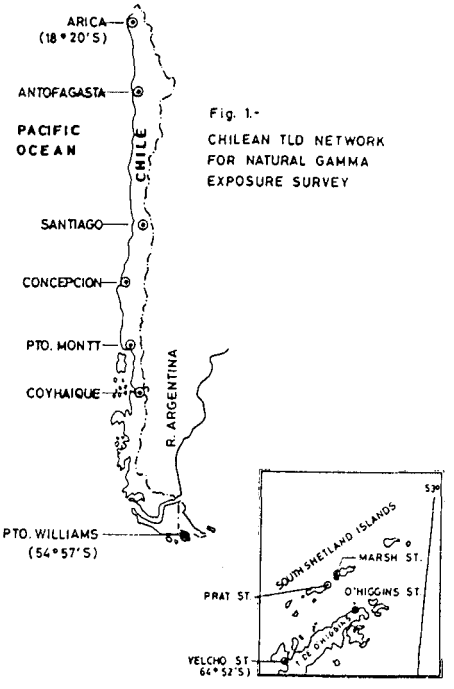

Fig. 1.- CHILEAN TLD NETWORK FOR NATURAL GAMMA EXPOSURE SURVEY

Method

The measurements were performed by two kind of TLD detectors : one energy compensated Ca SO_4: Dy Teflon dosemeter and one uncompensated Ca F_2: Dy, (TLD-200), the latter is required as a backup dosemeter and to assess any low energy gamma contamination. In this paper all our results are refered to the Ca SO_4: Dy detector. The dosemeters were calibrated at the CCHEN Secondary Standard Laboratory, with a Cs-137 standard source, with $\pm$ 4% of uncertainty at 93% confidence. The usual annealing and preheat procedures as well as the fading, transit dose and self irradiation corrections were described elsewhere (Stuardo, 1984). An 8300 Teledyne Reader and a 2000 D Harshaw Analyser were used for dosemeter readings.

Results and Discussion

Fig. 2 shows the exposure levels (1986-87) with bar indicating the total estimated uncertainty. The Antartic levels vary from $\sim$ 0,6 pC Kg^{-1} s^{-1} at MARSH and PRAT stations to $\sim$ 0,85 pC Kg^{-1} s^{-1} at O'HIGGINS and YELCHO sta - tions. The former stations are located in a volcanic origin soil (South

Shetland Islands) and the latest ones in a soil having intrusive rocks composed mainly by granodiorite (Alarcón, 1976). Direct TLD measurements in rocks of Yelcho station give a 0,95 pCi Kg^{-1} s^{-1} level which represents the upper exposure in the site and it is similar to the highest in the continental localities. The minimum exposure values, for the country stations, are found in Southern Chile ($\sim$ 0,7 pC Kg^{-1} s^{-1}) where the climate is rainy and the soil is mostly of volcanic origin. The maximum ($\sim$ 1,0 pC Kg^{-1} s^{-1}) corresponds to the northern localities with soil composed of stratified sediments.

Fig. 3 shows the annual average exposure levels (1983-87) along the territory. An increase of 40-60% is observed in 1985 to 1986 Antartic levels and similarly in some of country stations (30-40%). The energy uncompensated Ca F_2: Dy dosemeter did not detect any contribution of fresh radioactive contamination, then this variations can be attached to changes in stratospheric fallout plus some local changes in soil conditions, like: radon balance, ice and snow absorption thickness etc.

Table 1 presents the results of the annual mean gamma absorbed dose, for all the stations, as well as the total average dose $\pm$ 1 σ over a period of 3 to 4 years. A factor of 0.956 was applied to the corrected exposure to evaluate the gamma absorbed dose (Vold, 80). In the Antartic stations the average dose rate ranges between 0.46 mGy Y^{-1} (PRAT St.) and 0.72 mGy Y^{-1} (O'HIGGINS St.). These levels, correspond to 9% and 14% of the annual investigation limit for the public, respectively. In country stations the average dose rate fluctuates between 13% (PTO. MONTT) and 21% (SANTIAGO) of the limit.

For Southern Chile (Concepción to Pto. Williams), where most of people lives in wooden houses we consider an average dose rate of 0.74 mGy Y^{-1} which is equivalent, within the errors, to the 0.65 mSv Y^{-1} worldwide average normal dose equivalent, due to cosmic and terrestrial gamma radiation (UNSCEAR 82). In order to estimate the natural radiation contribution to the population dose, an average factor of 20% could be applied to outdoor levels of northern localities (Bouville, 85).

The annual fading factor of Ca SO_4: Dy Teflon detector at Sub-Antarctic Stations, during a four year survey, is shown in Table 2. We observe that 50% of values differ from usual results found in Laboratory controlled experiments (Piesch 81).

Acknowledgements

The author is grateful to the CHILEAN ANTARTIC INSTITUTE for financial and logistic support.

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Fig. 2:- NATURAL RADIATION EXTERNAL EXPOSURES LEVELS IN CHILE 1986-1987 (Compensated $CoSO_4$ Dy detector)

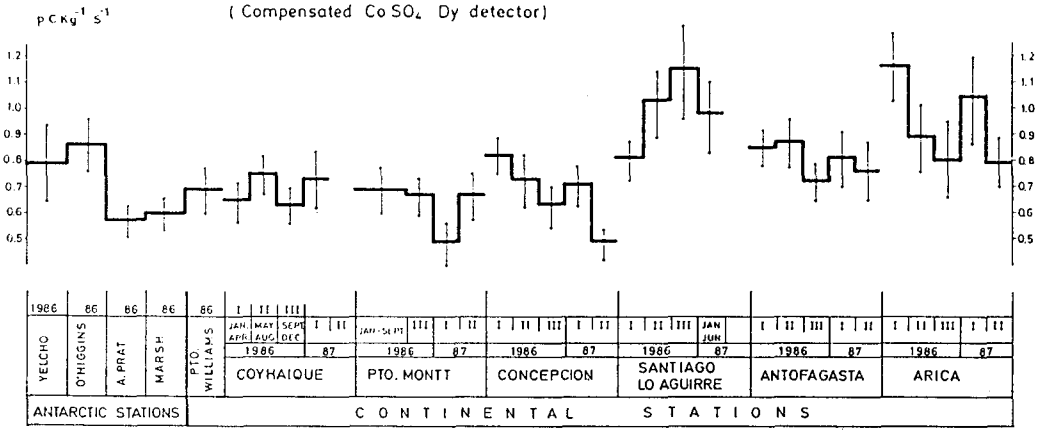

Fig. 3 NATURAL RADIATION EXPOSURES LEVELS IN CHILEAN ANTARCTIC AND CONTINENTAL STATIONS ANNUAL AVERAGE 1983-1986 (Compensated $CoSO_4$: Dy TLD detector)

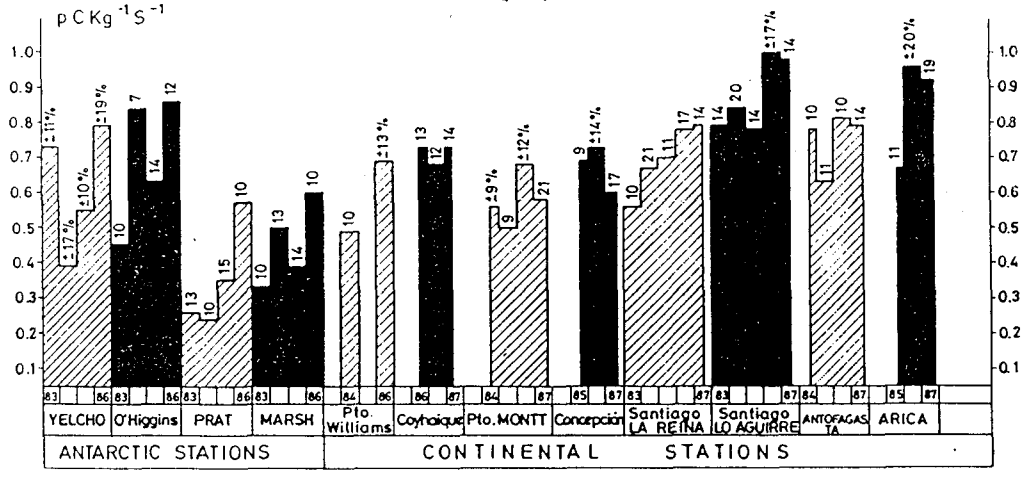

TABLE 1. NATURAL AVERAGE DOSE IN CHILEAN SUB-ANTARTIC AND CONTINENTAL STATIONS

STATION	ANNUAL mean dose ($mGy.y^{-1}$) 1983	1984	1985	1986	1987	Total average dose. $mGy.y^{-1} \pm 1\sigma$
YELCHO	0,85	0,45	0,63	0,92	---	0,71 ± 0,21
O'HIGGINS	0,53	0,60	0,75	1,00	---	0,72 ± 0,21
PRAT	0,31	0,41	0,46	0,66	---	0,46 ± 0,15
MARSH	0,37	0,57	0,50	0,70	---	0,54 ± 0,14
PTO.WILLIAMS	---	0,57	---	0,80	---	0,69 ± 0,16
COYHAIQUE	---	---	0,81 (Aug-Dec)	0,78	0,85 (Jan-Apr)	0,81 ± 0,04
PTO.MONTT	---	0,65 (Sep-Dec)	0,57	0,79	0,68 (Jan-Aug)	0,67 ± 0,09
CONCEPCION	---	---	0,77 (Aug-Dec)	0,85	0,71 (Jan-Aug)	0,78 ± 0,07
SANTIAGO LO AGUIRRE	0,92	1,10	0,93	1,16	1,14 (Jan-Jun)	1,05 ± 0,12
ANTOFAGASTA	---	0,91 (Sep-Dec)	0,71	0,95	0,91 (Jan-Aug)	0,87 ± 0,11
ARICA	---	---	0,64 (Aug-Dec)	1,12	1,10 (Jan-Aug)	0,95 ± 0,27

ANTARTIC

TABLE 2. CHILEAN SUB-ANTARTIC STATIONS
ANNUAL FADING IN $CaSO_4$: Dy, TEFLON TL DETECTOR

STATION / YEAR	MARSH	PRAT	O'HIGGINS	YELCHO
1983	(-1,6 °C) 1,36 < 163 >	(-2 °C) 1,05 < 283 >	(-3,1 °C) 1,06 < 568 >	(-1,2 °C) 1,05 < 389 >
1984	 1,31 < 414 >	(1,7 °C) 1,05 < 458 >	 1,21 < 244 >	 1,31
1985	 1,14	(-1,6 °C) 1,15 < 135 >	 1,14	 1,27
1986	1,25	1,26	1,15	---
Average ± 1S	1,27 ± 0,09	1,13 ± 0,13	1,14 ± 0,06	1,21 ± 0,14

( ) annual average temperature

< > annual snow precipitation in cm.

RESULTS FROM THE AUTOMATIC SWISS DOSE RATE MONITORING NETWORK NADAM

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INTRODUCTION

The swiss Network for Automatic Dose-rate Alarming and Monitoring, NADAM [1,2], is operated by the National Emergency Operations Centre. Since may 1987 the fully operational network covers the whole country with 51 stationary radiation monitoring equipements (Fig. 1). This network shall

- signal immediately a significant increase of the environmental exposure rate,
- enable the National Emergency Operations Centre to distinguish between the natural variation of the environmental radiation level and an artificial increase and
- give a rough survey of the environmental radiation levels over the whole country in a case of a nuclear accident with relase of airborne radioactivity.

The NADAM-stations are equipped with Geiger-Müller-counters and connected to the automatic weather-monitoring stations of the swiss meteorological network ANETZ. The local dose rate values and different meteorological parameters (e.g. air and soil temperature, windspeed and direction, amount of precipitation) are transmitted every 10 minutes through a fixed telecommunication network to a central data collecting computer where they undergo a qualitity and threshold check.

One of the mayor task of the National Emergency Operations Centre is to detect artificial increases of the environmental radiaiton level and to inform in such situations the emergency organisation [3]. Therefore it has to distinguish between increases due to natural variation and artifical reasons respectively. This needs a good knowledge of the local meteorological influences on the radiation level.

Since beginning of december 1987 the weekly mean values of the dose rate mesurements are regularly published and commented in the bulletin of the Federal Health Office [4].

MEASUREMENTS, RESULTS AND DISCUSSION

The following example shows the special influence on the hourly and daily variation of the dose rate by different meteorological parameters, i.e. air- and soil-temperature, humidity, precipitation due to the local situation.

Figure 2 shows the diurnal variation of dose rate at the NADAM-stations of Bern-Liebefeld (BER) and Payerne (PAY) together with the precipitation-values for a given period in march 1986. Shortly after midnight on 13th. there can be detected a sharp peak for the radiation values at both stations. A similar peak could also be found at other stations in the western part of switzerland. This increase could be related to precipitations in this region.

In the preceding and following nights we could find peaks too, but only in Bern-Liebefeld. They show a different shape with a steady increase from about 6pm to 8am followed by a steep decrease within about 2 hours. For this peaks there couldn't be found any relations to precipitation. The analysis of the meteorological datas and the weather-reportings of the gone weather (tab. 1) gives following results:

This increases are positively correlated to humidity (correlation-coeffizient r=0.74-0.94; Fig. 3) and to the temperature gradient between air- and soil-temperature (r=0.29-0.79) and negatively to the soil temperature (r=-0.62 to -0.90). They are also related to haze or fog respectively together with white frost.

We suppose that this special behavior of Bern in relation to Payerne is effected by the particular local situations and their influence on the microclimatic conditions: The NADAM-station of Bern is situated in a shallow depression surrounded either by small hills or buil-

dings; the Payerne station is on a lightly flat slope some 10 meters above the valleys ground. Due to this situation of Berne there can arise a shallow see of cold air with no vertical air movement and fog during cold nights (inversion-situations). Therefore it comes to a concentration of radon and its daughters in the lowest air layer and to wet depositions by dew or white frost during fog situations. In Payerne the cold air will flow downward and there are much less fog situations as table 1 shows.

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[3] Michaud B., Ramelet J., Schmid O., Imobersteg U.: "Improved emergency organisation for the protection of the public in case of nuclear accidents", this proceeding.

[4] Bulletin des Bundesamtes für Gesundheitswesen, Hallwag AG, Bern

observation date and time (UTC)		gone weather during the last 6 or 12 hours for BERNE	PAYERNE
10th.	0600	hazy	hazy
	1200	hazy	hazy
	1800	hazy	hazy
11th.	0600	hazy	hazy
	1200	hazy	hazy
	1800	hazy	hazy
12th.	0600	hazy, white frost	hazy
	1200	hazy, white frost	drizzle
	1800	hazy	no spec observ
13th.	0600	fog, white frost	hazy
	1200	fog, white frost	hazy
	1800	drizzle	hazy
14th.	0600	fog, snowfall	snowfall
	1200	fog	snowfall
	1800	hazy	no spec observ
15th.	0600	fog	hazy
	1200	no spec observ	hazy
	1800	drizzling	no spec observ
16th.	0600	fog, snowfall	no spec observ
	1200	fog	no spec observ
	1800	fog, snowfall	no spec observ
17th.	0600	snowfall	no spec observ
	1200	no spec observ	no spec observ
	1800	no spec observ	no spec observ
18th.	0600	no spec observ	no spec observ
	1200	no spec observ	no spec observ
	1800	no spec observ	no spec observ
19th.	0600	snowfall	snowfall
	1200	no spec observ	rain
	1800	fog, snowfall	rain-sonfall
20th.	0600	fog, snowfall	snowfall

Tab. 1: Weather-reporting for Bern-Liebefeld and Payerne for march 10th. to 19th. 1987. Observed weather-phenomenon during the last 6 or 12 hours respectively.

Fig 1:

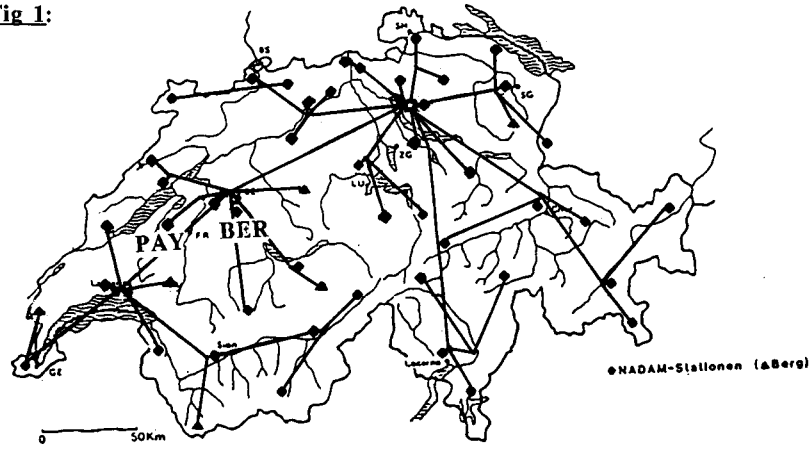


Fig 2: comparison of dose rate with different meteorolgical parameters

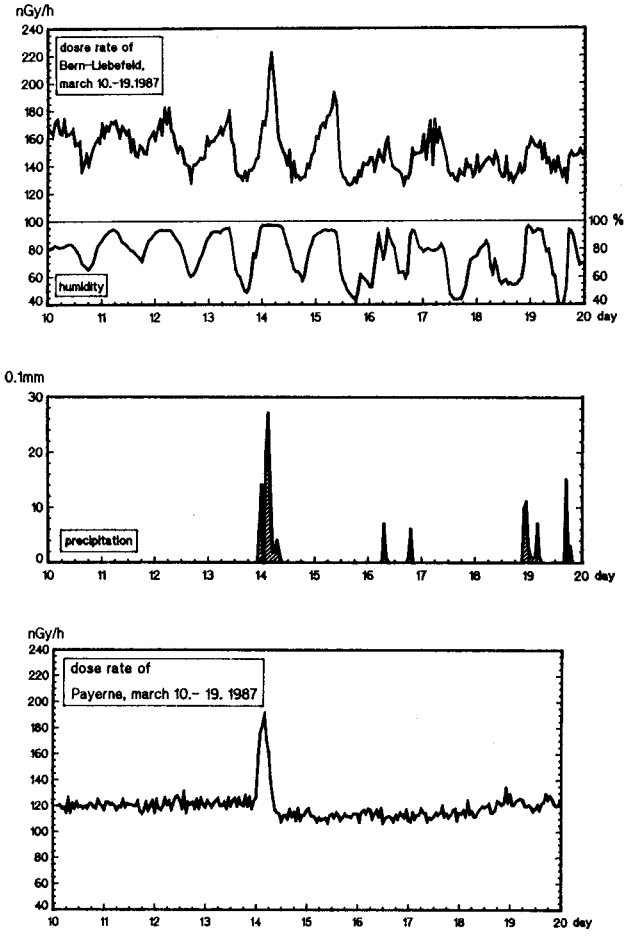
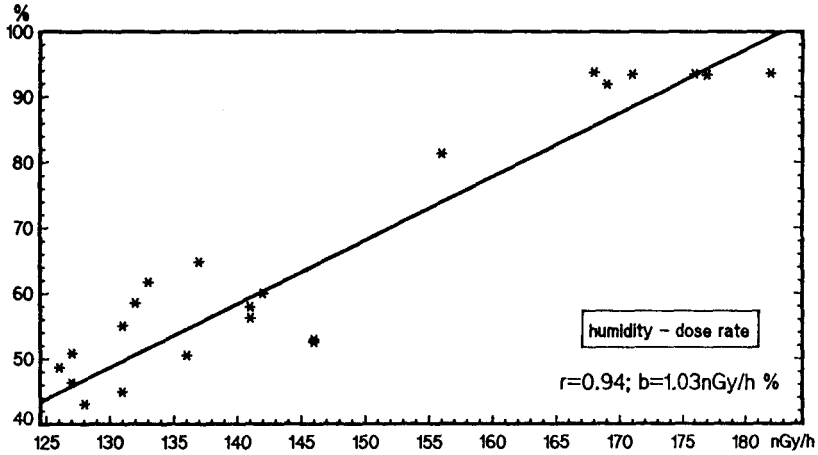
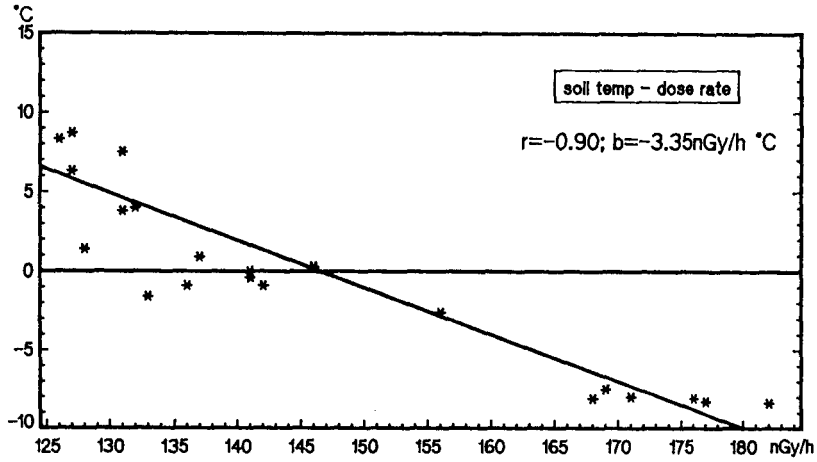
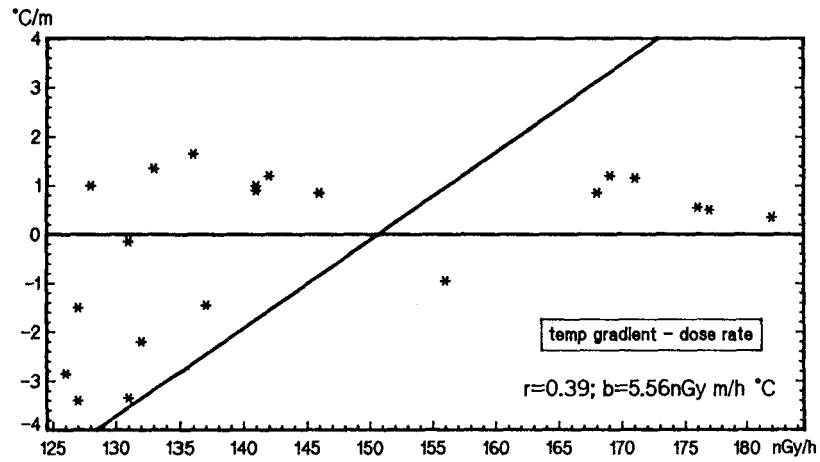


Fig 3: Correlations for Bern-Liebefeld, march 15th. 1987







MEASUREMENT OF ALPHA ACTIVITY CONCENTRATION IN THE GROUND AIR USING COLLULOSE NITRATE NUCLEAR TRACK DETECTORS

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ABSTRACT

A study on the measurement of alpha activity in the ground air has been carried out using CA 80-15 and LR 115-1 cellulose nitrate nuclear track detectors. The detection efficiencies of the detectors were determined by making use of an ^{241}Am alpha source of 0.1 uC in activity.

For field measurement of alpha activity of emanated radon and its progeny in the ground air, two different radon cups were installed for a certain period of time in two neighbouring gound holes of about 15 cm in diameter and 45 cm in depth. Of the two radon cups, one was kept closed space during the detecting period, while the other kept partly open space with a hole enabling the inner air to ventilate.

With the data evaluated in terms of alpha activity per unit volume of the air, detection efficiencies of the two different type of detectors and track recording characteristics in the two different cup circumstances were examined and discussed comparatively.

RADIOLOGICAL PROTECTION IN THE CEGB - THE CHANGING SCENE

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INTRODUCTION

The first two civil nuclear power stations in the U.K. were commissioned by the CEGB in 1962. There are now 12 operational stations (24 gas cooled reactors) and a further two Advanced Gas Cooled reactors being commissioned on another site. In the intervening years, improvements in designs and in operational practices backed up by sound health physics control and procedures, have led to a steady reduction in the mean annual dose to operators. During those years the statutory dose limits have only been exceeded on 4 occasions, and the maximum annual whole body dose equivalent was 57.4mSv (5.74 rem). Doses to members of the public have also been well controlled. Inevitably there have been minor incidents but on no occasion has it been necessary to declare an Emergency at any of our sites.

During the past decade a number of unconnected events have combined to put increased pressure on the nuclear establishments to effect ever greater improvements in radiological protection. ICRP 26 was a catalyst for a major re-think in this field. The accident at Three Mile Island and the subseqent disaster at Chernobyl focussed the attention and increased the concerns of the public and the media on the safety of nuclear power. A further event which was of relevance in the U.K. was the Public Inquiry held to consider the CEGB's application to build its first PWR. That Inquiry lasted over 2 years and guaranteed the spotlight remained on the subject.

Subsequent to the publication of ICRP 26, regulatory bodies re-considered their existing requirements and guidance. As a result the relevant European Economic Community Directive was revised. The U.K. nuclear regulations were subsequently re-vamped and published in 1985.

This paper briefly outlines changes which are being consolidated or introduced into radiological protection in the CEGB nuclear establishments.

OCCUPATIONAL EXPOSURE

In order to conform with the new U.K. nuclear Regulations, the CEGB Safety Rules (Radiological) were completely re-written and an extensive programme of re-training had to be implemented. ALARA formally became an integral

part of radiological protection. Procedures now require that should an individual reach a dose of 15mSv in a calender year an investigation shall be carried out to ascertain whether the control measures exercised during the time the exposures occurred were justifiable and in line with the ALARA principle. Further checks and inquiries must be made on any individuals reaching annual doses of up to 20,30 or 40mSv/y. In addition, if it is predicted that specific planned operations will lead to a collective dose in excess of 0.1 man Sv, an independant assessment of the planned radiological control measures must be undertaken. Particular care is necessary in controlling the doses to classified persons employed by contractors, and who carry out work for limited periods of time on a number of different sites within the U.K. Close collaboration between the employers and the Board is necessary in these circumstances. To assist in dose control a utility-wide computerised dose record/medical status system is being introduced to cover CEGB and Contractors employees. This record system and the dosimetry services used must meet the stringent requirements and gain the Approval of the Government's Health and Safety Executive.

EXPOSURE OF MEMBERS OF THE PUBLIC

All but one of the Boards nuclear sites are located on the coastline or on estuaries and in most instances doses resulting from discharges of low activity liquid discharges represent only a fraction of a percent of the ICRP dose limits. In these circumstances doses must be estimated, and increasing emphasis therefore has to be placed on the application of mathematical models using data from environmental samples taken for this purpose. Efforts are being made by the major relevant organisations in the U.K. to reach agreement on the models and parameter value data to be employed. Information on the estimated exposures of members of the public has been made publicly available for many years, but renewed efforts are being made to liaise on these matters with local authorities and organisations through Local Community Liaison Committees sponsored by individual site managements.

The Governments Authorising Departments continue to take and evaluate their own check samples from the environment. These confirm the validity of the Boards results. Nevertheless an increasing number of Local Authorities are carrying out their own independant surveys in order to inform them consituents. The Board collaborates with such bodies on these matters.

EMERGENCY PLANNING

In the period between the accident at Three Mile Island and the disaster at Chernobyl the CEGB introduced many significant changes into its emergency plans. It was therefore gratifying that after a preliminary consideration of the latter

event, H.M. Government felt able to state that the accident did not invalidate the principles on which emergency planning was based in the U.K. Detailed reviews of these plans have been undertaken by Government bodies and Agencies, but final recommendations are still awaited. Nevertheless it is clear that better and more rapid means of communication is needed between the Authorities and the Public.

In the overall response plan for any particular site, many separate organisations participate. Each such organisation acts in accordance with its own particular response plan. Action is being taken to ensure that the roles and responsibilities of these involved organisations are more clearly defined, that there is good communication between these bodies and that their efforts are co-ordinated.

The CEGB is taking the opportunity to harmonise the separate plans for its numerous sites. A "model" plan is well advanced in its preparation. A greater use of common procedures and terminology will lead to less confusion when staff from one site are required to assist those at another (affected) site.

Considerable debate has taken place on the scale of accident for which detailed pre-planning for emergency response should be developed. There may yet be some development in this position but there is increasing emphasis being placed on the ability to demonstrate that if necessary, emergency response actions can be extended out to any required distance.

RADIOLOGICAL PROTECTION - FUTURE DESIGNS

When a new station is to be built, the CEGB issue Design Safety Criteria (DSC) relevant to the project. These provide guidance for the designers. The criteria reflect the judgement at that time of the desirable and achievable standards of radiological protection and also incorporate judgement factors intended to anticipate possible changes in standards in future years. These DSC's are expanded and more fully interpreted in a companion document known as Design Safety Guidelines (DSG's). In addition to setting "targets" for doses to workers and to members of the public a criterion is also included for the annual collective dose. In 1984, a Public Inquiry began into the CEGB's application to build a PWR. During the course of that Inquiry many searching questions were directed at the criteria but from his report, it is clear that the Inspector was satisfied with the position.

In consideration of future designs, greater emphasis will be placed on formal cost benefit or other appropriate analysis to assist in determing whether design proposals are ALARA. It is hoped that in the not-too-distance future there will be agreement between industry and the regulatory bodies on the value of a man-sievert to be used in such considerations.

Currently it is particularly difficult to decide on future dose criteria. The ICRP has announced in its COMO Statement that they are reviewing all the data on risk estimates and consequently new recommendations can be anticipated, possibly in 1990. The CEGB has declared its intention of proceeding with a programme of PWR's. Application for one plant has already been made and this may be followd by 2 or 3 others at perhaps yearly intervals.Difficult decisions must therefore be taken on the radiological criteria to be issued for these plants as it is probable that some form of inquiry - perhaps limited - will be held for each one.

In a recent publication, the National Radiological Protection Board (which advises the Government on these matters) recently issued interim guidance that the revised Japanese bomb dose data was likely to lead to an increase in the radiation risk factor by a factor of 2 or 3. Nuclear operators are therefore urged to begin to work towards a "target" dose limit of 15mSv/y for workers and 0.5mSv/y for members of the public. This advice must be borne in mind also, in formulating design criteria.

THE FUTURE

The CEGB intends that nuclear power will play a significant role in the future production of power. The provision of an efficient radiological protection service is therefore paramount. There will be challenges in the coming years which will place demands on all practitioners in this field. From my own point of view, I look back at our past achievements with some satisfaction and am certain that ways of solving new difficulties will be found.

NOUVELLE LEGISLATION DANS LE DOMAINE DE LA RADIOPROTECTION

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ABSTRACT

A Bill on radiation protection will soon be discussed in the Federal Chambers (Swiss Parliament). This contribution shows the structure of the new law, the fundamental aspects and those points which were most controversial during the elaboration of the project . In particular, the following topics will be discussed: The principles of the radiation protection, the individual monitoring, the medical applications, the radioactive wastes, the civil liability and the prescription.

1. INTRODUCTION

Sur le plan légal, la radioprotection est actuellement réglementée d'une manière quelque peu rudimentaire puisqu'elle n'est régie que par deux articles de la loi de 1959 sur l'énergie atomique. De nombreuses dispositions en la matière ont été édictées par voie d'ordonnances. La nouvelle loi permettra pour la première fois de réglementer dans une loi le domaine de la radioprotection dans son ensemble.

2. STRUCTURE DU PROJET DE LOI

Le projet de loi est divisé en 7 chapitres. Le chapitre 1 décrit le but de la loi et son champ d'application. Le chapitre 2 est consacré à la protection de l'homme et de l'environnement. Il contient également des dispositions générales relatives aux déchets radioactifs. Le chapitre 3 régit les autorisations et la surveillance. Les dispositions régissant les domaines de la responsabilité civile et de l'assurance, de la protection juridique, des émoluments et des sanctions font l'objet des chapitres 4 à 6. Les dispositions finales sont au chapitre 7.

3. DISPOSITIONS FONDAMENTALES DU PROJET DE LOI

3.1. But et champ d'application

La présente loi a pour but de protéger l'homme et l'environnement contre les dangers dus aux rayonnements ionisants. Elle s'applique à toutes les activités, installations et événements qui peuvent présenter un danger lié à des rayonnements ionisants. Ces activités ne doivent être confiées qu'à des personnes techniquement qualifiées.

3.2. Principes de la radioprotection

Le système de limitation de l'exposition aux radiations de la

Commission internationale de protection radiologique (CIPR), basé sur les trois principes de justification, d'optimisation (ALARA) et de limites de dose individuelles, constitue le fondement sur lequel s'articule le projet de loi.

Le principe de l'optimisation a été traduit dans la loi de la manière suivante: "Pour réduire l'exposition aux radiations de chaque individu ainsi que de l'ensemble des personnes concernées, il y a lieu de prendre toutes les mesures commandées par l'expérience et l'état de la science et de la technique, et adaptées aux circonstances."

Le projet de loi prévoit la fixation de limites de dose pour les personnes qui par leur profession ou d'autres circonstances sont exposées à une irradiation accrue par rapport au reste de la population et contrôlable. Afin de permettre une certaine souplesse et de ne pas encombrer la loi de dispositions techniques complexes, les valeurs limites ne sont pas fixées dans la loi mais dans l'ordonnance d'application.

Une situation d'accident doit être jugée autrement qu'une situation dans laquelle la source de radiations est sous contrôle. Aussi la loi prévoit-elle de donner au Gouvernement (Conseil fédéral) la compétence de fixer les doses de radiations acceptables dans des situations de danger dû à une augmentation de la radioactivité.

3.3. Dosimétrie

Afin d'avoir la possibilité d'ordonner une dosimétrie différenciée en fonction des conditions d'exposition professionnelle, le projet de loi fixe comme principe que la dose de radiations doit être mesurée au moyen d'une méthode appropriée. Il appartiendra donc au Gouvernement de décider notamment pour quelles personnes l'exposition aux radiations doit être mesurée individuellement (dosimétrie individuelle) et à quels intervalles. Il pourra ainsi en principe dispenser d'une surveillance dosimétrique individuelle les personnes travaillant dans des conditions telles qu'il est très improbable que les expositions annuelles puissent dépasser une fraction des limites de dose. Il suffira dans ce cas d'évaluer les conditions régnant dans l'environnement de travail.

Le projet de loi précise en outre que lorsqu'une surveillance dosimétrique est prescrite, les personnes exposées aux radiations sont tenues de s'y soumettre faute de quoi elles risquent des sanctions.

3.4. Applications médicales des rayonnements ionisants

L'exposition du patient aux radiations est laissée à l'appréciation du médecin qui est toutefois tenu d'appliquer les principes de justification et d'optimisation. Le projet de loi donne

au Gouvernement la possibilité d'édicter des dispositions sur la protection des patients. On a renoncé par contre à réglementer le devoir du médecin d'informer le patient lors d'applications de rayonnements à des fins diagnostiques ou thérapeutiques et de demander son accord. En procédure de consultation, les organisations de médecins ont à juste titre relevé qu'il ne s'agissait en l'occurrence pas d'un aspect de la radioprotection, mais d'une ingérence dans la relation médecin-patient. Il n'est pas nécessaire de régler ces questions dans la loi sur la radioprotection car les applications de rayonnements à des fins diagnostiques et thérapeutiques sont soumises aux mêmes prescriptions que les autres actes du médecin à l'égard de son patient.

3.5. Déchets radioactifs

Le projet de loi contient des dispositions fondamentales valables pour tous les déchets radioactifs, y compris ceux provenant de l'utilisation de l'énergie nucléaire.

Le projet de loi statue une obligation de livraison pour les déchets radioactifs provenant de la médecine, de la recherche et de l'industrie. Les producteurs de tels déchets assument les frais de leur conditionnement, entreposage et élimination par l'Etat.

Une réglementation spéciale sur l'élimination des déchets radioactifs provenant de l'utilisation de l'énergie nucléaire est prévue à titre complémentaire dans le nouveau projet de loi sur l'énergie nucléaire. Les déchets radioactifs provenant de la médecine, de la recherche et de l'industrie, après leur ramassage, seront nouvellement régis par cette loi. Les installations de conditionnement et de stockage de tous ces déchets radioactifs sont des installations nucléaires au sens de la loi sur l'énergie nucléaire.

En ce qui concerne l'importation de déchets radioactifs, le projet de loi sur la radioprotection fait la distinction entre ceux qui proviennent de Suisse et ceux de l'étranger. En principe, ces derniers ne doivent pas être introduits en Suisse pour y être éliminés. On ne doit cependant pas faire obstacle à cette possibilité au cas où elle apparaîtrait judicieuse dans le cadre d'une répartition internationale des tâches. Mais elle ne saurait être admise que si un accord de droit public international a été conclu.

3.6. Responsabilité civile

Les prétentions en responsabilité civile ou en dommages-intérêts pour des dégâts occasionnés par des rayonnements ionisants et ne relevant pas de la loi sur la responsabilité civile en matière nucléaire, sont prescrites 3 ans après que le lésé ait eu connaissance du dommage et de la personne civilement responsable, et en tout cas au plus tard 30 ans après la fin de l'effet dommageable.

4. CONCLUSIONS

Le projet de loi sur la radioprotection sera discuté aux Chambres fédérales en 1988. Sa mise en vigueur est prévue en 1989. Plusieurs groupes de travail élaborent actuellement les dispositions d'application de la loi (révision de l'ordonnance sur la radioprotection).

RADIATION PROTECTION STANDARDS IN THE UNITED STATES

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Standards to protect workers and members of the general public against any harmful effects of ionizing radiation are numerous and complex in the United States. Many Federal agencies have protection responsibilities, our Congress limits the discretionary authority given to these agencies in providing for this protection, and our court system appears at times to render judgments that are illogical to our sense of the degree of radiological protection required. To many our standards appear to be overprotective in that they have, at best, marginal health benefits and without question are costly to implement. Government agencies, the Congress, industry, professional organizations, and others have expressed their concerns and interests regarding standards in a variety of ways:

* Need for consistent radiation policies;
* Need for mutually consistent and coordinated radiation regulations and standards, particularly those involving multiple agencies and jurisdictions;
* Establishment of radiation levels below regulatory concern (de minimis);
* Coordination of U. S. policies and positions on radiation issues at international meetings;
* Clarification of an ALARA policy;
* Need for scientifically-based standards;
* Examination of "umbrella" dose limits, dose commitment methodology, collective dose application, etc.; and
* Introduction of a risk-based standard system, standardizing risk estimation techniques and approaches to risk comparability.

It is against this background that the Committee on Interagency Radiation Research and Policy Coordination (CIRRPC) undertook a project to enhance its knowledge and understanding of the principle standards in the United States that limit and control radiation exposures. CIRRPC is a committee of eighteen Federal agencies, represented by senior policy makers, and complemented by a Science Panel whose membership is senior radiation scientists from fourteen of the member agencies of the policy body. Oak Ridge Associated Universities provides the necessary administrative and technical support to CIRRPC and was tasked to develop a compendium of U.S. radiation protection standards that would include the major legal and technical facts or requirements contained therein.

Fact Sheets (FS) have been developed for twenty-three "standards" (generally enforceable regulations or Federal guidance approved by the President) and two proposed revisions to existing regulations-NRC's 10CFR20 and DoL/MSHA's 30CFR57, NRC's basic protection standard and DoL's underground mining standard, respectively. Of these twenty-three standards, nine are in the form of guides and fourteen are enforceable standards. Seven enforceable standards have been promulgated by the U. S. Environmental Protection Agency (EPA), four by the Nuclear Regulatory Commission (NRC), two by the Department of Labor (DoL), and one by the Food and Drug Administration (FDA). Five of the seven Federal Guides, presently in-effect and Presidentially-approved, were promulgated by the former U. S. Federal Radiation Council (FRC) and the remaining two by the EPA Administrator, who was given the responsibility to "advise the President on radiation matters directly or indirectly affecting health" in 1970.

Selection of a "standard" to be in the compilation required that it be (i) published in the U. S. government's Federal Register as a proposed or final "standard"; (ii) provide basic public health protection requirements, i.e. not implementing requirements; and (iii) not be limited to controlling exposures within the agency promulgating the standard. This latter criterium excludes, for example, those requirements published as "orders" by the Department of Energy (DoE) that control exposures in facilities owned by them. It is noted, however, that intraagency requirements are expected to be in accord with Federal guidance, such as that published by the FRC in 1960 which places limits on population exposure and that published by EPA in 1987 which provides the basic protection requirements for occupational radiation protection.

The following information is addressed in each Fact Sheet:

* Short and full title;
* Authorizing statute: purpose and radiation protection provisions;
* Responsible agency(s);
* Description of standard: effective dates and background, general radiation provisions, rationale for detailed requirements, description of detailed requirements;
* Related standards.

Exposure concerns addressed by a specific standard or by multiple standards are:

* Occupational and general population;
* Occupational;
* Radon in underground mines;
* Levels of certain internal emitters in the environment, including TRU nuclides;
* Protective action guides for certain radionuclides;
* Specific standards to control air emissions;
* At-the-tap drinking water;

* Uranium fuel cycle, excepting mines and Rn from uranium mill operations;
* U and Th mill tailings: active and inactive;
* Mining effluents and underground injections;
* High-level and TRU waste operations and disposal;
* Ocean dumping;
* Low-level waste;
* ALARA design for commercial light-water reactors;
* Electronic consumer products: TV receivers, x-ray diagnostic and security systems.

Review of the Fact Sheets shows no clear intent to be consistent in either the statutory language or the explicit or implicit protection objective of the standard. For example, while the Atomic Energy Act, under which most of the EPA and NRC standards are established, leaves to the regulatory agencies the manner in which the objective is achieved to "protect health or minimize danger to life or property", the Clean Air Act and the Safe Drinking Water Act administered by EPA requires them to set limits that would avoid "any adverse effect on the health of persons" and "with an ample margin of safety." As a result of the wide interpretation of these "instructions" to agencies, agencies use an equally wide range of rationales in developing and promulgating standards. Control technology capabilities and cost (ALARA) may or may not be a consideration; current emission levels may be the basis; or a level of health risk expressed as annual or lifetime risk may be the operative criterium. Overall ALARA and non-degradation appear to be the principle criteria for protecting members of the general public and a level of lifetime risk no greater than that found in other industries for workers.

"Standards" are of the "umbrella" (general) form, such as an annual population limit for all sources of 5 mSv [0.5 rem] in the proposed revision of 10CFR20, or very source specific, such as ca. 0.7 Bq/m(2)-s [20pCi/m(2)-s] limiting radon emission from a disposal site for uranium mill tailings. The standards may be expressed in a variety of ways: activity per unit volume or mass, dose equivalent or effective dose equivalent, exposure or emission rate, total activity released over a period of time (e.g. 10,000 years), or activity per unit of annual electrical power produced. Standards to control exposures in the work place are near identical, limiting the workers annual exposure to 50 mSv [5 rem]. However, numerical limits for the general public are highly variable and in some instances quite low, for example, the limit for beta particle and photon emitters in drinking water is 0.04 mSv [4 mrem] per year.

Details of the information contained in the Fact Sheets and how the information might be used in examining standards applicable to a specific source can be illustrated by comparing standards related to controlling the environmental releases from commercial light-water nuclear reactors licensed by the U.S.N.R.C.

Shown below are the whole body dose equivalent limits for members of the general public. Not shown are the specific requirements in the actual operating license that controls the emissions from a given nuclear power plant. These requirements are stated in the plants Technical Specifications and together with the operators' actual procedures further control emissions.

Designation	Rationale	Whole Body Annual Limit
Federal Guidance (FRC#1) (1960)	No "undue hazard"	5 mSv
NRC's 10CFR20 (1960)	No "observable health effects, ample margin of safety	5 mSv
EPA's 40CFR190 (1977)	Cost control/ risk reduction ratio	0.25 mSv
NRC's 10CFR50, App. I (1975)	ALARA:Design and operation	0.03 mSv liquid 0.05 mSv air

The compendium of radiation standards provide CIRRPC with the data base necessary for a more detailed evaluation of U. S. standards and vital information useful to its policy coordination responsibilities.

THE USE OF RADIOACTIVE SUBSTANCES IN THE NETHERLANDS

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INTRODUCTION

Within the framework to study the possibilities to reduce the amount of low and intermediate level radioactive waste, detailed information was collected about the use of sealed and unsealed radioactive sources in the Netherlands [He84; He87].
Information was collected by inquiry amongst licencees (unsealed sources) and by studying the outstanding licences (sealed sources).
Quantities of radioactive materials will be expressed in terms of activity (GBq) and in terms of (radio)toxicity (ALI-units). The number of ALI-units is the quotient of activity and most restrictive value of Annual Limit on Intake for ingestion with respect to the stochastic effects [IC78].

Relative distributions are given according to type of practice, to radiotoxicity group and corresponding to the halflife of the respective nuclides. The classification of nuclides in four radiotoxicity groups is according to directives of the European Community [EU84]; classification of nuclides according to their halflives has been done in five groups; as follows:

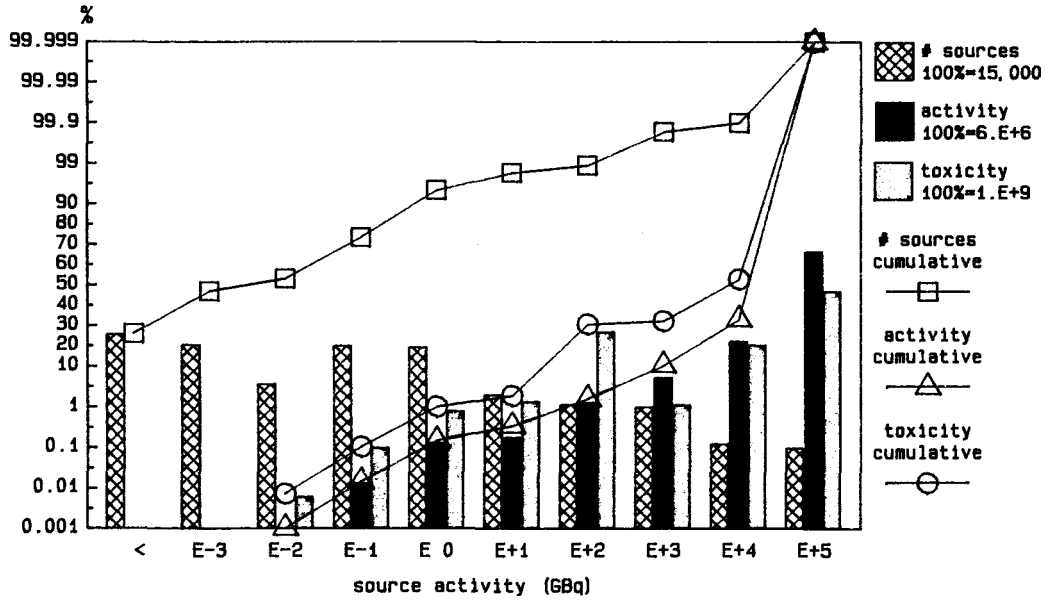
Group A: very high toxicity	$T_{\frac{1}{2}} \leq$ 10 days
Group B: high toxicity	10 days $< T_{\frac{1}{2}} \leq$ 100 days
Group C: moderate toxicity	100 days $< T_{\frac{1}{2}} \leq$ 1000 days
Group D: low toxicity	$T_{\frac{1}{2}} >$ 1000 days; β/γ-emiters
	$T_{\frac{1}{2}} >$ 1000 days; α-emiters

SEALED SOURCES

Sealed sources are used at approximately 1200 different locations in the Netherlands. There are two installations for irradiation purposes with a cobalt source of 1.5 10^7 and 1.5 10^8 GBq respectively. These are excluded in further evaluations.
Around 15,000 sources comprise a total activity of approximately 6 10^6 GBq. Differentiation per nuclide learns that this activity corresponds to 1 10^9 times the respective ALI values.
For about 2% of the number of sources the source activity is more than 1 TBq; the accumulated activity for these sources is 97% of the total activity. This corresponds to 70% of the total toxicity. The 75th percentile value in the frequency distribution of the number of sources corresponds to a source activity of 1 GBq. This group represents very low percentage of accumulated activity (0.03%) and toxicity (0.1%) (see fig.1).
For sealed sources the following types of practice are distinguished:
- measuring and control techniques (me & co)
- medical (medi)
- non-medical radiation sources, gammagraphy and neutron generators (rs, gg, ng)
- education and instruction (edu)
- electric components, deionisation and luminescence (ec,di,lu)
- manufacture of consumer goods (cons)
- a complexity of various types of practices (comp)

Fig.1

Number of sources, activity and toxicity sorted by source activity



Note: E+n means source activity varies between 10^n untill 10^{n+1}

Fig.2 shows that most of the number of sealed sources are used for measuring and control techniques (almost 40%) and that they are distributed over 900 out of 1200 locations (75%). In terms of activity this type of practice contributes less than 5% to the total but in terms of toxicity it represents approximately 35%. Other high contributions in terms of toxicity are related to irradiation sources for medical and non-medical use, respectively 45% and 15%. In terms of activity these irradiation sources together contains approximately 80% of the total activity.

Distribution according to radiotoxicity in four groups is given in fig.3. It shows for example for group B a number of sources of about 3000 (20%) spread over 420 locations (35%) with an accumulated activity of approximately 5 10^5 GBq (80%) corresponding to a toxicity of 6 10^8 ALI units (nearly 60%).

In fig.4 distribution is given according to halflives. It can be seen that nuclides with a halflife greater than 1000 days are dominant in all ways.

The inventory per nuclide learns that the activity is determined for nearly 100% by Co-60 (80%), Cs-137 (10%) and Ir-192 (10%).
The important contributions to total toxicity is from Co-60 (60%), Am-241 (30%) and Cs-137(10%).
With respect to the number of sealed sources the contribution of seven nuclides is essential: H-3 and Cs-137 (23% each), Co-60 (15%), Ra-226 (8%), Am-241, Ni-63 and Sr-90 (5% each).
Application of computer database techniques resulted in information which is now used to improve the effectiveness of the system of licencing and notification as well as the efficiency of inspection by competent authorities.

Fig.2 **Relative distribution of locations, sources, activity and toxicity sorted by type of practice**

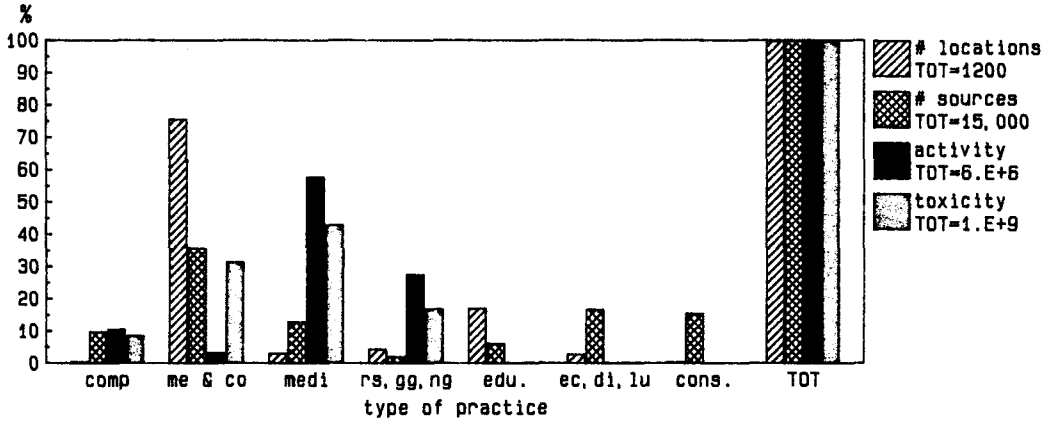


Fig.3 **Relative distribution of locations, sources, activity and toxicity sorted by toxicity group**

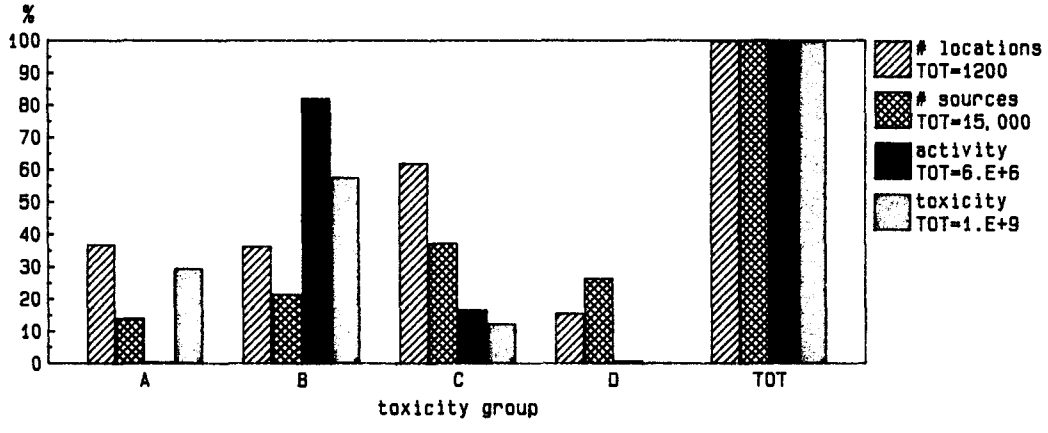
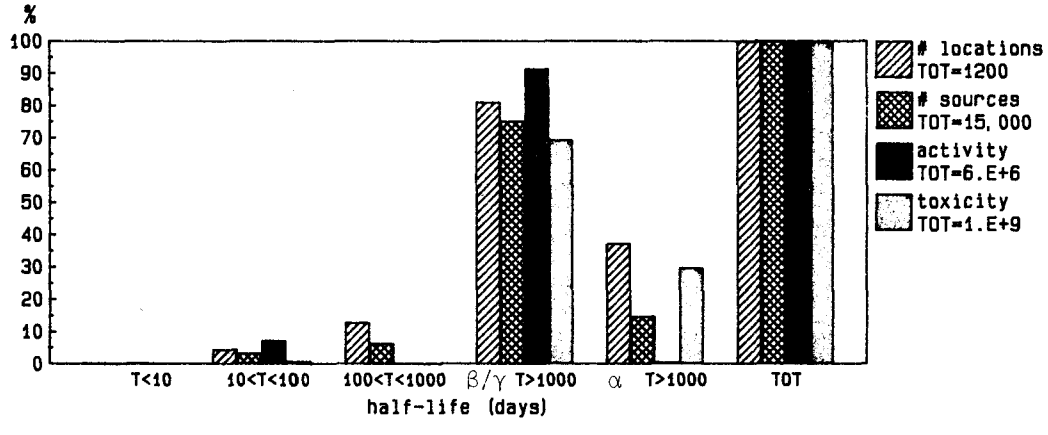


Fig.4 **Relative distribution of locations, sources, activity and toxicity sorted by half-life**



UNSEALED SOURCES

Unsealed sources are in use at circa 250 different locations. The accumulated activity on a yearly base is over 6 10^4 GBq. Differentiation per radionuclide learns that this activity corresponds to app. 4 10^5 times the respective ALI values.

Table 1 shows the relative distribution of activity and toxicity sorted by type of practice. About 85% of total activity is used for medical in vivo diagnostics with a corresponding radiotoxicity in terms of ALI units of 4 10^4 (10%). In therapy over 1 10^3 GBq (2%) is used on yearly base which corresponds to 80% of the total toxicity. Medical applications all together cover about 90% of both activity and toxicity.

TABLE 1

Type of application	Activity (%)	Toxicity (%)
Medical in-vivo	85	10
Medical therapy	2	80
Medical in-vitro	0.1	2
Research	6	6
Industrial	6	0.4
Animal experiments	0.3	1

Distribution according to radiotoxicity is shown in table 2. The major contribution (95%) to total activity is from nuclides of group D, but in terms of toxicity group B nuclides are dominant (about 90%).
From table 3 one can see that unsealed radioactive substances with a halflife less than 10 days correspond with approximately 90% of both total activity and toxicity.

TABLE 2

Toxicity group	Activity (%)	Toxicity (%)
A	<	0.5
B	3	92
C	2	3
D	95	5

TABLE 3

Half-life (days)	Activity (%)	Toxicity (%)
$T_{\frac{1}{2}} \leq 10$	90	92
$10 < T_{\frac{1}{2}} \leq 100$	0.7	7
$100 < T_{\frac{1}{2}} \leq 1000$	0.1	<
β/γ $T_{\frac{1}{2}} > 1000$	8	0.5
α $T_{\frac{1}{2}} > 1000$	<	0.5

'<' means less than 0.05%

With respect to activity the radionuclides of great dominance are Tc-99m (77%), H-3 (8%) and Kr-81m (7%). Toxicity is mainly determined by the radionuclides I-131 (86%), I-125 (6%) and Tc-99m (4%).
Collected information about the use of unsealed sources provided tangible points of application to reduce low level waste in the Netherlands.

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THE STATE OF RADIATION PROTECTION IN IRAN

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A national radiation protection authority (NRPA) is a vital need in a country to give proper advice on radiological safety and protection matters and to enforce rules, regulations and standards to reduce exposure of man and his environment from harmful effects of ionizing radiation to a level below which preventing detrimental non-stochastic effects and limiting the probability of stochastic effects while allowing useful applications of radiation to be advanced for human benefit (1). To achieve this goal, the NRPA needs to have governmental authority for regulatory enforcement and it should have a qualified leader, experts, proper equipment and dosimetry services as well as scientific and technical support.

Historically, radiation protection in Iran can be related to when the first x-ray machine was applied for medical diagnosis. However, organized activities were started with the establishment of the Tehran University Nuclear Center (TUNC) in 1959, and within a broader scope when AEOI research reactor went into operation in 1967. Until 1974, the RP activities were limited to those at TUNC and to those requested by medical, industrial and research institutions.

In 1974, the Atomic Energy Organization Law of Iran was ascribed the responsibility for radiological safety and protection to the AEOI (2). Then this responsibility was assigned by AEOI to the Radiation Protection Department (RPD), as the national authority. Its main objectives have been to achieve the following:

-Development of criteria, recommendations, guidelines, rules, regulations and standards in radiological safety and protection,
-Radiation control for production, importation, export, ownership or any other related actions regarding radioactive materials and radiation producing machines in the country including registration, inspection and licensing,
-Reseach and development on methods and techniques of internal and external dosimetry to promote dosimetry services,
-Establishment of nationwide dosimetry services for routine and emergency personal monitoring, environmental measurements, radiation field measurements and instrument calibration,
-Environmental protection due to natural and man-made sources,
-Education and training of radiation workers and public.

The RPD's organization and functions have been divided into three main RPD divisions: 1-Radiation Protection Control, 2-Radiation Dosimetry Research and Development and Services, and 3-Radiological Protection of the Environment, the functions and activities of each division are as follow:

1-Although radiation applications are widely spread in Iran in the fields of medicine, industry, agriculture, nuclear industry, education and research, the applications of x-ray machines and radionuclides in medicine predominates. Many types of sources such as

high acitvity sources, high energy x-ray machines, neutron generators, neutron isotopic sources, x-ray diffractometers, etc. are also applied in industrial radiography, radiation processing, gauging, material analysis, etc. Thanks to the Atomic Energy Law of Iran which has made radiation control program easy due to requirements for prior permission from RPD of AEOI for any transactions regarding radiation producing sources. In fact opening of any credit account in any bank for ordering radiation sources require prior permission from RPD of AEOI. So far as the rules, regulations and standards are concerned, basic radiation protection standards and requirements have been prepared previously which is under revision.

Eight separate sections, forming the Radiation Protection Control Division, have the resposibility over radiology departments, nuclear medicine and test laboratories, research and educational institutions, nuclear facilities, industrial centers, mines and non-ionizing radiation installations, as well as registration and accounting, and medical surveillance. The Medical Section is responsible for periodic examinations of AEOI personnel as well as monitoring medical results of radiation workers across the country and examinations of individuals overexposed. Non-ionizing radiation control is not well established yet, however efforts are being done to enforce NIR regulations. The division also provides dosimetry measurments and calculations for pregnant women with medically exposed fetus. Efforts have been done by TV programs as well as journal articles and posters to reduce such unwanted exposures (4). Also genetically significant dose due to medical exposure by x-rays was surveyed and found to be about 0.1 mSv (5).

2-Radiation dosimetry research and development to provide nationwide dosimetry services for public and radiation workers is the prime responsibility of this division. Five sections deal with different aspects of dosimetry for beta, x and gamma, neutrons, alpha particles, etc. using film badges, thermoluminescence dosemeters, soild state nuclear track detectors, ionization chambers, etc. for personnel and environmental dosimetry as well as radiation field measurements and calibration of instruments.

A film badge service provides personnel monitoring services to about 7500 persons from 1200 institutions as well as doing research and development (6). This service was established by TUNC in 1960 and it was transfered to AEOI in 1974. A thermoluminescence laboratory also gives dosimetry services to some groups of radiation workers as well as environmental monitoring. For neutrons and charged particle monitoring, solid state nuclear track detectors are used based on research and development done at AEOI (7-9). A neutron dosimetry service is being developed based on multi-component albedo dosemeters developed at RPD (8,10). Biological dosimetry is also provided to some groups of radiation workers especially those being overexposed. So far as the public monitoring is concererned, a national indoor radon monitoring program has been set up based on electrochemically-etched polycarbonate inside diffusion type chambers some results of which are being reported at this conference (11). Also, TLD's and film badges have been accompanied the radon chambers for indoor beta-gamma radiation measurements especially in high radiation area of Ramsar. Radiation field and instrument calibration is based on secondary standard ionization chambers. A joint project

with IAEA is also being carried out on SSDL. Many other dosimetry projects especially as B.S., M.S. and Ph.D. theses are being carried out at RPD.

3- A nationwide environmental protection against natural and man-made sources are being conducted to assess critical radionuclides from inside or outside the country's boundaries, critical exposure pathways and calculation of population exposure and evaluation of annual doses as well as the determination of internal contamination by bio-assay techniques and whole body counting are the main responsibility of this division. Four sections in this divistion do research for radioactive determination in air, water, soil, vegetables and foodstuff as well as measuring internal contamination to achieve above. One of the main functions of the division after Chernobyl accident has been environmental measurements and radioactivity level control in foodstuff imported to prevent food with radioactivity above Iran's Intervention Level. Over 15000 activity and spectrometry measurements have been done after chernobyl.

Besides the organizational responsibility of each division, the education and training is also of primary importance for RPD (12), as a part for enforcement of rules and regulations. Such education and training is divided into three main categories including: a) formal university courses for students in the field of science and technology, b) intensive courses for radiation workers, and c) education of general public. Such educational activities are summarized as follow:

a) The first formal university health physics course was organized by the Tehran University Nuclear Center as a three credit hour M.S. level course which is still being taught there. A one and half year M.S. program was offered twice in the School of Public Health of Tehran University, the first of which was in cooperation with WHO in 1973. In 1977, the RPD of AEOI also offered a one-year M.S. equivalent program in health physics. At the moment, health Physics courses at B.S. and M.S. levels are offered in nuclear physics options of physics or nuclear technology programs of almost all universities, as well as Ph.D. level courses at Amir Kabir Polytechnic University in collaboration with the Atomic Energy Organization of Iran. Besides formal courses, education and training through research as well as different level theses projects have been or being conducted at RPD of AEOI and other institutions.

b) Intensive national and international courses have been offered on different occasions for radiation workers. As examples, a six-week course on "Radioisotope Applications" were offered for several years since early in 1960's, with emphasis also on health physics for medical doctors, scientists and engineers. An international course on "Handling of Radiation Accidents" was offered by IAEA at TUNC in 1967. Since then several other intensive courses have been offered. In 1987, some national radiation protection courses being conducted monthly, have been organized by RPD of AEOI for radiation workers in different fields, consisting of 100 hours of theory and practice within three weeks. Over 800 applicants have registered. Since September 1987, about 200 radiation workers have been trained during five separate courses.

c) Public education and training have been of great concern to

RPD, and have been carried out by TV programs, journal articles, newspapers, brochures and posters. A formal TV serial was prepared byRPD in 1983 in cooperation with National TV Network consisting of 18 haf-hour programs on radiation protection matters epecially in medical applications of x-ray machines and radionuclides. This program have been very effective for general public and professionals.

Besides the above activities, some other organizations such as the Ministry of Health and Ministry of Labor have had some minor activities respectively in medical and industrial aspects of radiation protection in some centers under their influence. However, due to the Atomic Energy Organization Law of Iran, rules and regulations developed by EAOI are enforced.

In conclusion, our experiences with such a program as well as some personal views can be summarized. Before any rules and regulations can be enforced in a developing country, a well-organized RP program having a qualified leader,qualified experts, proper equipment, maintenace, inspection facilities, motives as well as budjet and govenment authority should be developed with emphasis on education and training. Although IAEA, WHO or other organizations may be effective for education and training, a developing country should develop its own potential to do so. A proper personal monitoring is a vital need. Although film badge is usually considered a "poor-man's" personal dosimeter, it still should be preferred before one can develop other personal dosimetry systems with self-support. The system of dose limitation including dose equivalent limit and derived limits for public should be adjusted for countries facing only the cost part of a nuclear program and not the benefit side of the coin. In fact, I personally think the dose limit for radiation workers and especially the public should be decreased from its present limit. So far as the quality factor for high-LET radiation especially for neutrons are concerned, the $\bar{Q}$ values recently proposed by ICRP (13) and ICRU (14) should be applied.

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RADIATION PROTECTION IN THE MINISTRY OF NUCLEAR INDUSTRY

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ABSTRACT

Nuclear fuel cycle has been established and operated in China. The two nuclear power plants being constructed, and more and more radioactive isotopes are produced and used year by year. This paper introduces on current status for radiation protection in nuclear industry as follows:

1. Administrative and scientific research structures for radiation protection in the Ministry of Nuclear Industry;

2. Development and implement of rules, regulations, guidelines and standards concerning the design, construction and operation of facilities for radiation protection;

3. Monitoring techniques and results are described for internal and external radiation doses to occupational workers and relative quality assurance and principle ALARA are involved too;

4. Environmental radioactive monitoring and public dose assessment relating to nuclear industry;

5. Radioactive waste management practices will briefly be given.

6. Specialistic education and professional training of management operating staff in radiation protection field.

A large attention has been paid by our Government to radiation protection work since founding the Ministry. The principles "safety first, quality first" and "radiation prevention leading the way" had been worked out and have been implemented strictly, and a series of necessary protection measures were taken earnestly, so that no acute radiation-induced death was occurred for near 30 years. A great quantity of radioactive wastes are basically handled, treated and conditioned. It is proved through environmental survey that no serious pollution was taking place.

The paper also points out some problems which need to be dealt with in the field of radiation protection for nuclear industry.

REVIEW OF IAEA RECOMMENDATIONS ON THE PRINCIPLES AND METHODOLOGIES FOR LIMITING RELEASES OF RADIOACTIVE EFFLUENTS TO THE ENVIRONMENT

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1. INTRODUCTION

The limitation of radioactive releases is governed by the basic principles of radiation protection as presented in the ICRP Publication No.26 and IAEA Safety Series No.9. Under its current programme on release limitation the IAEA issued Safety Series No.77 on principles for release limitation and Safety Series No.67 on protection against transboundary radiation exposures. A Safety Guide on global upper bounds [1] is now nearly ready for publication, and, to guide on the application of Safety Series No.77, four documents are in various stages of completion [2,3,4,5].

2. RELEASE LIMITATION

A practice as a whole should be justified with respect to the net benefit versus detriment, and the control of releases must be optimized. The principal annual dose limit for members of the public given in the ICRP's Paris Statement (1985) is 1 mSv. Since this limit is individual related, irrespective of source, account must be taken of all sources, present and future. For this reason a source related limit, lower than the dose limit and called the source upper bound, must be set by the competent authority as the boundary condition for optimization. Therefore, the primary dose limit cannot be used in the design of nuclear installations or for planning operations. When establishing 'source upper bounds' the authorities should consider exposures from foreseen global and regional sources, and also leave some margin for unforeseen practices in the future. The competent authority may choose a fraction F of the dose limit to arrive at the source specific upper bound (H_{UB}) given by:

$$H_{UB} = F\,H_{limit} - H_{regional} - H_{global}$$

Authorized limits for release are set by competent authorities taking many factors into account. From radiation safety considerations the release limit would be the result of satisfying both the upper bound and the optimization requirements carried out under the constraint of the upper bound.

The annual release upper bound may be derived from the dose upper bound by use of the overall transfer factor f_{jkl}, where j is the population group, k is the release mode and l is the radionuclide. Then, if no other radionuclides are released, the release upper bound R^*_{kl} for a critical group j' for radionuclide l is:

$$R^*_{kl} = \frac{H_{UB}}{f_{j'kl}}$$

Normally the situation is much more complicated. When there are several release modes k, subject to a common release limitation, but

still only one radionuclide 1, it is necessary to prescribe a set of release upper bounds R^*_{kl} such that

$$\sum_k f_{j'kl} R^*_{kl} \leq H_{UB}$$

When a mixture of radionuclides 1 contributes significantly to the exposure of the group j' that is critical for the mixture and release modes, the R^*_{kl} values must satisfy the condition:

$$\sum_k \sum_l f_{j'kl} R^*_{kl} \leq H_{UB}$$

This condition only defines sets of R^*_{kl} values that, together, constitute a release at the upper bound. When different radionuclides as well as different release modes affect different critical groups then a limitation based on a realistic critical group may become complicated. Two simplifications are possible: (1) to postulate the release modes and radionuclide compositions that are most likely, and to identify the relevant critical group j", and then to calculate R^*_{kl}; or (2) to define a hypothetical critical group assumed to have all the characteristics and exposure conditions of the various critical groups and then to calculate R^*_{kl} by using the condition:

$$\sum_k \sum_l f_{j"kl} R^*_{kl} \leq H_{UB}$$

Through optimization a new release R' is derived which will result in a collective dose less than or equal to that associated with R*, and must also satisfy the condition:

$$\sum_k \sum_l f_{j'kl} R'_{kl} \leq H_{UB}$$

The competent authority may select the optimized control option and set an authorized release limit close to R' but must never set the limit higher than R*. The authorized limits must make allowance for variations and uncertainties deriving from fluctuations due to operational variations.

3. TRANSBOUNDARY RADIATION EXPOSURES

Some practices may give rise to radiation exposures beyond national frontiers. National authorities should apply some restrictions to limit the transboundary components of releases. To optimize by using cost-benefit analysis, a monetary cost Y has to be assigned to the radiation detriment, such that $Y_\alpha = \alpha S$, where Y_α is the cost of the objective health detriment and S is the collective dose. Non-objective health detriments such as anxiety or risk aversion may also be considered by including Y_β such that $Y = Y_\alpha + Y_\beta$, but

for transboundary radiation protection it is essentially the Y_α which is important. Also, different costs might be assigned to the collective dose contributions inside and outside a country through the use of different values of alpha, so that $Y_\alpha = \alpha_n S_n + \alpha_t S_t$

where n and t denote 'national' and 'transboundary' respectively.

The value of α may be determined by a method based on direct calculation of the loss of economic output - the 'human capital' approach. The expected loss of healthy life attributable to a radiation dose of 1 man.Sv, averaged over a large population, is about 0.4 man-years. On this basis, the human capital approach would give a minimum value of alpha which would be given by:

$$\alpha_{min}\ [\text{US\$/man.Sv}] = 0.4\ [\text{man-years/man.Sv}] \times I\ [\text{US \$/man-years}] + C\ [\text{US\$/man.Sv}]$$

where I is the per caput gross national product (GNP) and C is the cost of additional medical care per man-sievert.

One basic recommendation of the IAEA is that the monetary value assigned to the unit collective dose for the purposes of costing transboundary collective doses should be no less than the value used for the national collective dose, i.e. $\alpha_t \geq \alpha_n$ provided that $\alpha_n \geq \alpha t\ min$, otherwise $\alpha t \geq \alpha t\ min$, where $\alpha t\ min$ is the recommended minimum value of α_t.

The value recommended for $\alpha_{t\ min}$ is US$3000 per man-sievert at 1983 prices. It is anticipated that national authorities in affluent countries would find it appropriate to use substantially higher values for α_n and consequently also for α_t.

4. REGIONAL AND GLOBAL UPPER BOUNDS [1]

Regional and global upper bounds should be fractions of the primary dose limit and will be less than the source upper bound, subject to the constraint that:

$$H_{UB} + pH_{UB\ regional} + qH_{UB\ global} \leq FH_{limit}$$

where p and q are the fractions of the regional and global dose upper bounds allocated to sources other than the one under consideration.

For regional upper bounds separate regions and separate regional upper bounds may need to be specified for releases to the atmosphere and releases to the marine environment, and separate dose upper bounds specified for releases to particular rivers. Account should be taken of relevant existing regional agreements. For sea dumping the source upper bounds for specific dump sites and regional upper bounds for ocean areas would need to be established by agreement between the countries within or

bordering the area. However, it would be advisable to have one upper bound for use in all regions and this would be the global upper bound for sea dumping for use in all regions. In establishing an overall global dose upper bound it will be necessary to apportion among present and future practices.

5. METHODS FOR INDIVIDUAL AND COLLECTIVE DOSE ASSESSMENT [2]

The estimation of individual dose is generally based on models and selected parameter values likely to lead to conservative assessments of actual doses. However, for assessing collective doses different parameter values, and in many cases also quantitatively different models, may be needed. For the steady state situation the concentration factor (CF) method is most widely used, and is adequate and preferable for assessing the consequences of planned routine continuous releases. For short, unplanned releases or when radionuclides from planned releases accumulate in the environment or persist for long periods after introduction, it may be necessary to introduce time dependence into the CF method or to use more complex time dependent models - the systems analysis approach.

In assessing the exposure of the general public, account has to be taken of the age of the exposed population, the chemical form of the radioactive material and effective doses per unit intake appropriate to people of different ages. In assessing collective doses, it is generally adequate to use annual limits on intake values for adults only. Age and sex differences will, however, affect the accuracy of the commonly used dose conversion factors given in ICRP Publication No.30 for adults, and therefore factors appropriate for the age group concerned should be used in most applications.

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APPLICATION OF THE IONISING RADIATIONS REGULATIONS 1985 TO A RESEARCH ESTABLISHMENT IN THE UK

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Three important reasons for the UK Health and Safety Executive (HSE) to embark on the preparation of the Ionising Radiation Regulations were:

1 the International Commission on Radiological Protection (ICRP) Publication 26 revised the basic recommendations for radiation protection on which national provisions are based.

2 As members of the European Commission the UK is bound by the Euratom Directives to align its national legislation with other member states. The Directives lay down the basic safety standards for the health protection of the general public and workers against the dangers of ionising radiation.

3 The UK Health and Safety at Work etc. Act 1974 allows outdated legislation to be progressively replaced by a system of regulations and approved codes of practice designed to maintain or improve the standards of health, safety and welfare in the workplace.

After a lengthy consultation period with industry, trade unions and independent experts the Regulations were made by the UK Secretary of State for Employment and came into complete operation on 1st January 1986. They are supported by a Code of Practice approved by the Health & Safety Commission (HSC) which provides practical guidance on compliance.

The Regulations extend to almost all places of work in the UK in which ionising radiations are used. The same basic requirements are applied to large industrial sites such as nuclear power stations or to persons using ionising radiations in teaching, research or medical applications. They impose duties on employers to protect employees and other people by keeping exposure as low as reasonably practical. They also impose duties on employees in the conduct of their work. In total there are 41 Regulations divided into nine Parts and supported by ten Schedules. They are enforced by the UK Health and Safety Executive (HSE).

The Wellcome Research Laboratories (WRL) is the major R&D centre of The Wellcome Foundation Ltd in the UK. Its activities are directed towards the discovery and development of new medicines, vaccines and diagnostic materials and it employs 1500 people. About 400 of them, covering many scientific disciplines, use ionising radiations as tools in a variety of biological, biochemical and pharmacological techniques. In the past radiological protection has been of a good standard, thus the major problems with the application of

the new Regulations were the modification of practices and the increased administration required to record compliance.

The administration of the Regulations and integration of radiological protection procedures throughout the Site is controlled by the Company's Health and Safety Services' staff. The key post being the Radiological Protection Safety Officer (RPSO) who is advised by an external consultant Radiation Protection Adviser (RPA). Operational responsibility for safety resides with the line management and individual worker. Within departments daily supervision of radiation work is provided by a Departmental Radiation Protection Supervisor (DRPS). In effect the RPSO and 23 DRPS's undertake the duties of the Radiation Protection Supervisor stipulated by the Regulations. The Parts of the Regulations that affect WRL operations are as follows:

PART I. INTERPRETATION AND GENERAL CONDITIONS

These regulations define the scope of and terms used in the legislation and the notification of work to the HSE.

To ensure tight control at WRL, all persons using ionising radiation have to register with the RPSO, so that the H&S Services are aware of the location and type of all work being undertaken.

PART II. DOSE LIMITATION

Employers are required to restrict as far as is reasonably practicable the doses received by employees and other persons. The means proposed to achieve this are engineering controls and design features including shielding, containment of radioactive substances, ventilation and the provision of safety features and warning signs.

Most work with unsealed sources is carried out in fume cupboards behind appropriate shielding. In addition, safe systems of work in the form of Site and Departmental Local Rules (covering experimental procedures, safe access to, manipulation of and storage of sources, hygiene, decontamination and waste disposal) and appropriate protective equipment are provided. Compliance with these features is monitored by the RPSO and DRPS.

To ensure that dose limits are not exceeded internal and external personal dosimetry is employed. The Regulations specify dose limits for the whole body, for individual organs and tissues, and for the lens of the eye. Lower limits are set for women of reproductive capacity.

PART III. REGULATION OF WORK

Areas in which persons are likely to exceed specified proportions of the dose limits have to be designated by the employer as 'Controlled' or 'Supervised' areas. The detailed criteria to be used are complex and allow for the external

radiation hazard, the internal radiation hazard and the combination of both.

As the use of sources at WRL fluctuates, all areas in which unsealed sources are used are classed as 'Supervised' areas. Only a few areas are designated as 'Controlled'. These are mainly storage areas and work areas in which quantities of isotopes persistently exceed the upper limit for supervised areas, e.g. fume cupboards and laboratories involved in radiochemical synthesis or iodination procedures. Access to controlled areas is restricted to 'Classified' workers and to persons working to a written system of work which can be shown to limit their exposure.

The Site and Departmental Local Rules are designed to enable the employees to carry out work within the dose limits given in the Regulations. This allows the use of $7.5\mu Sv.h^{-1}$ which relates to three tenths of the whole body limit of $50mSv.y^{-1}$ as the operational standard for the external radiation dose rate at the boundaries of 'Controlled' areas. To ensure exposure of other people is low, $2.5\mu Sv.h^{-1}$ is the external radiation dose rate for the boundaries of a 'Supervised' area e.g. a biochemical laboratory.

All registered workers undergo formal training in radiation protection and safe working procedures on courses developed and run by H&S Services, but functional training is carried out by each department.

PART IV. DOSIMETRY AND MEDICAL SURVEILLANCE

These regulations require the monitoring of the doses received by classified persons and certain other employees and the maintenance of dose records (for at least 50 years) by an HSE approved dosimetry service.

On registration each worker is categorised for personal dosimetry based on the type and quantities of sources to be used and the nature of the work. If the employee is likely to exceed three tenths of a set dose limit they are designated as Classified Persons who have to undergo more rigorous dosimetry and medical surveillance.

Less than 10 of our 400 registered workers are classified because most of the work at WRL involves the use of unsealed sources (mainly ^{32}P, ^{125}I, ^{14}C and ^{3}H) in tracer experiments.

Film badges and extremity TLD straps are used to monitor doses. These are supplied and analysed by the National Radiological Protection Board on a 4 week cycle. Small pocket monitors with pre-set alarms are used for the daily monitoring of operators of large sealed sources or X-ray fluoroscopes. The dosimetry service is administered by the RPSO. Quarterly and annual doses are calculated and the records archived for the statutory period. Termination records are prepared when an employee leaves.

Medical surveillance of registered workers is carried out by the Site Occupational Health Physician. The health record of Classified Persons, which must be kept for 50 years, is updated every 12 months, though a medical examination of the individual is not carried out unless personal dosimetry or other factors suggest that it would be advisable.

PART V. CONTROL OF RADIOACTIVE SUBSTANCES

This Part regulates the use of sealed and unsealed sources and defines the requirements for stock control.

The majority of sources at WRL are unsealed (e.g. labelled compounds or free isotopes). The system of accounting for all sources is administered by the RPSO to ensure that the limits certified for the Site are not exceeded. The system uses an IBM PC and LOTUS software to track all acquisitions, stock levels, movements and disposals collated from monthly returns made by each department. These records are kept for 2 years and loss of stock above certain quantities (e.g. 5×10^4 Bq for ^{125}I) must be reported to the HSE.

PART VI. MONITORING OF IONISING RADIATION

Monitoring of the levels of radiation in each 'Controlled' or 'Supervised' area is required with records being kept for at least two years.

The Site Rules demand that working areas and adjacent areas are monitored after the completion of each experiment and decontaminated if required. Hand held monitors or wipe tests are used. Monthly grid monitoring is carried out in all working areas with emphasis placed on those most likely to be contaminated. All areas external to working areas (e.g. offices and corridors) are monitored on a 3 monthly basis.

All monitoring equipment is tested annually. As WRL is liberally provided with monitors, periodic testing is a significant burden and has been contracted out.

PART VII. ASSESSMENT AND NOTIFICATIONS

Hazard assessment is an important requirement. At WRL it occurs when a person seeks registration as a radiation worker, when new work is introduced and for major changes in existing work or facilities. Contingency plans for dealing with emergencies are developed and incorporated into the Local Rules.

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Although complex and procedurally demanding, The Regulations combine many aspects of good practice that responsible UK establishments have always observed.

LICENSING REQUIREMENTS FOR USERS OF IONIZING RADIATION SOURCES IN SOUTH AUSTRALIA: TRAINING COURSES AND ASSESSMENT PROCEDURES

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ABSTRACT

South Australia recently replaced its 1962 ionizing radiation legislation with a new Radiation Protection and Control Act. The new legislation does not include any transition provisions for licences. That is, the holder of a licence under the previous legislation does not receive any preferential treatment when his application for a licence is considered under the new legislation . In fact, the applicant cannot be granted a licence under the current legislation unless he has:

1. a "prescribed qualification" such as a diploma in radiography for a radiographer, or

2. "appropriate knowledge of the principles and practices of radiation protection to carry on such activities".

For any category of persons who do not have a prescribed qualification the only route by which a licence is granted is by formal assessment. It, therefore, has been necessary to provide training courses and formal assessments for various categories of professionals. Some of the features of the new legislation and the difficulties encountered in implementing it are described.

INTRODUCTION

South Australia has introduced new controls over the use of ionizing radiation: the Radiation Protection and Control Act, 1982, and its Ionizing Radiation Regulations, 1985, which became effective on 1 April 1986. These replace the 1962 legislation which did not require any test of competence to be completed before an applicant was licenced.

The new legislation controls all aspects of the use of X-ray equipment and radioactive substances, including radioactive ores. The Act requires X-ray apparatus, sealed sources and premises upon which unsealed sources are kept or handled to be registered and the users of ionizing radiation to hold an appropriate licence issued by the South Australian Health Commission.

LICENSING REQUIREMENTS

The legislation has two unique features:

1. it incorporates the general objective of the ALARA principle, and

2. does not include any transition provisions for licences. That is, the holder of a licence under the previous legislation does not receive any preferential treatment when his application for a licence under the new legislation is considered. In fact, the Act prevents the Commission from issuing a licence to anyone unless the individual concerned has demonstrated competence in radiation protection principles and practices relevant to the activities he intends to carry out.

There are only two routes by which competence in radiation protection can be demonstrated:

1. by formal assessment or

2. by having a prescribed qualification. In this case the Commission accepts that members of some occupational and professional groups have sufficient training in radiation protection for a licence to use ionizing radiation to be granted without further training or assessment.

The legislation also makes provision for certain classes of people to be exempted from holding a licence. These include persons working in low activity laboratories under the supervision of a licensed person in charge of the laboratory, or with closed cabinet X-ray units, or enclosed X-ray analysis apparatus.

PRESCRIBED QUALIFICATIONS

Several occupational or professional groups have prescribed qualifications. These are: diagnostic and therapy radiographers, radiologists, radiotherapists, veterinarians, dentists, dental therapists and dental assistants. For chiropractors the only recognised qualifications are those from the Phillip Institute of Technology, Victoria and those from the Sydney College of Chiropractic, New South Wales. Practically all of the prescribed qualifications are medical.

ASSESSMENT

For users of radiation in industry and science and in some medical areas, there were no qualifications which adequately demonstrated competence in radiation protection. Hence there was a need for assessment of these occupational and professional groups.

Between mid 1983 and the end of 1985, written examinations were set covering a variety of areas of speciality. The examination papers incorporate a core of radiation safety knowledge, particular aspects relating to each area of speciality, and knowledge of regulatory requirements. As at the end of 1987, the Commission provides formal assessment for 15 categories of non-medical speciality such as borehole logging, industrial radiography, users of unsealed radioactive substances and installers of medical apparatus.

In the medical area the Commission provides assessment for medical specialists, such as cardiologists, who are seeking a licence to operate fluoroscopic apparatus. In this case the medical specialists need to complete successfully both a written and practical examination.

TRAINING COURSES

It was realised from the outset that to equip persons not holding prescribed qualifications to pass the appropriate assessment, training courses would need to be provided. A number of courses were organised and targeted to particular groups of people. In planning the courses, several options were considered. Techsearch Inc., the commercial arm of the South Australian Institute of Technology, was approached to organise and run courses for users of sealed sources and unsealed radioactive substances. A 3-day course for sealed source users is run regularly. It assumes no previous background in physics or radiation protection.

The University of Adelaide Postgraduate Committee in Dentistry has run courses for dentists who graduated prior to 1984 and who wish to operate OPG X-ray apparatus. This training is now incorporated in the undergraduate course.

Under the previous legislation a large number of general practitioners and registered general nurses were taking radiographs. To enable these health professionals to continue to provide a radiographic service, particularly in country areas and to be licensed under the current legislation, the Commission arranged a number of courses of instruction in basic radiography both in regional centres and in Adelaide. To the end of 1987, 19 training courses have been held: 8 in regional centres and 11 in Adelaide. So far 249 general practitioners and 118 nurses have been licensed. Of these, 197 general practitioners and 112 nurses are practising outside the Adelaide Metropolitan area.

DIFFICULTIES ENCOUNTERED IN IMPLEMENTING THE LICENSING REQUIREMENTS

During implementation of the new licensing requirements a number of difficulties were encountered. The major ones were:

1. the preparation and management of large numbers of different types of examinations to cater for the diverse use of ionizing radiation in industry, science and medicine with limited resources and within a short period of time.

2. preparation and organisation of large numbers of courses for general practitioners and registered general nurses in a short period of time.

3. antagonism against the legislation. This has been the most difficult problem to deal with and has arisen mainly through the absence of transitional provisions in the legislation. Some previously licenced individuals had been using ionizing radiation for up to 40 years!

CONCLUSION

In South Australia as at the end of November, 1987 there are over 2220 persons licensed to operate X-ray equipment and over 620 licensed to use or handle radioactive substances.

A great effort has been put into implementing the new legislation both in licensing people and in registering X-ray equipment, sealed sources and premises upon which unsealed radioactive substances are used. The indication so far is that the standard of radiation protection in South Australia has improved with the introducton of the new legislation. However, it is neither possible to quantify the improvement nor to determine its cost!

APPLICATION OF INTERVENTION LEVELS IN SWEDEN AFTER CHERNOBYL

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PURPOSE OF INTERVENTION LEVELS

The International Commission on Radiological Protection, ICRP, has stated that the purpose of intervention in case of an accident should be that the reduction in detriment achieved by the countermeasure should more than outbalance the detriment carried by the countermeasure itself (1). This has later been elaborated into three goals for intervention (2): a) avoid serious nonstochastic effects; b) achieve a positive net benefit to the individuals involved by limiting the risk from stochastic effects; and c) limit as far as reasonably practicable the overall incidence of stochastic effects.

Item b) has been criticised as lacking clarity and is likely to be revised in the review process started by the ICRP after the Chernobyl accident. In general, however, the three goals have been accepted as a basis for international recommendations about countermeasures following an accident.

A large number of factors influence the achievement of the three goals a), b) and c) following a reactor accident, e. g. the time pattern of the releases, the condition of the reactor, the number of people and domestic animals affected by a decision, the sheltering capacity of buildings, the time of the day or year, the weather and traffic conditions and so on. In the case of the consequences of the Chernobyl accident in Sweden, nonstochastic effects were never possible so the only goals applicable for countermeasures were a) and b), both related to stochastic effects. The international recommendations recognise the many factors relevant to the decision in such a case and suggest a decision aiding tool in the form of a range of individual radiation doses, within which an intervention level of dose (IL) should be chosen for the particular situation at hand. The simplest expression of this dose range is for the countermeasures of sheltering and control of foodstuffs, for which the range is an effective dose equivalent of 5 to 50 millisievert (mSv). In the case of foodstuffs this applies to the first year after the accident. With the aid of suitable conversion factors, an intervention level of dose can be converted to a derived intervention level (DIL) for e. g. an initial external radiation dose equivalent rate from ground contamination, or an activity concentration in food (3).

INTERPLAY BETWEEN RADIATION PROTECTION AND NATIONAL POLITICS

In the initial phase of the Chernobyl accident, the recommendations about the range of individual doses were explained to the Swedish minister of the environment and energy, who represented the government. She was told that the radiation protection authority

was at liberty to aim for an intervention level of dose within this range. A low intervention level of dose would entail costs of many millions of US dollars but would give the population the information that the radiation risks were taken very seriously.

The response from the minister was very unequivocal. She would back up any decisions that were in line with the traditionally strong concern for environmental contamination in Swedish politics, even at the expense of considerable costs to the government budget. This meant that in the following, the authority aimed for the lower limit of the range of individual dose when deciding upon countermeasures and derived intervention levels.

EXAMPLES OF COUNTERMEASURES

The attached table contains a list of several countermeasures discussed following the Chernobyl accident. The purpose of the countermeasure is given as well as various factors entering into the decisions about its application. The experiences after the countermeasure was instituted are reported, including the dose savings and their estimated costs.

DISCUSSION AND CONCLUSIONS

The table shows that many decisions on countermeasures were based on scarce information about radiation doses. The need for countermeasures that can be simply explained conflicts with the desire to adapt decisions to radiological conditions. It was difficult to adjust decisions to such factors as sex, age, status of health and geographical conditions even though this would have been desirable according to the ICRP recommendations.

The conclusion is therefore that preestablished intervention levels tend to focus interest on the individual radiation doses and hamper the consideration of other factors which are important for the achievement of the goals b) and c) give above. Nevertheless, the discussion of the role of intervention levels in relation to other factors has an important didactic role in the emergency organisation.

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Countermeasure	Factors entering the decision	Implementation and result
Recommendation to stay indoors to secure time for consideration of further countermeasures	o respiratory protection o expected additional ground contamination next few days o holidays coming up o monitoring still incomplete o IL 5 mSv in one day o DIL 0.15 mSv/h from ground contamination	DIL was never reached but one measurement late one night indicated the possibility. Before next morning this measurement was shown to be erroneous.
Recommendation not to take iodine tablets to avoid adverse reactions	o pharmacy sales were booming o the public was concerned about the lack of tablets o maximum thyroid dose estimated to be less than 5 mSv o lower limit of dose range for IL of 50 mSv was not explicitly quoted	Recommendation widely published. Typical reaction from the public: We still want to buy tablets to keep in case the situation impairs. Later enquiry showed less than 1 % took iodine pills, 10 % considered buying pills but most of these refrained from buying.
Recommendation not to travel closer to Chernobyl than 100 km and not to abstain from travelling to areas more distant than 500 km to avoid possible exposures without being overcautious	o very little was known about radiation doses near Chernobyl o exposure times would in general be less than a few weeks o additional information might become available about radiation doses o assuming inverse proportionality to distance, first week doses would be 5 mSv and 1 mSv, respectively: no IL was quoted	Travellers generally more cautious than the spirit of the recommendation. Later information showed doses at distances above 100 km to be no higher than the highest doses in Sweden.
Recommendation not to drink rainwater and eat fresh leafy vegetables to avoid ingestion of shortlived radionuclides	o high activity levels reported in rainwater o very few leafy vegetables were exposed since growing season had not started o countermeasure was very inexpensive and thus considered to keep radiation doses as low as reasonably achievable; no IL was quoted	Countermeasure criticized for having increased public concern unnecessarily, particularly for vegetables with small annual consumption, such as parsley.

Countermeasure	Factors entering the decision	Implementation and result
Recommentation not to let cattle graze outdoors to decrease iodine levels in milk and later iodine and cesium levels in milk and meat	o grazing season was just about to start o costs could rapidly rise to millions of US dollars per week o measure judged to be very efficient with respect to iodine, and later necessary to ensure compliance with DIL for cesium in milk o IL for I-131 was 50 mSv to infant thyroid o DIL for I-131 was 10 kBq/sqm ground area or 3 kBq for the grass collected from 1 sqm o DIL for cesium was to be derived from special test farm experiments	Very extensive measurements led to a gradual lifting of the recommendation. Iodine was replaced by cesium as the main problem after about 2 weeks. The recommendation was lifted from the last parts of the country after about two months. Thyriod doses were less than 5 mSv. About 100 manSv were averted at a cost of several hundred thousand US dollars per manSv, but cost estimates have large uncertainties because of the difficulty to account for the costs saved due to absence of decreased milk consumption.
Recommendation not to market food with elevated cesium content to limit individual effective dose equivalents	o implications for Sweden's large international food trade o IL 5 mSv in first year, 1 mSv following years o DIL 200 000 Bq Cs-137 with other nuclides additional, as intake in the first year; 40 000 Bq following years o DIL 300 Bq/kg for all foods in the first year; changed to apply to basic foods only in the following years with DIL for other foods being 1500 Bq/kg	Initially strong public adherence to DIL for activity concentration in food. After about a year more understanding of the significance of annual intake limitation. Some 10% changed their food habits significantly and body burdens of cesium as a mean for Sweden were one-third of those predicted. Effective dose equivalents were less than 1 mSv first year but may for special groups reach 5 mSv following year. About 500 manSv were avertad first year at a cost of about a hundred thousand dollars per manSv.
Recommendation for special protection for farm and other workers to ensure protection when working with air filters, hay slaughter, leaf raking, ash handling etc.	o large groups exposed who were not tradiational radiation workers o Implementation the responsibility of the occupational health authority which had little experience of radiation o LI 5 mSv first year o DIL 0.02 mSv/h at 1 m distance from the source	Extensive inquiries to the occupational health authority. Respiratory protection and work planning were main measures. Studies of farmers revealed no elevated internal contamination with cesium in spite of outdoor work without special precautions i areas with ground contamination of about 100 kBq/sqm.

ORGANISATION D'INTERVENTION EN CAS D'AUGMENTATION DE LA RADIOACTIVITE

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ABSTRACT

For more than 20 years, Switzerland has devoted great attention to the emergency organization for the protection of the public in case of an increase in radioactivity (following accidental explosion of nuclear weapons or accidents in a nuclear power plant). The concepts intensively elaborated within the last ten years have on the whole been efficient in the case of the radioactive fallout caused by the accident in Chernobyl. However, some improvements were necessary in order to bring together the technical side of the radiation protection and the political authorities in the decision making concerning protective measures for the population and in their implementation. The information of the public has been improved as well. This contribution will show the most important elements of the emergency organization.

1. INTRODUCTION

L'expérience acquise en Suisse à la suite de l'accident de Tchernobyl a rendu nécessaire une révision de l'organisation d'intervention en cas d'augmentation de la radioactivité. Il s'est agi essentiellement de renforcer la relation entre les organes techniques de la radioprotection et les autorités politiques dans la prise de décision concernant les mesures de protection de la population et leur exécution, et d'améliorer l'information du public.

Une nouvelle ordonnance (datée du 15 avril 1987) définissant l'organisation d'intervention et décrivant ses tâches est entrée en vigueur le 1er mai 1987.

2. COMITE DIRECTEUR RADIOACTIVITE (CODRA)

La direction générale de l'organisation d'intervention en cas d'augmentation de la radioactivité incombe au CODRA. Celui-ci est placé sous la présidence du secrétaire général du Département fédéral de l'intérieur et comprend le vice-chancelier responsable de l'information ainsi que les directeurs des offices fédéraux les plus directement concernés par une augmentation de la radioactivité, au nombre de huit. D'autres directeurs d'offices fédéraux peuvent être appelés à participer aux délibérations si la situation l'exige.

Le CODRA a pour tâche permanente de superviser les travaux de préparation et de coordination de l'Etat-major de protection sanitaire en cas d'augmentation de la radioactivité (SARA) et de veiller à ce que cet état-major soit prêt à l'engagement.

Lors d'une intervention, il analyse la situation et tient le Gouvernement (Conseil fédéral) au courant de son évolution. Il lui propose les mesures de protection nécessaires par l'entremise du département compétent et en supervise l'exécution par la Confédération et les cantons.

Le CODRA veille à ce que l'aptitude fonctionnelle de l'organisation d'intervention soit contrôlée au cours d'un exercice annuel au moins.

3. ETAT-MAJOR DE PROTECTION SANITAIRE EN CAS D'AUGMENTATION DE LA RADIOACTIVITE (SARA)

En cas d'intervention, il est nécessaire de disposer d'un élément de conduite pour évaluer la situation radiologique du point de vue de la santé publique et préparer des propositions à l'intention du Gouvernement. L'Etat-major SARA a été créé à cet effet. Il est composé de spécialistes de la radioprotection, de médecins, de biologistes, de spécialistes de l'alimentation, de la sécurité des installations nucléaires, du renseignement et de l'information, ainsi que de collaborateurs pour l'infrastructure.

Se basant sur les messages et propositions qui lui parviennent de la Centrale nationale d'alarme (CENAL), l'Etat-major SARA informe le CODRA et lui propose à l'intention du Gouvernement les mesures de protection correspondant au degré de danger. Il met sur pied la Centrale d'information et informe les autorités fédérales et cantonales ainsi que la population.

En cas de grande urgence, il avise directement les autorités et la population et ordonne les premières mesures de protection.

4. CENTRALE NATIONALE D'ALARME (CENAL)

La CENAL est le service chargé de donner l'alerte aux autorités et l'alarme à la population en cas de danger dû à:
- la radioactivité (explosion atomique, accident dans une installation nucléaire, accident de transport, accident industriel, etc.);
- la chute d'un satellite.

Elle doit être en état de détecter tout danger et de recevoir les avis de danger; un poste d'alarme fonctionne en permanence afin de transmettre sans délai les messages reçus de l'étranger et du pays.

La CENAL analyse le danger, engage l'organisation de prélèvement d'échantillons et de mesure et exploite les résultats. Elle informe l'Etat-major SARA de la situation radiologique et lui propose des mesures pour la protection de la population.

Si une catastrophe est probable à bref délai dans tout le pays ou dans certaines de ses régions, la CENAL donne directement l'alerte aux autorités et recommande à la population les premières mesures de protection.

Lors d'accidents nucléaires survenus sur territoire suisse, elle avertit sans délai l'Agence internationale pour l'énergie atomique (IAEA), à Vienne, conformément à la Convention internationale du 26 septembre 1986 sur la notification rapide d'un accident nucléaire, ainsi que les pays voisins, conformément aux traités bilatéraux.

La CENAL a également des tâches en cas de danger d'inondation due à la rupture d'un barrage ou au débordement des eaux, ainsi qu'en cas de danger causé par des substances chimiques.

5. ORGANISATION DE PRELEVEMENT D'ECHANTILLONS ET DE MESURE

L'organisation permanente comprend:

- un réseau de 6 stations de mesure pour la surveillance permanente de la radioactivité de l'air; ces stations, appelées postes de préalerte, sont équipées d'une installation de filtrage de l'air et d'un dispositif de mesure qui déclenche une alarme lorsque la radioactivité dépasse le seuil fixé;

- un réseau automatique de 51 stations de mesure pour la surveillance permanente à grande échelle de la contamination du territoire (NADAM); le débit de dose est mesuré à 1 mètre au-dessus du sol; les résultats sont transmis toutes les 10 minutes à la CENAL; une alarme est déclenchée lorsque le seuil fixé est dépassé.

L'organisation peut être complétée par:

- un réseau de 111 postes d'alerte atomique couvrant l'ensemble du territoire, en complément du réseau NADAM; le débit de dose est mesuré manuellement à l'aide d'un appareil de détection; ce réseau mobilise essentiellement des postes de police;

- des équipes mobiles disposant de véhicules de mesure et d'hélicoptères militaires;

- des équipes de mesure du service de protection AC de l'armée.

L'évaluation du danger par irradiation interne nécessite en outre l'engagement de laboratoires de mesure chargés en particulier de déterminer la contamination radioactive des denrées ali-

mentaires et des fourrages, ainsi que des eaux potables et d'abreuvement. Ce sont les laboratoires engagés dans la surveillance normale de la radioactivité auxquels on peut adjoindre en cas de nécessité les laboratoires AC de l'armée.

6. CENTRALE D'INFORMATION

L'information des cantons et de la population s'effectue en règle générale par le canal de la Centrale de renseignement, d'information et de presse de la Chancellerie fédérale. La Centrale d'information dispose de l'appui de spécialistes de l'Etat-major SARA en tant que conseillers techniques.

EMERGENCY PREPAREDNESS IN SWEDEN AFTER NUCLEAR SAFETY IMPROVEMENTS AND THE CHERNOBYL ACCIDENT

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THE NUCLEAR POWER SITUATION IN SWEDEN

In Sweden there are 12 power reactors distributed at four sites, four at Ringhals, two at Barsebäck, three at Forsmark and three at Simpevarp (Oskarshamn). Nine are BWR (Asea Atom) and three PWR (Westinghouse, at Ringhals). The total installed capacity is about 10 GWe. After the TMI accident 1979 there was a referendum in 1980 about the nuclear future in Sweden. As a consequence of that it was decided to terminate nuclear power in year 2010, at the latest. The order and rate of closing the reactors should depend on energy demands and alternative energy sources as well as on nuclear safety and radiation protection aspects. After Chernobyl it has been further decided to consider the possibility of closing two of the reactors already in the middle of the 1990's.

THE EMERGENCY PREPAREDNESS IN SWEDEN UP TO CHERNOBYL

The dimension and shape of the emergency plans were mainly formed with nuclear accidents at Swedish nuclear power plants in mind. The four affected county administration authorities are responsible for the local planning and for the actions taken in an emergency situation. At the National Institute of Radiation Protection (NIRP) there is a special advisory group with experts on radiation protection, reactor safety, radiobiology, meteorology etc to give advice to the local authority. The advisory group collects information on releases, contamination and exposure rates, on weather conditions and other dispersion parameters in order to assess consequences and trends. The group works very closely with the Swedish Nuclear Inspectorate which assesses the technical safety aspects.

The emergency planning area around each nuclear power plant is divided into circular zones with graded emergency planning. The central alert zone extends to 5-10 km, the inner emergency zone to 12-15 km and the radiation measurement zone to about 50 km from the plant.

In the radiation measurement zone measurements by mobile instruments have been prepared to be made along several specified routes. Measurements will be made by the local fire brigade teams. Booklets with information have been distributed to the farmers within this zone. In the inner emergency zone furthermore there are plans for evacuation, there is a network of permanent measurement sites with TLD, iodine tablets have been given in advance to each household and an alert system for telephone alarm signals will reach each home. An information booklet is also

given to each family. In the central alert zone furthermore there is a permanent system with sirens for outdoors alarms.

The emergency plan is triggered into action at one of three levels
a) information; something has happened at the plant that requires special information to the authorities and the public. The safety systems are intact.
b) increased preparedness; something has happened at the plant that is or can be of significance for the safety of the reactor and there is a threat or indication of releases of radioactive material above normal releases. The essential safety systems are intact. The responsible authorities will put the emergency organisation into operation, inform the public and make ready all preparations for further actions (if needed).
c) accident alarm; something has happened at the plant that requires countermeasures outside the plant, abnormal, significant releases are expected, occurring or have occurred, the reactor safety systems do not operate satisfactorily. The alarm system will be activated and the local and central emergency organisations will operate in full.

The levels given above are specified in technical terms in the control room of respective reactor. The general guidance given to the public in the neighbourhood of the plant in case of an accident is to go indoors, close windows and ventilation and listen to the radio for further information and advice. Iodine tablets would probably be used at a rather low level of radioactive iodine release whilst evacuation would only be undertaken in a very serious accidental situation. Evacuation is not warranted below 10 mSv but would be urgent above 100 mSv and no evacuation should occur during releases. The reference doses are those obtained if evacuation is not made. To keep authorities and others in the organisation for emergency preparedness alert, education and exercises are performed at a regular basis, e g a large exercise each four years for each site including all personnel concerned at the site and the local and central authorities.

IMPROVED NUCLEAR SAFETY AND SYSTEMS FOR MITIGATING THE CONSEQUENCES OF AN ACCIDENT

By continual efforts, including technical and administrative means, nuclear safety is improved all the time. By a governmental decision in 1981, the reactors at Barsebäck (in the south of Sweden) have been connected to a large filter system, "Filtra", containing 10 000 m3 of stone, to enable the containment to be vented in case of a serious accidental situation with disfunctioning safety systems and an uncontrolled, increasing pressure in the containment. "Filtra" has been in place and ready for operation since 1985. By accident management and the filtering effect of "Filtra" the expected releases of particulate aerosols will be reduced by a factor of about 1000, and doses warranting evacuation will occur only within close distances up to a few kilometres from the reactor. Because of the large volume of "Filtra", there will also be a considerable delay in the

release of noble gases from the containment leaving more time (several hours) for planning and for carrying out the necessary protective actions.

By the governmental decision in 1981, also the other reactors shall have similar filtering systems with the same requirement on filtering efficiency. They are required to be in place in 1988 at the latest. These filter systems will be of the water-scrubber-type.

LESSONS LEARNED FROM THE CHERNOBYL ACCIDENT AS REGARDS MEASURES AFTER AN ACCIDENT

The major conclusions from the experience of the events in Sweden after the Chernobyl accident are:

- The need of personnel was much higher than planned, because of the prolonged release, the large geographical distribution of the contamination and the extraordinary need of information to the public, between authorities and to the Government.
- Preplanned information, the systems and means for rapid information and the number of qualified informers were quite insufficient. The message given in the information was sometimes contradictory and ambigous, leading to unnecessary anxiety.
- The need of measurements of land contamination, on milk and foodstuffs, on vehicles, on people, etc was much greater than made possible by initially available resources and organisation.
- The alarm functions were insufficient. In Sweden, the first alarm signals came from routine measurements on personnel at the Forsmark power station in the morning of 28 April 1986.
- The responsibility of the various authorities was unclear in this type of situation, when the whole country was contaminated. Because of this NIRP voluntarily took the initial responsibility in many questions belonging to other authorities' jurisdiction such as food restrictions, recommendations on farming, etc. However,these problems were in practice solved within a week or two.
- The wide geographical distribution of released radioactive materials created a lot of international problems that were not planned for, e g as regards information, tourism, trade, transport, intervention levels etc.

CHANGED EMERGENCY PLANNING

Since the Chernobyl accident there have already been several governmental decisions on immediate improvements of the emergency planning and special investigations have been initiated on further appropriate changes of the planning in general and in detail. Since these investigations are still in progress the final result might in some part be slightly different than that reported here.

- The extent and ambition level of emergency planning in counties with nuclear power plants will not change significantly. There are a few additional sirens for outdoor alarms to get a better areal coverage. The level of education and frequency of exercises are quite appropriate.

- There will be emergency planning in all the 24 counties in Sweden although not (with one exemption) to the same extent as in those with nuclear power plants.
- The early warning system will be improved. Since many years there are 25 stationary ionization chambers geographically distributed in Sweden which continuously measure the gamma radiation from air and ground. The number of these will be increased 20-100 % and they will give an automatic alarm to NIRP at a significantly increased radiation level. The number of automatic air-samplers will be increased to about 10 and give automatic warning signals.
- The strategy for measurements will be as follows: In the area outside nuclear power plants 6-8 stationary measuring points in every 60 degrees sector are given in advance distributed out to a distance of about 50 km from the plant. Before and during the release gamma measurements (from about 1 μGy/h) and airsampling are made at the points which are within a 60 degree sector in the wind direction. The reason for using pre-determined measuring points is to obtain an organisation that operates automatically and will identify the plume passage and the limits of the influenced sector area. When the release has ceased the ground contamination is measured along given routes as previously planned. At a distance of 10-20 km there might be continuously measuring gamma instruments recording the average ambient dose equivalent for every 10 minutes. At sea outside the coast limited measurements might be made on a vessel moving in wind direction at a few nautical miles distance. Measurements from airplanes are made to map the ground contamination over large areas. Hand instruments at every municipal and county administration authority (in total about 400) are used to meet local requirements on measurements and information but also to complement the assessment of ground contamination. Milk is measured at dairies by their own staff and measurements on food and water at special qualified laboratories. The possibilities to analyse Pu and hot-particles will be improved.
- The communication system between central and local authorities will be improved by telefax, telex etc.
- The information system will be much improved by an enlarged staff of informers in preparedness, preplanned principles and means for information of the public, manuals to central and local authorities with basic data and guidance how to handle emergency situations, improved medical information, improved education on radiation matters in schools etc. Information will mainly be given by local authorities and backed up by the central authorities on general matters and recommendations.

In conclusion, the emergency preparedness will be much improved particularly to be able to handle a large land contamination from accidents in Sweden or abroad. The extra costs are of the order of 10 milj US $ the first year and 1-2 milj US $ per year the following years including research and development. As has been pointed out above, a large part of the emergency planning is motivated by the risk of an accident at some of the about 100 nuclear power reactors in other European countries and is thus independent of the existence of Swedish nuclear power.

EMERGENCY PLANNING LESSONS LEARNED FROM A REVIEW OF PAST MAJOR RADIOLOGICAL ACCIDENTS

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In examining a range of nuclear accidents from the 1950s to the present that were reported in the literature, the authors have identified a number of contributing factors which affected human judgment during these events. One common thread found in a large number of accidents is the time of occurrence; a second is the adequacy of emergency training.

The data show that events, whether severe accidents or operational incidents, appear to occur more frequently during off-normal hours such as the early morning shift, weekends, or holidays. Accidents seldom occur during the day shift when the full management team and senior operations personnel are present. As a result, those facility employees most expert in coping with the situation may not be available, and the normal chain of command may be disrupted. At several nuclear power plants, it was also observed that new or less experienced technicians are often assigned to night shifts. The lack of experienced human resources and the pressure of an accident situation can have an adverse impact on individuals who are faced with making important decisions.

An in-depth review of the literature conducted for the U.S. Nuclear Regulatory Commission (NRC) by staff members at the Pacific Northwest Laboratory (PNL) [1] determined that human errors generally increase at night, and the chance for error is significantly greater if the worker has already been working four or more hours before midnight. The highest error rates were reported to occur between 3 and 6 a.m. The most efficient work is typically performed during the day. When a person is required to work at night and sleep during the day, both work performance and sleep were found to be degraded.

During a recent annual meeting of the Academy of Behavioral Medicine Research, members of the Harvard Medical School research staff observed that three of the recent major disasters--Three Mile Island, Bhopal, and Chernobyl--all occurred at night. The concern expressed by behavioral scientists was not only that

lowered alertness might cause accidents at nuclear facilities during off-hours, but that the concurrent human ability to detect and correct the problem in a timely manner might also be reduced [2].

Dr. C. A. Czeisler of the Harvard Medical School, Department of Medicine, has conducted extensive studies of the effects of rotating shift work schedules [3]. Dr. Czeisler observed that workers on night or rotating shifts experience adverse consequences because their circadian rhythm and physiological functions, such as body temperature, hormone secretions, cell division, and antibody formation, vary over a 24-hour period. Whenever a worker's normal wake/sleep schedule is interrupted, a mismatch occurs between the body's ability and the demands placed upon it in the workplace. Stress, gastrointestinal disorders, low morale, high rates of accidents and illness, as well as low productivity, result from this mismatch. Sleep-deprived shift workers often experience involuntary lapses of wakefulness; while they appear to be awake, they may actually be drifting in and out of sleep. Because of these lapses, the ability to respond to warning signals or lights may be impaired [4,5]. Czeisler reported in field studies of 1500 workers at a number of industrial facilities, that over 55% of the workers admitted to "nodding off" or falling asleep on the job during any given week. At the Fast Flux Test Facility at the Hanford Site in Washington, PNL researchers are currently evaluating the effects of a 12-hour-a-day rotating shift. The shift was adopted to help reduce the attrition of operators.

Another common thread is that, no matter how well emergency scenarios are developed and emergency planning exercises are conducted, no scenario can adequately simulate an actual emergency [6]. During an emergency, there is little time to consult the emergency operating plan to decide on a response. Response performance will reflect the quality of the training imparted to emergency response personnel from plan-and-procedure implementation, as well as from periodic drills and exercises.

How individuals respond to an actual emergency is another uncertainty. In Japan, analysis of safety evaluations performed at commercial nuclear installations revealed that an appropriate response of operators to an accident could not be expected for at least 10 minutes after they became aware of the emergency situation [7]. Part of this delay is due to the time required to effectively analyze the situation. Another part of the delay may be due to the reluctance of the operators to acknowledge that they have an emergency. Irrespective of the number of emergency drills and exercises conducted, the real situation may reveal flaws in the emergency plan and response. Therefore, everything that borders on an emergency at a facility should be treated as an emergency.

The authors of this paper have examined over fifty accidents and derived the most important lessons learned from each accident. Indeed, a review of nuclear facility accidents since the 1950s revealed that human error contributed to a majority of the incidents. Most of these accidents began during night shifts, on

weekends, or holidays. This suggests the possibility that fatigue and/or the absence of experienced personnel could have been among the causes of the incidents. Human error during off-normal hours was clearly involved in the following representative cases:

- Windscale Works of British Nuclear Fuels Accident - 7:25 p.m., Monday, 10-7-57
- SL-1 U.S. Army Reactor Accident - 9:01 p.m., Monday, 1-3-61
- Recuplex Nuclear Criticality - 10:59 a.m., Saturday, 4-7-62
- Wood River Junction Criticality - 6:00 p.m., Friday 7-24-64
- Browns Ferry Nuclear Power Plant Fire - 12:20 p.m., Saturday, 3-22-75
- Three Mile Island Unit 2 Nuclear Power Plant Accident - 4:00 a.m., Wednesday, 3-28-79
- Chernobyl Nuclear Reactor Accident - 1:23 a.m., Saturday, 4-26-86

Contributing factors to the earlier incidents were perhaps deficiencies in the design of facilities or equipment involving this new technology. Later accidents, however, involved facilities and equipment that had become "safer" as a result of many redundancies built into the operating systems to safeguard the facility and human health. The more recent accidents appear to be the result of errors on the part of humans trying to respond to unusual events involving this increasingly complex technology. Compounding this problem was the fact that many of these accidents may have occurred when the most qualified and best trained personnel were not on duty.

Several lessons have been learned from an analysis of these accidents:

- Nuclear accidents can have worldwide impact on the public, governmental agencies, and the nuclear industry.

- Rotating of operating personnel to work during off-normal hours requires careful planning, taking into consideration the body's natural rhythm, to maximize performance.

- The response of an individual to an emergency cannot always be predicted. Also, operating personnel require some time to respond to an emergency. Ultimately, the major response will be based on the quality of the training imparted to emergency response personnel.

- Emergency preparedness instrumentation capable of assessing high radiation fields is required. This equipment must be operable in a wide range of environments.

- Remotely-operated retrieval and surveillance equipment is essential during emergencies. Control rooms should be designed with optimum consideration for human factors.

- Potential accidents must be anticipated through formal safety studies. Exercise scenarios should focus on higher probability accidents as well as worst-case accidents.

- Regularly scheduled, rigorous emergency preparedness exercises are needed. These exercises must include objective, post-exercise critiques and a commitment from facility management to correct any deficiencies identified.

- Improved emergency response training and retraining is needed for plant personnel.

- Emergency responses require well-directed coordination.

- Rigid administrative control of fissile materials is essential.

- Nuclear facilities should be sited where population density is low.

ACKNOWLEDGMENTS

This work was performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

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DECONTAMINATION STRATEGIES IN URBAN AREAS AFTER NUCLEAR ACCIDENTS

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1. INTRODUCTION

In the case of an accident in a nuclear power plant the release of a higher amount of activity may lead to high surface contamination. This causes high background radiation over a large area with dose rates of more than 10 µSv/h for many months. Such dose rates are too high for people to live there permanently. On the other hand, the cordoning off of an area of some hundreds of square kilometres for an extended period is not acceptable for social, economic and political reasons, especially in densely populated and highly industrialized regions. Therefore, decontamination procedures must be provided to reduce the original contamination by one or more orders of magnitude. Around Chernobyl ploughing of the soil and scraping of surfaces like roads and pavements were the main decontamination methods apart from natural decontamination. But in urban areas there is still a need for other decontamination methods which leave the surfaces practically intact. The efficiency of such methods is investigated in our research programme, supported by the Federal Ministry of the Environment, the Protection of Nature and Reactor Safety. First results are reported in this paper.

According to previous experience from accidents in nuclear power plants, caesium 137 is the most relevant radionuclide for people at risk due to its long half-life and the amount of emitted radioactivity. In the case of the accident at Chernobyl, about 10^{18} Bq Cs 137 were released. Cs aerosols with mean diameters of roughly 0.5 µm have been observed far away from the plant and with mean diameters of 1 mm nearby (1). The chemical composition of the particles is not known exactly. There are some indications that it might be CsI, CsOH or other soluble compounds (2).

2. NATURAL DECONTAMINATION

The contamination level after the Chernobyl accident offered a unique opportunity to study subsequent natural decontamination processes on urban materials, especially on concrete which is the material of main interest in this paper. These investigations show that 3 different time-dependent phases can be distinguished: Run-off, short-term weathering and long-term weathering. Run-off proceeds within a few hours and the decontamination efficiency during this phase greatly depends on the weather situation (3). The short-term weathering corresponds to a period of roughly 80-100 days, during which 30 % of the nuclides was removed. The decontamination of the residual activity by long-term weathering can be approximately assessed over a period of several decades (4).

The natural decontamination processes by weathering were investigated in an experiment using concrete plates with basalt splits in the surface layer used as paving stones for 4-5 years. The plates were contaminated with Cs 137 in aqueous solution by a spraying device which allowed a uniform irrigation. The activity concentration on the plates was 41 Bq/cm^2. The contaminated plates were exposed to open-air conditions last summer when precipitation was relatively high. Fig. 1, Curve 1 shows that the contamination decreased to 50 % after 168 days and 365 mm of rain.

The deviation of results for all experiments was low. Every decontamination experiment was undertaken using 5 plates which were treated in the same manner but independently. The deviation of the parallel results was lower than 25 %. Therefore, the reproducibility of the experimental data is good.

In another experiment some of the contaminated plates were sprayed with demineralized water 24 h and 15 days after contamination. The results are shown in Fig. 1, Curves 2 and 3 respectively. We recognize that the decontamination efficiency of demineralized water is very low, lower than that of rain water, and that the efficiency is lower the longer the contamination could influence the surface without any disturbance.

3. ARTIFICIAL DECONTAMINATION

For artificial decontamination the following methods have been used:

- Mechanical removal by fire hosing (4), shot blasting (5) and sand blasting (6)
- Cleaning tests with hydrochloric acid and steam (6)
- Decontamination tests with ion exchange (6)

Of these studies, the only efficient method is the exchange of cations, for example Cs^+ and NH_4^+ cations. A large-scale experiment (6) shows a good decontamination efficiency with NH_4NO_3 solution with increasing spraying time. After a spraying time of 3h with 0.05 m NH_4NO_3 solution the smooth cement surface of coping stones responded well and about 80 % of the caesium was removed.

Our results from a decontamination experiment with the concrete plates mentioned above and ammonium nitrate solution (0.2 m NH_4NO_3) are shown in Fig. 2. The activity concentration of Cs 137 was 81 Bq/cm^2. Curve 1 gives the results of the decontamination procedure by spraying the decontaminant on the plate. Subsequently, the experiment was repeated and the plate surface was simultaneously brushed. The results are given in Curve 2 of Fig. 2. It can be seen that after spraying 20 l of solution on to a plate about 90 % of the caesium contamination was removed. Simultaneous brushing reduced the amount of solution required to 70 %.

This shows that the most important influencing parameters are the physicochemical interactions between the deposit and the surface due to the choice and the amount of the decontaminant and the duration of the decontamination procedure. Material parameters like the surface structure and the age of the material are important too. But they are not treated here.

4. SUMMARY AND OUTLOOK

These preliminary results are of some practical interest under the following aspects:

- Natural decontamination on urban structures like concrete is not very efficient. A decontamination factor of 2 is only reached after more than 100 days.
- Decontamination by spraying with demineralized water is not very efficient, either. Decontamination by spraying with cation exchanging solutions gives the best results but the duration of this procedure is still very long and the amount of decontaminant required is high.
- Decontamination by spraying must be started immediately. Any delay reduces the decontamination efficiency.
- Additional brushing of surfaces improves the decontamination efficiency only insignificantly.

It seems that decontamination by spraying cation exchangers could become a practicable method for urban surfaces if it were possible to reduce the time effort. Especially ammonium nitrate seems to be a very acceptable decontaminant because it is effective in low concentrations, available in large quantities, environmentally compatible and reasonably priced. Further investigations will prove whether modifications of the decontamination strategy are advantageous, such as high pressure, variation of concentration and temperature and primary treatment of the material.

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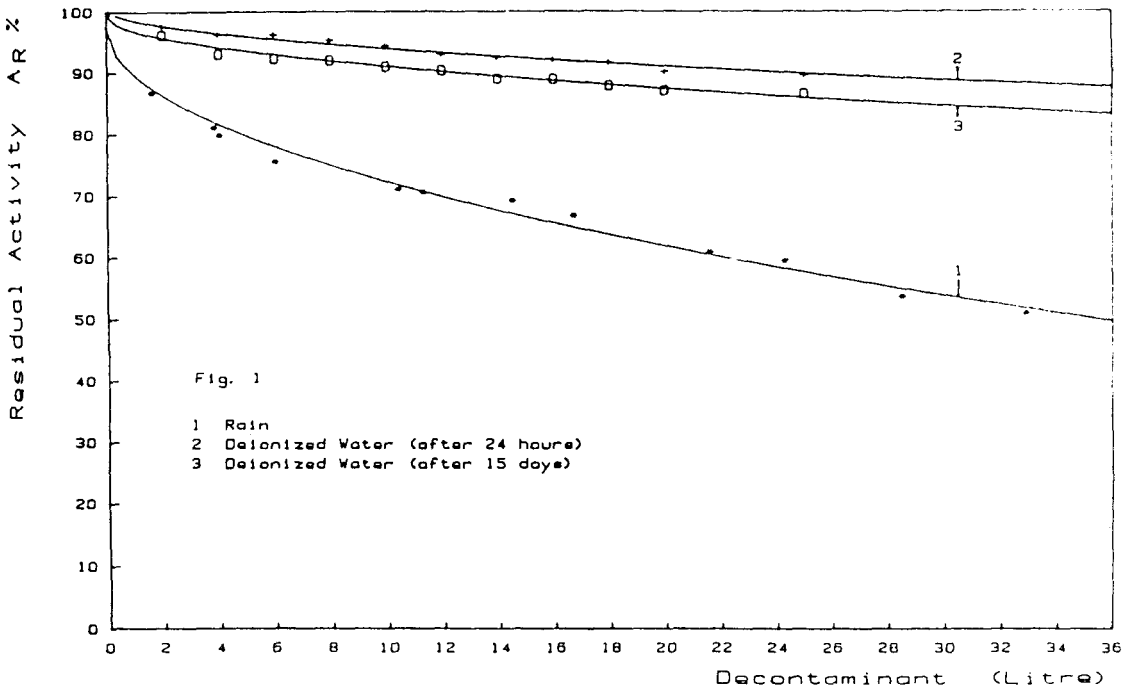
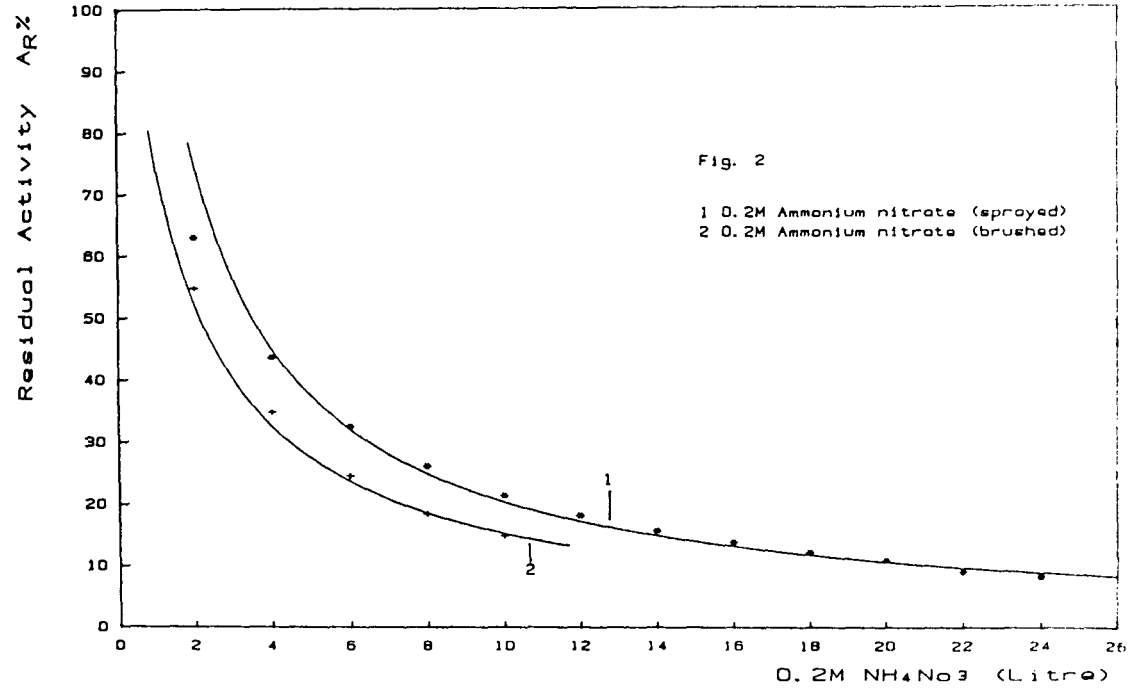
Decontamination Experiments



Decontamination with Ammonium Nitrate



NEW INTERNATIONAL LAW CONCERNING NUCLEAR ACCIDENTS AND RADIOLOGICAL EMERGENCIES

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ABSTRACT

In 1963 the Scandinavien States and the IAEA signed the "Nordic Mutual Emergency Assistance Agreement in Connection with Radiation Accidents". It was the first multilateral agreement in this special field of the international atomic energy law. The IAEA tried to encourage other national governments to agree also such assistance agreements and elaborated in 1966 four model agreements as multilateral or bilateral treaties. But at that time no state followed the recommendations of the Agency. Later on some Middle European governments agreed bilateral treaties on the mutual assistance and the notification in cases of radiological emergencies.

After the Chernobyl accident in April 1986 a spontaneous reaction led to the initiative of some governments and of the Tokyo Economic Summit and to the decision of the IAEA-Board of Governors to prepare two worldwide agreements on mutual assistance and early notification in cases of nuclear accidents on the basis of the IAEA recommendations published in 1982 as INFCIRC/310 and 321.

The author, member of the German delegation of Government experts at the IAEA meeting in July/August 1986 at Vienna, gives a survey on the two conventions which has been adopted by the special session of the IAEA General Conference on September 26, 1986:

- Convention on early notification of a nuclear accident;
- Convention on assistance in the case of a nuclear accident or radiological emergency.

The following legal problems will be treated: Scope of application; notifiction and information to be provided; competent authorities and points of contact; functions of the IAEA and other international organizations; international assistance in the establishment of radiation monitoring systems; procedure and provisions for mutual assistance; direction and control of assistance; reimbursements of costs; privileges and immunities; claims and compensation (liability).

Finally a report is given of the status of signatures, ratifications and coming into force of the two Vienna Conventions and of other treaties signed in the meantime.

SITE EMERGENCY DATA INTERPRETATION

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Harwell Laboratory operates a number of nuclear facilities including materials testing reactors, accelerators, post irradiation examination facilities & remote handling radiochemistry laboratories. The range of radionuclides that could be released in quantities sufficient to produce a radiological hazard can be grouped into three broad categories for measurement purposes: fission product noble gases, volatile materials (radio-iodines) & refractory particulates such as Pu, Cs & Sr compounds.

Emergency monitoring systems include individual alarms for appropriate buildings, TEMPO - a site & perimeter network of GM counters linked by radio telemetry to the control room, a source dispersion/dose prediction code run on a dedicated microcomputer (Reference 1), & a dedicated radiological survey vehicle. Additional Health Physics vehicles are available to carry portable monitoring equipment & to collect samples for laboratory analysis. Routine environmental monitoring systems, eg High Volume Air Samplers & TLD's are also incorporated into the emergency monitoring plan.

In the event of a suspected or genuine release of radioactivity a source term would be rapidly estimated backed up by measurements taken both on & off site. The dedicated survey vehicle is equipped with a range of dose-rate meters, contamination meters & twin Rotheroe & Mitchell L50 air samplers which draw air through ports on separate sides of the vehicle. Air sampling analysis is carried out in the vehicle using a beta castle fitted with a Nuclear Enterprises BP4 probe & an alpha drawer assembly, both of which are connected to a Harwell 6000 series counting system. A mobile TEMPO station is also carried. This can relay gamma dose rate data directly to the control room from within the vehicle or alternatively it can be removed & placed in any desired location, provided that the telemetry link can be established.

Fission product noble gases may be detected & measured using a gamma dose rate meter at concentrations below those at which intervention is desirable. Volatile materials from the initial cloud or resuspension are sampled using twin Maypacks (Reference 2), containing a combination of GFA filter papers, charcoal filter papers & charcoal granules. The complete Maypacks or the components are assessed in the beta castle. It is assumed that all elemental iodine is collected on the charcoal filters & that any methyl iodine is absorbed by the granular charcoal. Particulate airborne activity is collected on a glassfibre filter paper which can be removed from the pack for beta & alpha analysis. Ground deposition is initially assessed with doserate & contamination meters.

A key feature of the system is the Survey Report Form partly reproducd in Figures 1 and 2. The form contains a further two graphs for assessing integrated dose over 14 days from alpha and beta/gamma ground contamination. In addition there are two nomograms for calculating airborne activity in Bq/m^3 from air sample volume and counter readings for alpha and beta/gamma sources, tritium dose calculation data, a table of Emergency Reference Levels of dose and a calculation section which can be used instead of the nomograms. The three data display pages include information on time and location of a survey to allow them to stand alone if required, eg for transmission by fax. Identical versions are used by the survey teams and the control room, although the former would only use the first page unless providing assistance to another site.

The system relies on the following assumptions:

1. Emergency Reference Levels are the lower values taken from Reference 3.
2. Airborne & ground beta/gamma activity is all I-131.
3. Airborne & ground alpha activity is all Pu-239, Class W.
4. For airborne activity calculations the local concentration is constant for the duration of the release & the expected maximum release duration is 4 hours.
5. The duty health physicist understands the limitations of these assumptions & will use professional judgement & all other information available to analyse the situation.

The sensitivities of the source term assumptions have been tested against computer predictions for the Harwell reactor fuel inventory for standard irradiation & various cooling times. The results show that for inhalation of fission products released from a 6-160 day cooled fuel pin the form will give a dose assessment correct to within a factor of two. For airborne alpha activity the form will overestimate by a factor of 2.6 if the material is Class Y rather than Class W. The system is designed to ensure that the initial response assumes the most pessimistic interpretation of the field measurements. The validity of the initial assessment is backed up with laboratory analysis of collected samples & the experience of the full emergency response team.

The authors wish to acknowledge the assistance of R Fox & H C Orchard of the CEGB & the work described in reference 2.

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3. White I F. NRPB Emergency Data Handbook. NRPB R182, March 1986.

FIGURE 1

HARWELL SURVEY REPORT

Reporting form for survey crews & control room radio operators. The call signs allow data to be transmitted as numbers without further clarification. Similar formats are being adopted by a number of organisations within the UK, which will improve the existing arrangements for mutual assistance between sites.

DATE	**HARWELL SURVEY REPORT**
SURVEY VEHICLE CALL SIGN	

Location (sample point or grid reference)		ALPHA		
Air sampling start time		BRAVO		
GAMMA DOSE RATE	Outside at 1 metre above ground	CHARLIE		micro Sv/hour
GROUND CONTAMINATION MEASUREMENTS	1828, alpha background (inside van)	DELTA		counts/sec
	1828, alpha ground contamination	ECHO		
	1828, beta background (inside van)	FOXTROT		
	1828, beta ground contamination	GOLF		
AIRBORNE BETA/GAMMA ACTIVITY	Background count time	HOTEL		seconds
	BP4 beta/gamma background	INDIA		counts
	Air sampling stop time	JULIET		
	Air sample volume	KILO		litres
	Maypack 1 (10 secs count)	LIMA		counts
	Maypack 2 (10 secs count)	MIKE		counts
AIRBORNE ALPHA ACTIVITY	Alpha counter background	NOVEMBER		counts
	Filter paper 1(60 secs count)	OSCAR		counts
	Filter paper 2(60 secs count)	PAPA		counts
AIRBORNE TRITIUM	Scintrex reading	QUEBEC		MBq/m^3
AIRBORNE RESULTS	Beta/gamma	WHISKY		kBq/m^3
	Alpha	X-RAY		Bq/m^3
GROUND CONTAMINATION RESULTS	Beta/gamma	YANKEE		kBq/m^2
	Alpha	ZULU		Bq/m^2

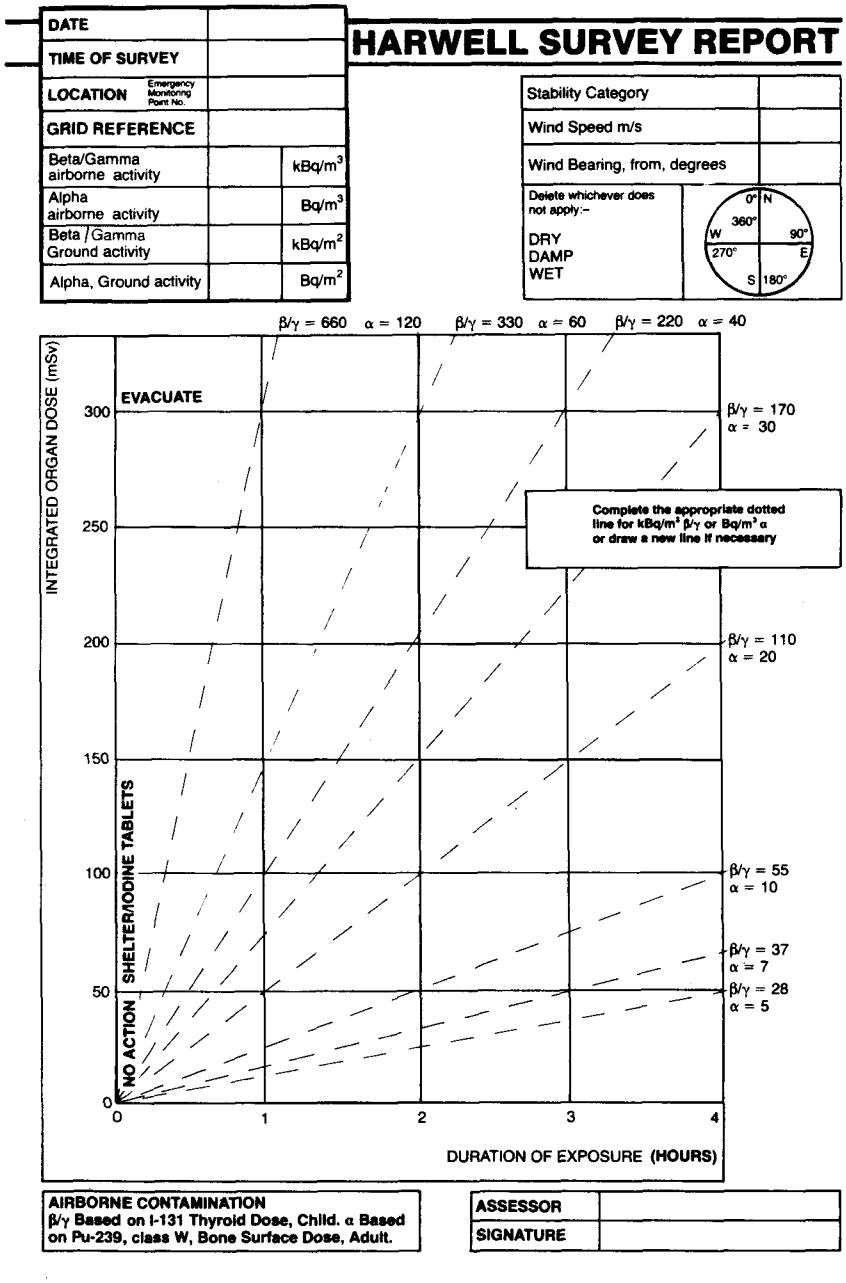
Survey team	
Signature	

DO NOT REPORT WHISKY, X-RAY, YANKEE AND ZULU
(WHEN PROVIDING ASSISTANCE TO ANOTHER SITE REPORT ALPHA, BRAVO, CHARLIE, WHISKY, X-RAY, YANKEE AND ZULU ONLY)
ISSUE 1, APRIL 87

FIGURE 2

HARWELL SURVEY REPORT

Interpretation graph for airborne Beta/Gamma & Alpha activity.



THE PREVENTION OF ACCIDENTS IN INDUSTRIAL GAMMA RADIOGRAPHY

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ABSTRACT

In France, in 1979, after the accident at Montpellier, due to a source of Iridium 192 which fell from an apparatus and was picked up by a workman, the "Institut de Protection et de Sûreté Nucléaire" (Nuclear Protection and Safety Institute) carried out a detailed investigation of past incidents in radiography. It was considered whether other techniques using non-destructive control were equivalent and whether X-rays could be used instead of radioactive sources. It became apparent that gamma radiography is indeed necessary. In order to put this technique to use in adequate conditions of safety, an analysis of risks was therefore undertaken. This analysis was brought to bear on the following topics : particular working conditions, staff training, equipment reliability, source selection and lessons learned from incidents. It has also been established that transport of apparatus by road takes place very frequently. A special set of regulations was therefore implemented for this type of transport. It was found necessary to inquire into the causes of those accidents which are more liable to occur than others, and whose degree of seriousness is higher. 90 % of the apparatus are portable and fulfill three functions : as containers for storage, as gamma radiography cameras on the work site, and as packagings for transport. Many measures have been taken. However, the ultimate responsibility lies with the operator who must, at any time, know the location of the source and make sure, at the end of the process, that the source has been safely brought back into storage position inside the apparatus.

SPANISH PRACTICE AND EXPERIENCE IN THE IMPLEMENTATION OF NUCLEAR EMERGENCY PLANS

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ABSTRACT

In Spain, as in other countries, the emergency plans associated with nuclear installations have known three well defined periods: from the beginning of the nuclear national program (in Spain about 1956) to the Three Mile Island-2 accident (March 1979; from TMI-2 to Chernobyl-4 event (April 1986); and from Chernobyl up to now.

The second period has been the most fruitful for development of nuclear emergency plans in Spain; during that period a Basic Plan for nuclear emergencies at national level was established. This Basic Plan follows the international recomendations, mainly from IAEA, EURATOM and lessons learned from Chernobyl accident.

In the paper the criteria and principles on which the Basic Plan is based are firstly described. Organizational and operational aspects are then discussed; the extend these aspects is brought into play is determined by the type of abnormal event which occurs in the facility; since the evolution of this event can not be exactly predicted, there must be enough flexibility in the organizational and operational aspects so that it can be adapted rapidly and effectively to the circustances. Another section deals with the protection measures as a function of intervention (or reference) levels. For planning purposes such measures are taken gradually up to 3, 5, 10 and 30 km. depending on the severity of the situation.

The Basic Plan has been adapted for each of the provinces where a NPP is located; this adaptation takes into account the particular organization and resources of each province, so there are Plans at provincial levels; besides and derived from these Plans, each municipality within a radius of 10 km from the NPP also has its own Municipal Plan for nuclear emergencies.

Finally, details are given of the Training Courses for specific teams and Information Conferences for the population which are applied in Spain and also the accumulated experience with the implementation of the nuclear emegency plans.

RADIATION RISKS OF LARGE SCALE NUCLEAR ACCIDENTS - A CASE STUDY

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ABSTRACT

The object of this paper is the estimation of the radiological consequences of the latest 1000 MW Kozlodui reactor, a Soviet type PWR in Bulgaria, to the population of Greece. To estimate these consequences a series of severe accident scenarios are utilized. The results of the analysis indicate that under the conservative assumptions adopted, the radiological consequences of the most severe accidents considered would be non trivial. The magnitude of the potential effects from the accident scenarios analyzed is such that multinational emergency planning for nuclear installations may be required, even when nuclear power stations do not lie in proximity to national borders.

INTRODUCTION

The experience gained from the Chernobyl accident has shown that nuclear accidents may involve releases of radioactive materials to the environment, which extend to large distances. It is therefore appropriate to assess the radiological consequences to the population of Greece of a few representative severe nuclear accidents of the nuclear power station(NPS) at Kozlodui, Bulgaria, which is located closer to Greece than any other similar station. The Kozlodui site with 4 operating PWR units of 440 MW and 1 PWR unit of 1000 MW, is located near the northwestern borders of Bulgaria near the river Danube, at a distance of 225 km from the northern borders of Greece. In the present analysis attention is focussed on the larger unit, which in the improbable event of a severe accident would produce the most adverse effects.

The accident consequence calculations were performed using the CRAC.GAEC code, a version of code CRAC2[1,2]. The model describes the progression of the radioactive cloud released from the reactor building and predicts its interaction with and influence on the environment and man.

ACCIDENT RELEASES AND INPUT DATA

Reactor accident consequence calculations require the radionuclide inventory of the reactor under consideration. Since detailed data of VVER-1000 core inventories were not available, the simplifying assumption that they will be similar to the inventory of a standard PWR[3] multiplied by the ratio of the corresponding reactor thermal powers was made. The larger unit of 1000 MW has a thermal power of 3000 MW and a unit efficiency of 33.3%. The standard PWR inventory corresponds to an end-of-cycle equilibrium inventory of a 3412 MWt PWR, calculated with a burnup of 33000 MWd/MTU[3].

In order to delineate the upper bounds of the spectrum of consequences, the analysis is focussed on the consequences of four serious hypothetical PWR accidents, or release categories, namely the PWR-2, PWR-3, PWR-4, and PWR-5[4], which we assume that are also applicable to VVER-1000 reactors. Therefore the same release fractions applying to these accident categories are used for VVER-1000. The consequence assessment requires a large number of input data. Some of these data are taken identical to those of the reference case of the CRAC2 User's Guide document[1], while the rest are calculated specifically for the cases assessed.

The meteorological record used in the consequence model consists of 8760 hourly observations of wind speed, atmospheric stability and accumulated precipitation, and corresponds to a typical meteorological year of Athens. The associated wind rose is not taken into account, but a constant southward wind direction is assumed since the aim is to estimate the upper bound of the consequences, which obviously take their maximum values for Greece when the radioactive cloud is directed towards the centre of the country. In order to represent all possible weather conditions about 580 weather sequences are sampled from the meteorological record. The weather category sampling technique utilized[5] takes into account effectively the very low probability sequences, which have potentially high consequences. In addition we assume that no emergency measures are taken during the progress of an accident. For the estimation of latent health effects the BEIR method is used.

RISK AND LATENT HEALTH EFFECTS

The consequences resulting from Kozlodui's four hypothetical accidents would be due to early exposure, which includes direct irradiation by the passing cloud, exposure from inhaled radionuclides and exposure to deposited radionuclides, and to chronic exposure, which includes exposure to groundshine from contaminated ground, inhalation of resuspended radionuclides and ingestion of contaminated foods. The results are presented either as dose or cancer risk curves for exposed individuals versus distance from the Kozlodui site, or as complementary cumulative distribution functions (CCDFs) for latent health effects and collective exposure. The health effects considered include early deaths and early injuries, thyroid effects and latent cancer fatalities. Although the postulated releases would be very serious, the distance of 225 km of the northern border of Greece from the Kozlodui site prevents the manifestation of early deaths and early injuries among the Greek population. The effects that were manifested were only latent cancer fatalities and thyroid nodules as expected.

In Figs. 1-2 the variations of the mean and peak acute bone-marrow dose and the cancer risk from initial exposure versus distance from the reactor site are depicted for the four accident releases. Figs. 3-4 show the collective dose of the whole population of Greece as a CCDF and the latent health effects both resulting from the total exposure for all four cases analyzed. A summary of all latent health effects and the collective exposure is shown in Table 1.

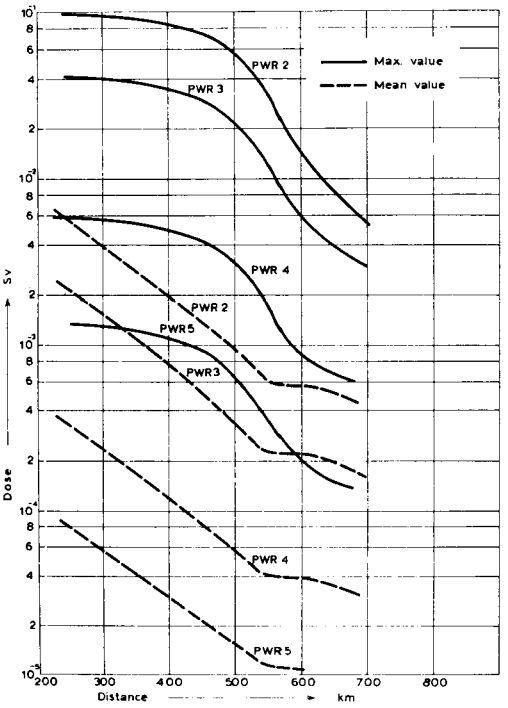

Fig. 1. Acute Bone Marrow Dose versus Distance from Kozlodui Site

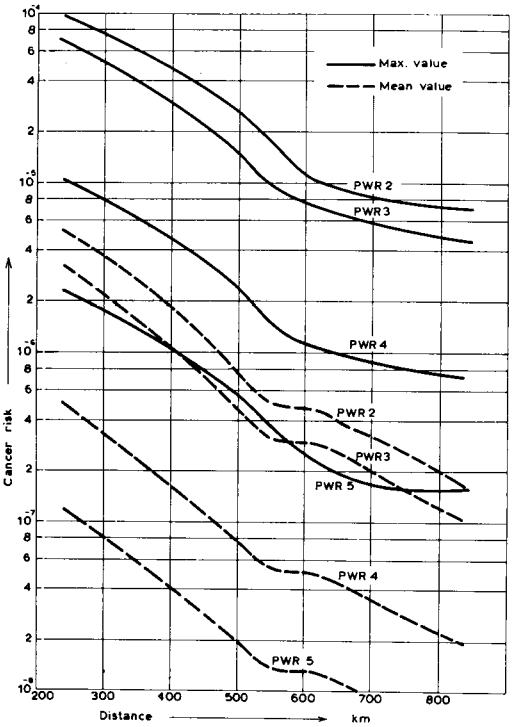

Fig. 2. Cancer Risk versus Distance from Kozlodui Site

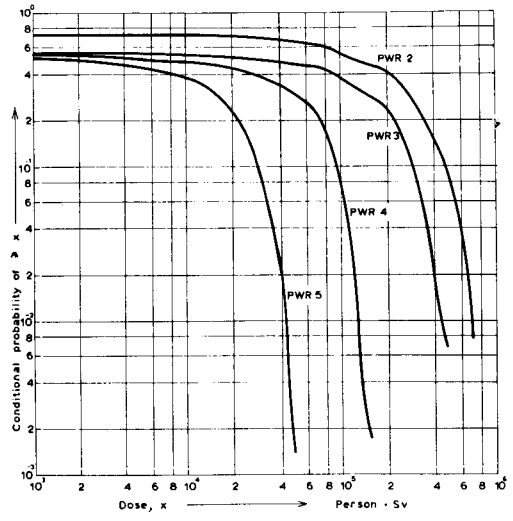

Fig. 3. Whole Body Collective Dose CCDF

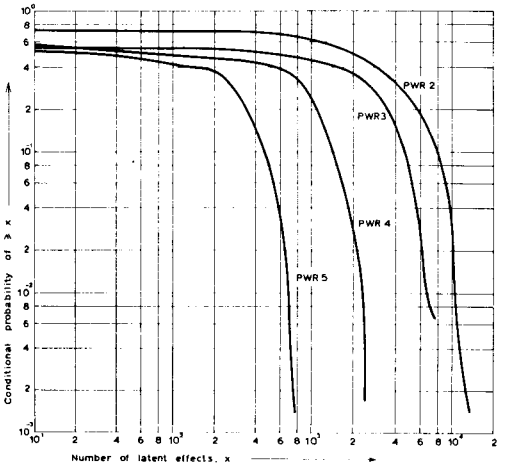

Fig. 4. Whole Body Latent Effects CCDF

Table 1. Summary of number of latent effects - Total exposure

Effect	Mean Values				Peak Values			
	PWR2	PWR3	PWR4	PWR5	PWR2	PWR3	PWR4	PWR5
Whole body	2830	1560	514	156	13200	7620	2460	780
Thyroid	2010	1000	391	199	11500	5960	1960	883
Leukemia	596	317	101	30.3	2850	1560	485	151
Lung	519	324	94.4	28.2	2410	1680	459	141
Breast	623	337	107	32.1	3030	1680	516	161
Bone	295	150	45.6	13.0	1410	747	221	64.4
GI tract	284	171	63.2	20.0	1180	782	300	99.2
Other	632	342	109	32.6	3070	1700	523	163
Whole Body Collective Exposure (10^3 person·Sv)	179	99.1	32.6	9.89	839	484	156	49.4

CONCLUSIONS

The results presented in this paper have been obtained under a series of conservative assumptions, such as wind blowing continuously southwards, most serious accident scenarios at the larger unit of the NPS etc., and are non-trivial. One must note however that if for example the wind direction was probabilistically treated, and less severe, more realistic scenarios were analyzed, the expected consequences (mean values), would be reduced by 2-3 orders of magnitude and then they would be minimal. Under the severe scenarios adopted and in the totality of 9746000 inhabitants of Greece some non-trivial effects would result during long periods after such accidents. The magnitude of the potential effects from the releases analyzed indicate that multinational emergency planning for nuclear installations, may be required even when nuclear power stations do not lie in proximity to national borders.

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ACKNOWLEDGEMENTS

This work was supported partially by the C.E.C. 1985-1989 Radiation Protection Programme.

OFF-SITE RISK STUDY FOR THE THREE OPERATING NUCLEAR POWER STATIONS IN TAIWAN

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ABSTRACT

A systematic study on the risk resulting from postulated accidents were performed for all the three operating nuclear power stations (NPSs) in Taiwan: Chinshan (CS) NPS/2x1775 MWt BWR, Kuosheng (KS) NPS/2x2894 MWt BWR, and Maanshan (MAS) NPS/ 2x2775 MWt PWR.

The study is based on source terms reported in WASH-1400 (1975) NUREG-2239 (1982), IDCOR program (1984), and NUREG-0956 (1986). For the KS NPS, additional study was made with the release categories defined by the plant specific probabilistic risk assessment (KS PRA, 1985).

Major part of the consequence evaluation was performed using the plume model based computer code CRAC2. Sensitivity studies were made for key input parameters. Calculations using the puff model based computer code EDM were also implemented for selected cases. The EDM results, with the effect of complex terrain taken into account, can be used to form correction (or reduction) factors for the CRAC2 results.

The calculated results include conditional values, expected values, and/or complementary cumulative distribution functions (CCDFs) of various concentrations, doses, risks, and health effects. The diminution in risk with improved estimation of source terms is discussed in detail.

Emergency planning for the three NPSs was commented on the basis of protective action guide (NUREG-0396, 1978) and exposure risk guideline (IDCOR, 1984). Optimum evacuation schemes for the three NPSs are proposed based on over two-hundred CRAC2 runs. Parameters considered include delay time/warning time, evacuation speed and distance, protective countermeasures, and weather conditions.

FIRES INVOLVING RADIACTIVE MATERIALS
TRANSFERENCE MODEL . OPERATIVE RECOMMENDATIONS

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ABSTRACT

In all aspects related to the nuclear activity, the occurrence of an explosion, fire or burst type accident, with or without victims, is directly related to the characteristics of the site.

The present work analizes the different parameters involved, describing a transference model and recommendations for evaluation and control of the radiological risk for firemen.

Special emphasis is placed on the measurement of the variables existing in this kind of operations.

PROBLEM PRESENTATION

Fires involving radiactive materials, make face firemen to a radiological risk which is added to the normal risks existing in the more conventional fires that are defined e.g. by the following parameters:

- Type of the stored materials: chemical and physical characteristics.

- Layout and constructions materials of the place: dimensions, openings, access, etc.

- Atmospheric conditions: pressure, temperature, wind speed and direction, etc.

TRANSFERENCE MODEL

For the elaboration of the present model, we took into account the following frame of reference (not analizing the transference ways to the public).

The radionuclide can be:

- spread into the air, without chemical change due to heating of the substance;
- burned, generating smoke and aerosols that will spread on the media;
- spilled over the floor and/or combustible materials;
- dragged, by equipment and substance used for extinction; and
- oversuspended on air from floor and extinguishing substance.

Also:

- The conbustible material can be contaminated by the deposit produced from air and from the extinguishing substance and the air is contaminated by combustion and the extinguishing substance by dragging.
- The extinguishing substance is contaminated by washing the radionuclide dispersed in the air, in the floor, or on combustible material and extinguishing equipment.
- The extinguishing equipment is contaminated due to the radiactive air deposit, through extinguishing substance and through floor dragging.
- The floor is contaminated through radiactive air deposit, through extinguishing substance and extinguishing equipment.

Then the firemen can be:

- superficially contaminated by direct contact with the radionuclide;
- internally contaminated by inhalation; and,
- externally contaminated by dragging of radionuclide which contaminates the combustible material, to the equipment used for the extinction, to the extinguishing substance and the floor.

And additionally they can receive a dose due to external irradiation from the radiactive source and/or receive a dose by external irradiation when they go into the radiactive cloud and because the contaminated floor.

OPERATIVE RECOMMENDATIONS

These operative recommendations will have as main objetive to control the emergency reducing the dose due to internal and external contamination and the dose due to external irradiation up to a level as low as it will be reasonably achievable.

As a first step, with the pourpose of personnel's control, if there is radiation affected personnel and to avoid that people that will not operate in the emergency enters into the place, the emergency zone will be physically separated from the rest of the instalation (e.g. with a barricade).

To limit the external contamination the following provisions will be taken:

- To avoid the physical contact with the elements of the disaster place, specially with the radiactive source.
- To avoid throwing the extinguishing substance over the radiactive source.
- To control the spilling of the extinguishing substance.
- To realize tasks of decontamination of personnel and equipment when the operative has finished. For this it is convenient to mantain one fireman free of any contamination (this may be the fireman in charge of the fire pump since he does not work into the emergency zone) then he will be responsable for the decontamination activities.

Also to minimize the dose by internal contamination and as the only possible way is inhalation, it is mandatory to use the breathing autonomous apparatus.

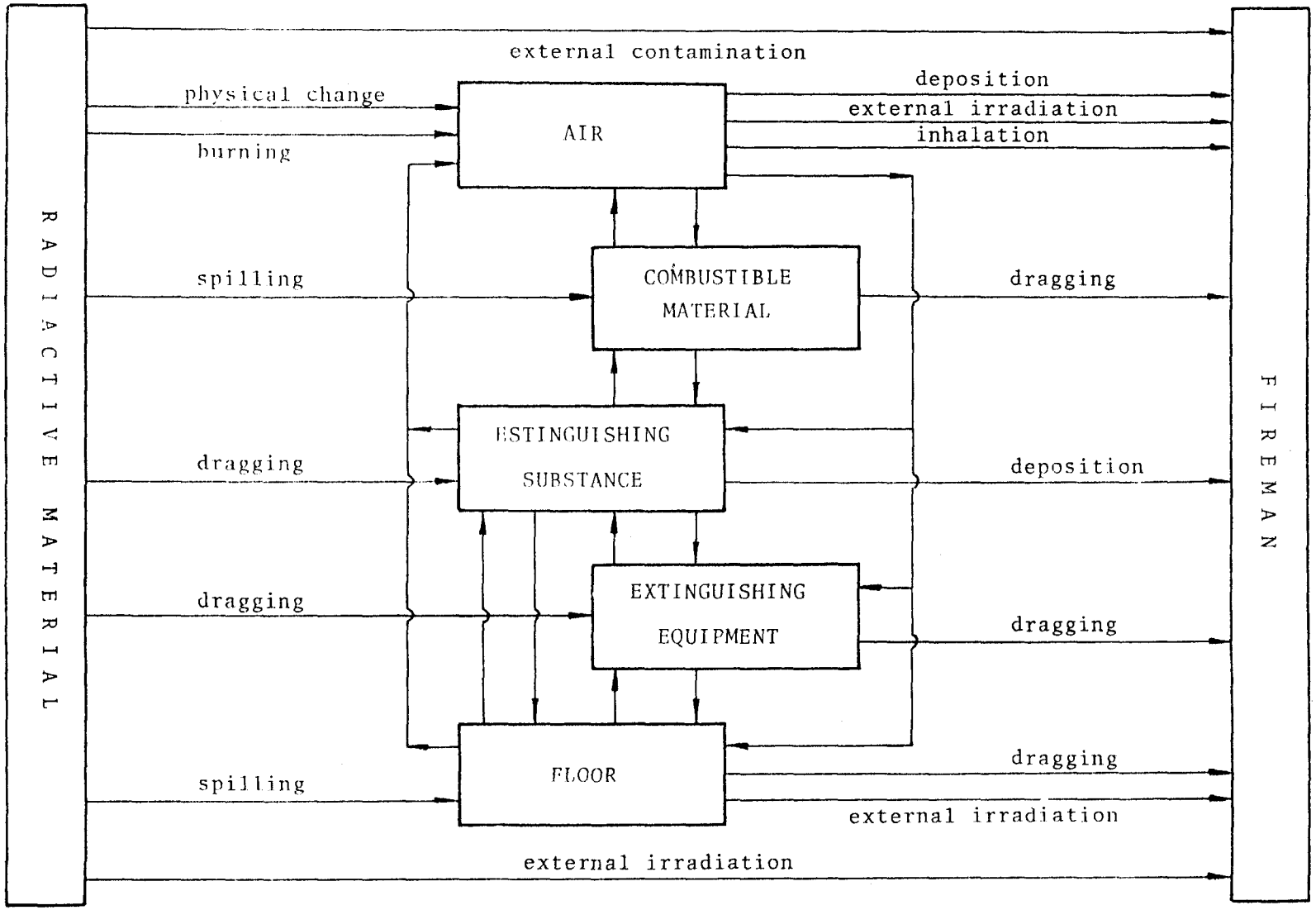
With reference to external irradiation, the dose can be minimized taking into account the following:

- To spray the enviroment with fog water to reduce the concentration of smokes and radiactive aerosols in air.
- To change, as often as possible, the personnel involved in the different emergency tasks (firefighters, rescue team, etc.) to reduce the individuals exposure time.
- The maximum distance must be placed between the extinguishing team and the site as much as the access and layout of the emergency site allows it.
- To place an adecuate shield between the source and personnel in charge of extinction using the existing walls of the building or any other alternative means at hand.

Moreover it is convenient and sometimes mandatory, apart from the aforementioned procedures to planify a recuperative task of the radiactive source. This can be done e.g. preparing a cooled access road with the neccessary equipment (handlines, etc.) assigned to this exclusive operation and a convenient pike-pole to keep the smallest distance from the source to the fireman in charge of the tasks.

CONCLUSIONS

As the firemen activity does not exclude to face a radiological risk during any kind of operation it is convenient to keep always in mind and if possible written the recommendations already outlined to reduce the radiological risk as much as possible.


RADIACTIVE MATERIAL
FIREMAN
external contamination
physical change
burning
AIR
deposition
external irradiation
inhalation
spilling
COMBUSTIBLE MATERIAL
dragging
ESTINGUISHING SUBSTANCE
dragging
deposition
EXTINGUISHING EQUIPMENT
dragging
dragging
FLOOR
spilling
dragging
external irradiation
external irradiation

SOME EXPERIENCES ON EFFECT OF EXTERNAL DECONTAMINATION IN THE PREVENTION OF IRRADIATION

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Investigations are included experimental materials in the frame of decontamination of normal and damaged skin contaminated with radioiodine or radiocaesium and acquired experiences in human practice as external decontamination of persons contaminated with different radionuclides working at RA Vinča reactor and in laboratories which produce and use radionuclides, as well as after Chernobyl accident. In experimental conditions the efficiency of decontamination based on evaluation of the total body burden of radionuclides and residual radioactivity in the decontaminated region. In human praxis the decontamination treatment presents the most effective medico-prophylactic procedure in the prevention of local and total irradiation.

M E T H O D S

Methodology include the experiments and the evaluation of data taken from human praxis.

Experimental

The experiments were performed on the white rats. The skin made "fat-free" using soap before contamination. On the shaved skin of the extremity was made perforating wound by dental drill. The radioactivity of the applied amount of solution NaI^{131} and $^{137}CsCl$ was 1,85 MBq. Decontamination of the skin:five treatments were performed in duration of one minute each.Treatment of radiomixte based on rinsing action by use of vacuum system with choosen solution for decontamination. Means for decontamination: soap,saline,1% Cetavlon,2,5% Sterigal,0,5% iodine tincture, 5% Na-HMF,0,25% DBS-TR,PAM-03.

Human examination

A-Decontamination of persons at an operating nuclear plant

- Period of observation: over 25 years.
- Number of cases accidentally contamination employed on the Reactor RA in Vinča: 34.
- The efficiency of decontamination in dependence of physiological,radiological, physiological and chemical characteristics have been analysed, including: state skin and visible mucous membrane; anatomical locality (the head-frontal bump, supercilliary arch,aye,cheek,nose,mouth,chin,hair,hand,fingers,finger-nail etc.); duration of contact radionuclide with barriere (usually a few minutes, seldom 30 minutes); levels initial activity, chemical from of contaminant; state aggregation (dust,gas,liquid); time of duration of treatment at different cases (about 20 minutes);
- Radiocontaminants: ^{60}Co,^{3}H,^{24}Na,^{27}Al,^{109}Ag,^{131}I. Mixture of fision products. Unknowen composition (especially when contaminant is in from dust). Contamination with dust was in 35% cases.
- Localization: handfuls(16 cases-47%),face (9 cases-26%)etc.

B-Decontamination of persons contaminated in laboratories of Institute "Boris Kidrič" -Vinča

- Place of performance: Block for human decontamination-Vinča.
- Period of observation: over 20 years.
- Number of persons: 42
- Anatomical locality and number of cases:fingers of handful(20),handfuls (10), forehead (4) by three (hair forehead,nose,neck) by twice (face,lips) and by once (temple,ears,foot,fingers of foot).
- Radiocontaminants: ^{131}I (16 contamination-38,1%),^{198}Au (6 contamination-14,3%), ^{239}Pu (4 contamination-9,5%), by twice contamination with ^{32}P and ^{137}Cs(4,8%), by once contamination with ^{60}Co, ^{27}Al and ^{111}Ag. Registered also 4 contamina-

nation with mixture fission products and 5 unidentifiable contamination.

C-Decontamination of Yugoslav workers in Soviet Union

This materials concerning effect of decontamination of workers building plant Komgrap on building site Micord-Žlobin (Soviet Union) contained due to the Chernobyl are analysed.

Common procedures and means for decontamination from human praxis

Treatment: Decontamination of the skin contaminated with different radionuclides contained usually 4-5 procedures by use of cotton-wool and alternately by water rinsing or by cleaning with solution of means for decontamination. Means for decontamination: 5% detergent, toilet, medical and liquid soap, 3% citric acid, 2% tannic acid, saline, permanganate, 2% boric acid, emery; Detergent mixture, Hypex-detergent emulsion, 3% Versen,Lotion MO 8.385; Combination 5% Detergent and 2% Versen; Protective creams-Lanolin, Vaselin,Jecoderm,"Lek 48".

RESULTS AND DISCUSSION

Experimental data enable following confirmation:

- The efficiency of decontamination of the skin conditioned more by radioactivity decontaminated region but at perforating wound is greater influence of body burden of the penetrated radionuclides.
- Decontamination efficiency of the skin contaminated by radioiodine and radiocaesium was 90-99% when tretaed 30 minutes after contamination.
- Soap and saline used as a decontamination means shows the greater efficiency in the comparison with other used means.

Human data - it remains only for a summary of these findings to be made here:

- In majority of cases at personnel operating nuclear plant contaminants were ^{60}Co and ^{3}H but at contamination in laboratory conditions contaminant was ^{131}I.

-Application surface active substance (detergent,soap) as decontamination means show the greates efficiency (that reach to 100%).

- At 468 Yugoslav workers building plant Komgrap on construction site Micord-Žlobin that is 90% supervised persons established the contamination of clothing.
- Maximal number of contamination was registered on the hands (20%).
- The contamination of hair was evident at about 10%.
- Signs of contamination of the body established at 6 controled persons.
- Low level of contamination under 10 i/s registered at 38 cases, middle at 86 cases and higher level (over 100 i/s) evident at 16 cases (by use instrument KOMO-TN).

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INTERVENTION LEVELS FOR CASES FOLLOWING NUCLEAR ACCIDENTS

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To accomplishing with the safety criteria applied in Argentina, nuclear accidents with severe consequences in the public should have very low probability of occurrence. However, if one of these very improbable accident occurs, significant quantities of radioactive material could be released in the environment and, therefore, significant doses in the public would be incurred.

Radiological consequences from these accidents could be minimized if efficacious countermeasures are taken. Intervention levels are then an essential tool for decision making in each emergency.

The criteria used in Argentina in establishing intervention levels are in accordance with ICRP, publication #40 [1] and IAEA, Safety Series 72 [2]:a) Serious non-stochastic effects should be avoided; b) The risk from stochastic effects should be limited by introducing countermeasures which achieve a positive net benefit to the individual involved, and c) The overall incidence of stochastic effects should be reduced as low as reasonably achievable by reducing the collective dose commitment.

INTERVENTION LEVELS FOR EVACUATION

Based on the first criteria, the Argentine authority has established intervention levels for evacuation, as a courtermeasure to limit the dose due to the external irradiation from the material deposited on the ground. These intervention levels are 0.1 Sv integrated during the first six hours after the accident, for irrestricted evacuation, and 0.1 Sv integrated in the first twenty-four hours after the accident as a value below which no countermeasures are needed.[3]. In the middle, a case by case analysis is required, taking into acount the risk that could be avoided, and the risk introduced by the countermaesure itself.

Later on, when the ground deposit has sufficiently decayed, a decision is required about areas where reentry would be allowed for permanent occupancy. The optimized rate of dose for reentry of evacuated people should be such as to minimize the sum of the remanent detriment cost and the cost to maintain the countemeasure.

For purposes of emmergency planning, cost-benefit techniques for optimizing protection have been used for relocation decisions (4). As result of this analysis, the optimized time for reentry of evacuated people should be such that the dose rate at that time, $H(\theta)$, is equal to the ratio of the cost rate, C, for keeping evacuated an average person and the monetary value, α, assigned by the regulatory authority to the unit of collective dose.

$$H_\theta = \frac{C}{\alpha}$$

The Argentine authorities have selected a value of α equivalent to US$ 10,000 per man sievert for purposes of optimization on radiation protection. The cost (additional to the usual cost of living) of maintaining evatuated an average person is estimated in order of US$ 100 per month and per person. Therefore, the optimum dose rate for deciding reentry in Argentina will be in the order of 10^{-2} Sv/month.

INTERVENTION LEVELS FOR FOODS

Ground contamination also implies to take some decision concerning contaminated foods. A lower intervetion level was selected, below which no actions are necessary, using a cost-benefit analysis similar to that used in case of reentry. The cost of imposing a countermeasure such as the introduction of a ban on the consumption of a foodstuff, is taken as a first approximation, as the cost, C_f, of replacing that foodstuff at market-price. With this approximation, $C_f = K.V.$ where K is the foodstuff cost per unit mass or volume and V is the average consumption per person and per unit time. The optimum solution becomes:

$$H_{op} = \frac{K.V}{\alpha}$$

In practices the intervention levels of dose are more readily compared with the results environmental measures if derived intervention levels in terms of concentration are determined. The above method can be used to calculate the derived intervention levels, since: $DIL_I = \frac{H_{op}}{Fd.V}$; where Fd is the dose per unit intake.

$$DIL_I = \frac{K}{\alpha.Fd}$$

On the other hand, an upper derived intervention level that if overpast the consumption is authomaticaly prohibit, was derived from an individual dose limit of 50 $mSv.a^{-1}$.

$$DIL_U = \frac{50\ mSv.a^{-1}}{Fd.Vm}$$

Where Vm is the consumption rate representative of a hyphotetical critical group.

A list of DIL_I and DIL_U for Cs ^{137} for the principal foodstuffs consumed in Argentina are shown in Table I.

Intermediate situations are resolved by a case by case analysis, taking into consideration social and economic aspects.

Table I

Derived intervention levels for different tipes of foodstuff

(Bq/kg)

Foodstuff	Cs^{137}	
	DIL_U	DIL_I
Milk	15000	1800
Milk products	130000	16000
Meat	20000	8500
Green vegetables	50000	1300
Root vegetables	15000	950
Fruits	20000	1500
Cereals	15000	900

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Radioactivity and fire:a hard challenge for the Risk Management

UN UNACCEPTABLE RISK

by Antonio Bazzan

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The intervention on a fire situation in which radioactive sources or nu clear materials are involved entails a whole of risks for the operator, the sum of what must be considered unacceptable.

The Chernobyl event and the fatalities occurred to the Firemen present in the first phase of the accident are the more dramatic example of this almost axiomatic concept.

But also in situation less dramatic the contemprary presence of fire and radiactive materials must be carefully considered in all the phases of realiza tion of the facility (project, construction, exercise) in a way that the prevention measures put in action or so sure and effective that the probability of occurrence of the accident come be very very low.

The sum of two events must become pratically impossible. As a matter of fact parallel measures can be kept.

THE RADIOACTIVE SOURCES:

1. must be contained, when not in use, in closed metallic containers and stand in a "storage area" in which the "fire-charge" is very low;
2. the mouvements of radioactive sources inside the facility have to take place along well defined ways;
3. the experiences should be possibly managed in boxes with inert gas atmo sphere and the room should be slightly depressurized.

FROM THE FIRE PREVENTION POINT OF VIEW:

A. in the project the internal distribution of "non-nuclear" areas must be well defined in parallel whit the escape ways;
B. the "fire areas" must also be defined;
C. the Risk Analysis must be very deep;
D. the air-conditioning circuits must be able to avoid fire propagation (fire-stop shulters, fire-resistent filters etc.)

MEASURES OF PREVENTION

The principal concepts are well defined by the International Guidelines (I.G.) of the British Insurance Committee and collected between the principal insurance pools around the world.

They can be outlined as follows:

1. AUTHOMATIZED CONTROLS
2. VENTILATION DEVICES PROJECT
3. ALL THE ALARM SISTEMS MUST BE WELL VISIBLE AND WELL AUDIBLE IN LOCAL AND CENTRALIZED PANELS FOR IMMEDIATE IDENTIFICATION OF THE INCIDENT LOCATION
4. ALL THE INTERVENTION AND CONTAINMENT DEVICES (fire plugs, estinguishers, sprinklers, fire-doors, fire resistent comparts etc.) MUST BE REALIZED FOLLOWING THE BEST STANDARDS.
5. PARTICULAR ATTENTION MUST BE PUT IN INSTALLATION OF ELECTRIC CABLES.
6. THE "FIRE-AREAS" MUST BE SEPARATED FROM OTHER AREAS BY ANTI-FIRE BARRIERS 120 OR 180 R.E.I.

All those rules must be collected in the safety report and in the exercise book.

Finally, a detailed EMERGENCY PLAN must be prepared and frequent exercises must be performed.

THE INTERVENTION ON THE FIRE

Authomatic systems must be preferred, for example CO2 systems or the more modern Halon systems according with the facility characteristics.

All the systems will be doubled and have the possibility of authomatic and manual intervention.

The probability of human intervention has to became very low. Nevertheless if the intervention is needed it must be planned knowing exactly the containe ment conditions of the radioactive materials.

If necessary, a previous health-phisic technical control can be required.

Once verified those conditions, the intervention can take place with mask, aqualung and a suitable suit, following the rules and procedures of the EMER GENCY PLAN.

RISK MANAGER ROLE

But first of all, the point which is considered the basement by the I.G. is the precise definitions of the roles and responsibilities, inside the faci lity, in regard to fire-protection and other safety measures:

1. THE RESPONSIBILITY OF FIRE PREVENTION BELONG TO THE HIGHER DEGREE OF THE ORGANIZATION.
   SUCH A RESPONSIBILITY CANNOT BE DELEGATE.
2. THE PROJECT AND REALIZATION OF THE PREVENTION PROGRAM CAN BE DELEGATE TO THE FIRE PROTECTION MANAGER, WHOSE POWERS MUST BE AT THE SAME LEVEL OF THE RESPONSIBLES OF THE VARIOUS LINES OF ACTIVITY.

HOW TO COPE WITH EMERGENCY FROM TRANSBOUNDARY RELEASES OF RADIOACTIVITY

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Radiation protection in Yugoslavia is regulated by federal legislation. During the accident in Chernobyl, the current Code of Federal Regulations defined only the level of radioactivity above which the authorities had to be informed. The latest Code of Federal Regulations was issued only in August 1986, three months after the Chernobyl accident, lacking many instructions, which could have helped to avoid at least part of the financial losses. The greatest ommission was the lack of defined intervention levels. Despite all this, Yugoslavia was one of the few countries in Europe which had an operating radiological network, not established for the control of nuclear power stations. It was a left over from the time of nuclear experiments and was not disolved after the moratorium. In each republic and region one institution is licensed to monitor environmental radioactivity. In the Republic of Croatia this is the Institute for Medical Research and Occupational Health, the Laboratory for Environmental Radioactivity.

After the first information the about the Chernobyl accident, in the morning of April 29 the Republic Committee for Health Education and Social Welfare was informed that the frequency and type of radioactivity measurement in the republic network was increased. After the first increase of ^{131}I in air the Federal Committee for Health Education and Public Welfare was imediatelly informed of the situation. The republic authorities formed an ad hoc Interdepartmental Coordination Committee for Radiation Emergency including members representing health, agriculture, veterinary and meteorology services, civil defense, the army, tourism, public information and radiation protection experts. All the members of the interdepartmental committee made their decisions after discussions. On the basis of the informations recived through experts and some field information from civil defense centers, decisions were made concerning necessary protective measures. The measures consisted in warning the population to shield the drinking water wells from fresh fallout and cover all the accesses to cisterns, along the Adriatic sea. Those cisterns filled with rainwater are in many

places the only source of drinking water. Fortunatelly, the accident happened already during the dry season and only a few cisterns were contaminated by fallout. Al the others were proparly shielded and no contamination occured. The rest of proposed measures was the same as all over Europe. Two aditional handicaps aggravated the situation. First: the national holiday of the first of May which extended to four days, when everybody who could, left their domicile to go somewhere else. Second: the national elections were just over and the outgoing goverment and lower officials were not willing to be too deeply involved with an unprecedented situation. The newly elected members on all governmental levels, hoped, that this trouble would pass them by. This delayed in some instances the supply of reliable informations because the delegation of authority locally, was very often disturbed by the two above mentioned.

The permissible contamination levels were defined also by a Federal Interdepartmental Coordination Committee. This Committee coopted on May 7 radiation protection experts from all the institutes involved in radiation measurements and countermeasures from all the republics and autonomic regions.

On May 15 all interdepartmental committees were dissolved and a permanent commission for radiological protection was founded at the federal level, in all the republics and autonomic regions.
The following lessons have been learned from the experience:
In Croatia, due to the good organization of the monitoring network few well chosen samples defined the situation in the whole republic. A TLD network established several years ago, gave immediate results. Moss game and Iichen from locations under yearly control have been immediatelly collected and analysed. Naturally, that later on, thousands of samples have been analysed, many of them as a result of public fear.

Unexpected good cooperation in the ad hoc commission was probably a result of professional diversity of the members added by responsability. The cooperation with informationa media such as radio and TV-network was excellent but some difficulty presented several reporters from more sensation-oriented papers.

A serious setback came, where not expected, from the medial service, especially gynecologists suggesting unnecessary abortions. Even people from nuclear medicine were often unsure where to stand.

Future PRA studies must undergo serious changes. New social and health factors must be included, such as heavy losses in future life through abortions, disruption of normal life through elections and changes at the governmental level and holidays which reduce governmental staff at the decision-making level.

Other factors are: inadeqhuacy of the existing legislation necessity for unification of methodology to improve radiation protection.

THE COMPREHENSIVE EVALUATION OF THE RADIOLOGICAL HAZARD FROM SURFACE CONTAMINATION

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ABSTRACT

The aim of this paper is to point out the inadequacies of indirect assay for the assessment of the radiological hazard from surface contamination and draw attention to the new International Standard methods for comprehensive evaluation.

The measurement methods by which the immediate hazard as well as the medium and long term hazard potential can be fully assessed and the action necessary to control occupational exposure, are described.

It is argued that the level of particulate activity revealed by the assay of wipes, is only part of the immediate hazard if the surface contamination contains volatile fractions as these are not detected by this assay.

The hazard from volatile contamination is greater than that from an equivalent activity in particulate form because some forms can be absorbed through the skin of the whole body surface as well as inhaled.

Total surface activity must be determined to indicate the potential hazard from, contamination and surface, decomposition. Both the medium term between surveys and the long term for decomissioning are considered, taking into account ageing and working practices. The combination of direct measurements and assay by indirect means by which these hazards may be assessed is discussed, using tritium-surface-contamination as an example, as is the subsequent question of decontamination and its evaluation.

It is concluded that technically it is not difficult to assess all aspects of the hazard from low energy emitters as will be required by legislation incorporating the new International Standards.

COMPARATIVE EVALUATION OF WHOLE BODY COUNTERS FOR USE IN THE IMPLEMENTATION OF AN ICRP 30 INTERNAL DOSIMETRY PROGRAM

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Introduction

In order to implement an ICRP 30 based internal dosimetry program, a rapid and reliable bioassay evaluation method is needed. Direct bioassay (whole body counting) is the most desirable form of exposure assessment since it eliminates many underlying assumptions. Two commercially available whole body counting systems were evaluated for use in the assessment of intake (i.e., percentage of an annual limit of intake) and subsequent calculation of a fifty year dose commitment. A closed geometry three detector chair and a standup whole body counter were the two systems considered.

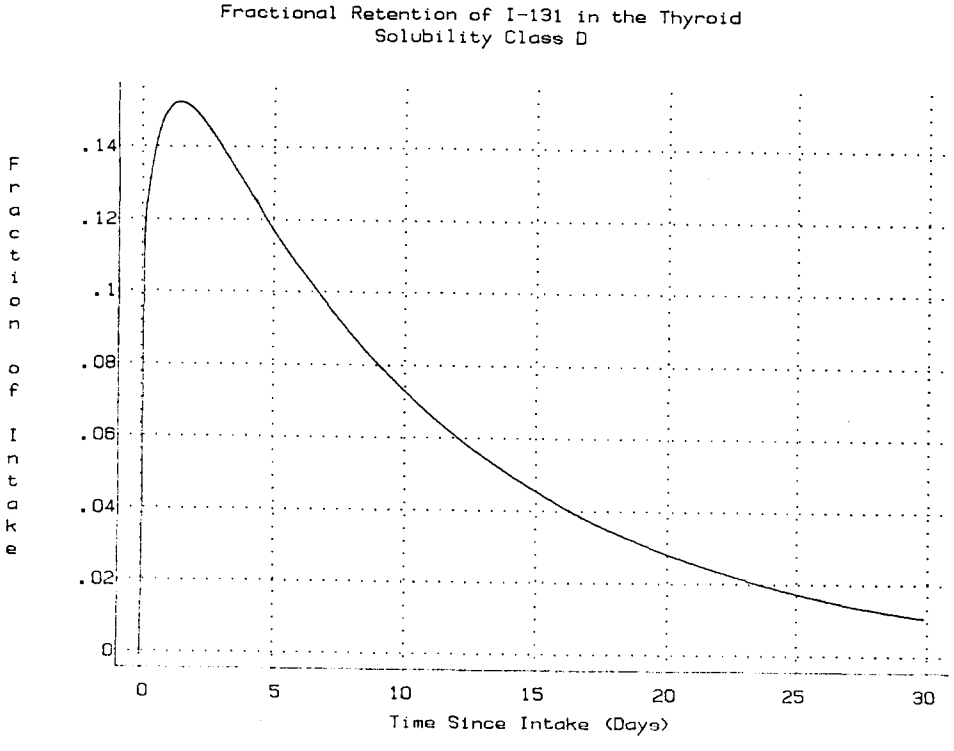
The three detector chair simultaneously measures the lung, the lower torso and the thyroid gland and is considered to be a diagnostic device to be used in the assessment of intake and internal dose assignment. The standup counter is composed of two 4"x4"x16" NaI(Tl) detectors mounted linearly in front of the subject. This detector arrangement provides for relatively high geometrical detection efficiency, but does not possess the organ specificity of the chair counter. Because of this, the standup counter is considered to be useful for screening purposes. Using the metabolic information contained in ICRP 30; however, both of these units can be used to calculate intakes.

Using newly developed computer software based on ICRP 30 methodology, organ specific retention patterns were generated for a variety of commonly encountered fission and activation products. The retention patterns were varied as a function of time. Calibration phantom data was then used to determine the response of each whole body counter to a particular nuclide distribution.

Calculation of Retention Fractions

The software package used to generate the retention curves calculates an intake by analytically solving a first order non-recycling compartmental model. The biological distribution of radioactivity in the body is described as a 20 compartment model with the initial compartment contents set to a predetermined value depending on the mode of intake. In addition, each compartment has an associated first-order transfer rate constant. Given the initial amounts of radioactive material in each compartment and the transfer rate constants between each compartment, the amount of material remaining at any time post-exposure can be calculated. A

representative retention curve generated by the program is given below in Figure 1.



The initial amounts of radioactive material deposited in each compartment is set by the program and depends on several factors. For an ingestion intake, the amount of material in the gastrointestinal tract is set to unity. For an inhalation intake, the lung compartment contents are set to the deposition fractions contained in the report of the ICRP Task Group on Lung Dynamics. The default deposition fractions for the initial lung contents are those for an assumed activity median aerodynamic diameter of 1.0 micron. For purposes of this investigation, all exposures were considered to be to inhalation of 1.0 micron particulate.

Once the initial contents of each compartment were set, the fractional amount remaining in each compartment was calculated. The time periods post-exposure selected for evaluation were zero, 1 hour, 10 hours, 1 day and 30 days. The compartments in the field of view of each detector measurement geometry were then summed. For the closed geometry chair the lung measurement was assumed to view all of the lung compartments with the exception of the naso-pharyngeal region while the lower torso measurement was assumed to view the entire gastro-intestinal tract including the stomach and the upper and lower intestines. The thyroid measurement was assumed

to view only the thyroid gland. For the whole body standup counter, all of the body compartments were assumed to be in the field of view. The minimum detectable intake observed by each detector geometry was then simply calculated using the following expression:

$$\text{Minimum Detectable Intake} = \frac{\text{Minimum Detectable Activity}}{F}$$

where: F = the total fraction of the initial intake viewed by the detector geometry.

Calculation of Detector Responses

The efficiency response of each measurement geometry from the chair and standup counter was obtained using bottle sources which were placed inside of a standard polyethylene based phantom which contains cavities for the simulation of lung, lower torso and thyroid organ distributions. The phantom was designed to conform to MIRD pamphlet No. 5 specifications. Efficiency response values were obtained for a series of isotopes covering an energy range from 165 keV to 1836 keV.

After the efficiency response for each measurement geometry was generated, background counts for each detector were obtained by measuring an uncontaminated subject whose height and weight approximated that of standard man. In this way, the minimum detectable activity for the closed geometry chair and the stand up counter could be calculated. The minimum detectable activity calculation was performed at the 95 per cent confidence interval. The minimum detectable intake was then calculated by dividing the minimum detectable activity by the fraction of the initial intake present at each measurement geometry.

Results

Tables 1 and 2 give the results of the minimum detectable intake calculations for both the chair lung measurement geometry and the standup counter geometry at various times post-exposure.

Table 1

Minimum Measurable Intakes at Various Times Post-Exposure Standup Counter

	Minimum Measurable Intake (Bq)					
Nucl.	Init	1 Hr.	10 Hr.	1 Day	10 Day	30 Day
Cr-51	1406	1406	1443	1591	6660	11803
Mn-54	152	152	152	163	481	629
Co-58	152	152	152	167	666	851
Fe-59	274	274	278	296	814	1184

Table 1 (continued)

Nucl.	Init	1 Hr.	10 Hr.	1 Day	10 Day	30 Day
Co-60	163	163	163	178	629	666
Zn-65	307	307	307	326	518	629
Ru-106	1406	1406	1443	1554	5476	6031
I-131	163	244	355	444	1408	9361
I-133	181	281	518	1036	----	----
Cs-134	144	144	144	148	170	200
Cs-137	148	148	152	155	178	200

Table 2

Minimum Measurable Intakes at Various Times Post-Exposure Closed Geometry Chair (Lung Measurement)

	Minimum Measurable Intake (Bq)					
Nucl.	Init	1 Hr.	10 Hr.	1 Day	10 Day	30 Day
Cr-51	2257	2331	2923	3552	6401	11063
Mn-54	207	237	274	322	518	703
Co-58	207	215	266	322	518	629
Fe-59	363	407	481	555	1036	1887
Co-60	203	211	263	315	444	481
Zn-65	407	407	518	629	925	999
Ru-106	2516	2627	3256	3885	5698	6068
I-131	259	351	555	1184	2.3E8	2.4E21
I-133	259	359	777	2368	----	----
Cs-134	252	337	518	1036	9.2E7	1.6E20
Cs-137	274	366	555	1110	9.9E7	1.7E20

The results demonstrate that the both systems are capable of measuring small intakes at short times post-exposure for both the insoluble (i.e., Co-60) and the soluble (i.e. Cs-137) nuclides examined. At longer times post-exposure; however, the lung measurement in the chair does not provide adequate sensitivity for soluble nuclides. This is due to the relatively rapid clearance of these nuclides out of the lung and subsequent deposition into other organs. For example, at 10 days post-exposure, intakes for both Cs-137 and Cs-134 can not be adequately evaluated in the chair while the standup counter is still quite sensitive even out to 30 days post-exposure.

Conversely, for some insoluble nuclides deposited in the lung such as Co-60 and Co-58, the chair measurement geometry provides somewhat better minimum measurable uptake capabilities at measurement times greater than 10 days post-exposure. In addition to this, the organ specific measurement capabilities of the closed geometry chair can be used to establish a subjects's own clearance patterns which can be used to calculate a more exact dose commitment. This precludes the use of using the default metabolic parameters contained in ICRP 30.

RAPID MONITORING OF PERSONNEL FOR INTERNAL CONTAMINATION

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Introduction

There is a need to monitor persons for ingested radioactive materials. This requirement is related to both radiation workers and to the public at large where affected by such incidents as Chernobyl and the more recent problem in Brazil with caesium.

For many years this need has been met by the use of whole body monitors. However these are extremely expensive to install and operate and as a consequence are not available in all centres that need them. This means that persons may have the worrying experience of travelling a long distance to be exposed to a claustrophobic experience where in most cases there is no contamination problem at all.

Recently however a number of simple monitors have been developed which are cheap enough to be installed in most establishments and are able to confirm the necessity or not of the more thorough investigative properties of the whole body monitor.

Practical Limitations of Measuring Internal Contamination

The problems involved in designing equipment to meet this need are great. One being the typical person to be monitored, who is not typical.

Heights of European adults vary from 143.5 to 194cm (excepting the 1 in 1000 extreme). If equipment was designed to check the chest of a tall standing man it would be over the head of a small woman. Height adjustments to equipment can be time consuming and can be embarassing to some users. However if the user is sitting down the variation in shoulder height is only 48 to 64.4cm and the variation in the height of the mid point of the lung even less. So one should seat the user.

The user can also be slim or rotund; the thickness of the trunk varying from 14 to 43cm. Figure 1 shows the variation in sensitivity with position of a point source of caesium through a body using a large area detector (100mm x 100mm). The variation will be even greater with smaller area detectors. So even in the case of a thin person the variation of performance through the body is 14.1 even assuming critical organs start 2cm from the body extremity. If however two detectors are used front and back the performance approximates to Figure 2.

How can even this variation in detection efficiency be reduced? By the use of large area detectors. As well as the obvious increase in counting efficiency and uniformity of response in planes parallel to the plane of the detector, uniformity of response is also increased at right angles to this plane as illustrated in Figure 3.

We are therefore drawn to the use of two large area detectors with the user seated to provide the most uniform response to ingested material. There are however structural as well as background limitations to be considered in deciding on the area of the detector i.e. E^2/B should be maximised as much as possible.

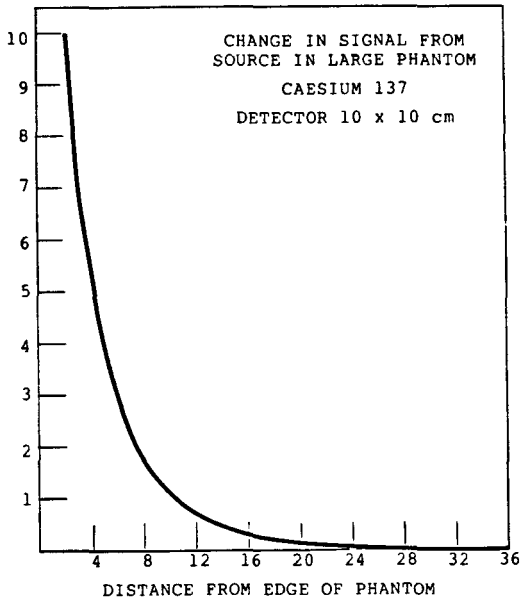

FIGURE 1

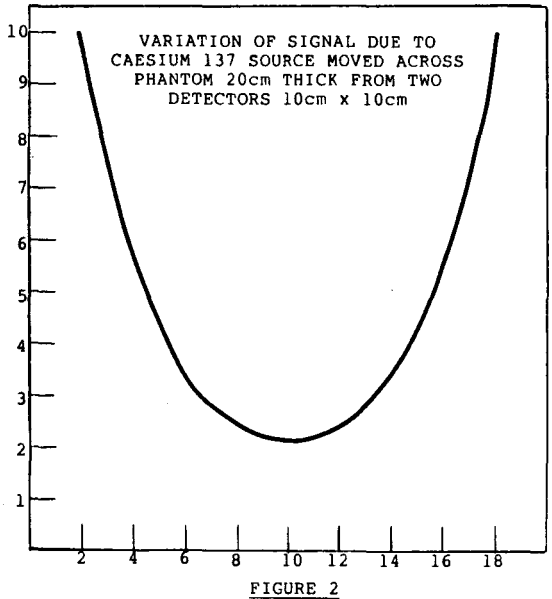

FIGURE 2

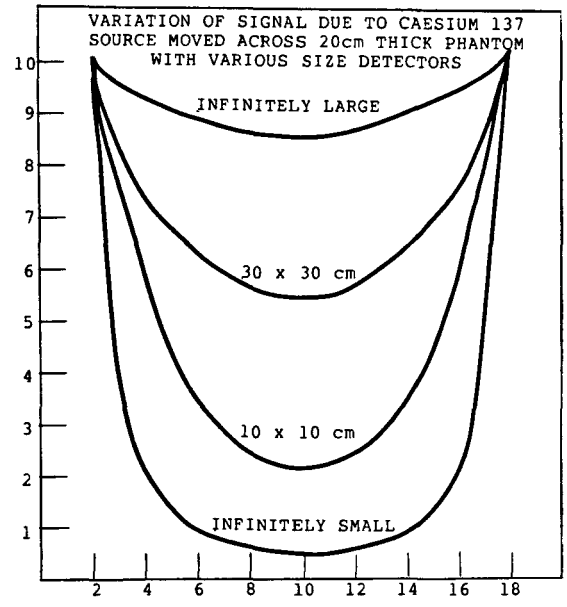

FIGURE 3


FIGURE 4

Design

For high sensitivity to gamma radiation the scintillation counter is ideal and to obtain large areas the only economic solution is the use of plastic or liquid. Very large inorganic crystal scintillators would be prohibitively expensive. Liquids have containment problems.

However plastic detectors have a very poor energy resolution, but, the primary intention is to detect contamination not to identify it. The use of plastic phosphor also has the advantages of mechanical robustness and stability with temperature change, making instrumentation easy to maintain and allowing for designs which can can be used by unskilled operators.

There is however a major problem in using very large gamma sensitive detectors, and that is the high sensitivity to background. The effect can be minimised by monitoring the background when not in use and subtracting this rate from the measured value. The user can however have considerable effect on the level of background both absorbing it and scattering it. To assess this effect a fairly standard sized person, known to have no internal contamination, can be used during setting up to determine the effect of the user on the radiation background being measured. This information can then be stored and subsequently used to adjust the background count subtracted from the measured count.

The front detector can be made moveable against the body so that information can be provided as to whether the user is thin (ectomorphic) or rotund (endomorphic) so that further adjustments to the background signal can be made to account for body shapes. The thickness of the body can also be taken into account in calculating the contamination present. The signal to unit contamination relationship is highest for thin persons and this relationship is calculated on the assumption that the contamination is in the centre of the body. If the contamination is elsewhere the indication of contamination is likely to be slightly higher than the true contamination level. This depends on the variation of response of the equipment to contamination at that place from that at the centre.

Conclusion

We took account of all the considerations given above in the design of the equipment shown in Figure 4. This equipment is extremely simple to operate and can if necessary be used by personnel without supervision. The only precautions necessary are to limit persons standing or moving near the equipment when operated in an elevated or very directional background environment since they will disturb that environment considerably.

The minimum detectable activities that can be achieved when used for 10 seconds in a 0.1mSv/h (10μR/h) background are as follows allowing 3.1 S.D on background.

^{60}Co	180Bq	(5nCi)
^{54}Mn	370Bq	(10nCi)
^{137}Cs	440Bq	(12nCi)
^{133}Ba	440Bq	(12nCi)
^{57}Co	930Bq	(25nCi)

The equipment has the advantage of high sensitivity to contamination trapped in any organ so no ingestion hazard should be missed even if the location of the entrapment is unforeseen.

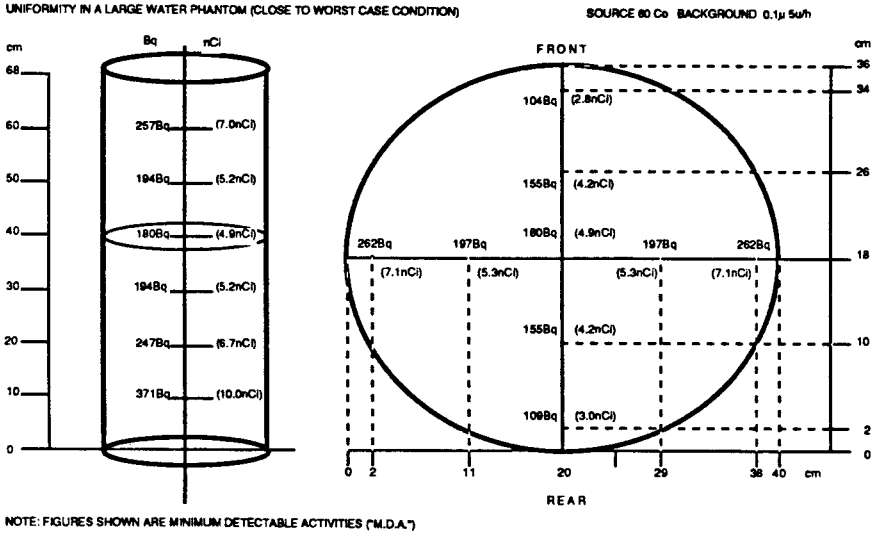

FIGURE 5

A measure of the effectiveness of the use of the very large scintillation counters is illustrated in Figure 5; the water phantom simulating almost the most rotund. The uniformity with the 340mm by 300mm and 300mm by 600mm detectors is in fact very slightly better than would be indicated by Figure 3, this is due to the fact that the advantageous effects of scattering had not been taken into account in the preliminary calculations. One case where the performance of the final equipment proved better than that indicated by the initial theoretical predictions. The variation with Caesium 137 are very very similar.

The technique used in this personnel monitor can also be used elsewhere where very high levels of sensitivity are required to detect small quantities of materials which may be anywhere in fairly large objects. A prime example is the monitoring to de minimus levels of sacks of waste prior to shipment from nuclear sites to standard domestic or other garbage dumps.

A VERTICAL SCANNING GERMANIUM WHOLE BODY COUNTER: THE ACCUSCAN-II

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INTRODUCTION

The ACCUSCAN-II was designed to fulfill the needs of two types of users. Nearly all nuclear power plants require two WBC systems for redundancy. One of these counters should be like the Canberra FASTSCAN (contains two large 4" × 4" × 16" NaI detectors) for rapid processing of the vast majority of the workers who have little internal deposition, or have uptakes that are simple to interpret. The ACCUSCAN-II is the ideal second counter and allows the user to more easily interpret these few, but very important situations where uptakes do exist.

For the second type of user, facilities that only need one WBC system, the ACCUSCAN-II is a perfect choice. It can be operated in a high sensitivity mode, with the subject's lung (or thyroid, GI, etc.) in contact with a non-moving detector for maximum sensitivity. Subjects can be quickly screened this way. When activity is found worth investigating further, the subject can be recounted in the total body scanning mode for better accuracy and to determine the precise source location.

FIGURE 1 - MODEL 2280C2 ACCUSCAN-II

The ACCUSCAN-II uses one or two germanium detectors (Figure 1). The inherently superior energy resolution makes it easy to separate isotopes that are difficult to resolve with NaI detectors, and to quickly identify new or unusual nuclides. The detectors generally scan during the count. As the detector moves, a positional spectrum can be simultaneously acquired. This positional spectrum will aid the user to determine the organ of location, or reduce the interference from external contamination, as an aid to more accurate dose determinations. A third advantage of germanium detectors is their wide dynamic range. This is especially important in emergency situations where large internal depositions or contamination (e.g. one MPBB of ^{137}Cs) will render high sensitivity NaI counters useless.

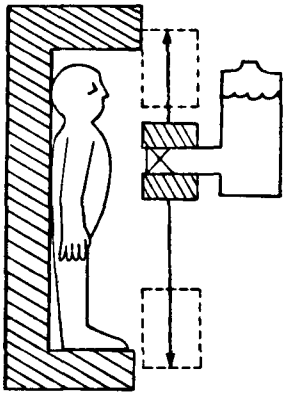
ACCUSCAN-II COMPONENTS AND SYSTEMS

The shield is a full 4" of low background steel around the subject and 2" lead around the detectors. It is a complete 4 pi shadow shield and can be used in elevated background areas. There is an optional cryostat design which includes external lead shielding between the detector and the dewar. The shield weighs about 9000 lbs (4000 kg), but is in small enough components to allow manual assembly, if necessary. The subject can be counted standing (for full body scans), or seated (for longer count times of a limited portion of the body). When standing, counts can be done from the front and then from the back to achieve the maximum accuracy and independence of source location. The subject can stand or sit against the back wall for best accuracy, or can lean against the detector in a high sensitivity mode. Figure 2 shows these various ways the ACCUSCAN-II can be used.

The ACCUSCAN-II is available in three basic versions. Model 2280A is the most basic and least expensive. It has the full complete shielding, an adjustable speed motor drive for the detector, a 25% relative efficiency detector, and a Series 35 PLUS 2048 channel MCA. The internal firmware of the MCA is used to find peaks, determine their energy, determine their identity, and to determine net peak area. This is an ideal system for those facilities which have few people to count, or limited funds. As needs change, the system can be upgraded.

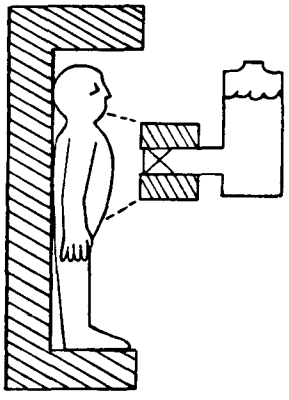
Model 2280B includes a DEC computer and the ABACOS-II software. It is delivered as a turnkey system, with phantoms, sources, factory calibration, all parameters in the software loaded, and is ready to count. The standard computer is the MicroVAX-II, however the MicroPDP 11/53 is a less expensive alternative.

The 2280C1 is a fully automatic system. The scan drive is by means of a computer driven programmable stepping motor. With the added multichannel scaling (MCS) capability, a spectrum containing total counts vs. position is stored in a portion of the MCA memory and displayed. This occurs automatically and simultaneously with the energy spectrum acquisition and display. The information derived from the positional display is useful to properly determine the organ of deposition, which is necessary to correctly calculate dose. The 2280C2 adds a second Germanium detector. This increases efficiency, reduces the LLD or the count-time, and increases reliability through redundant components.



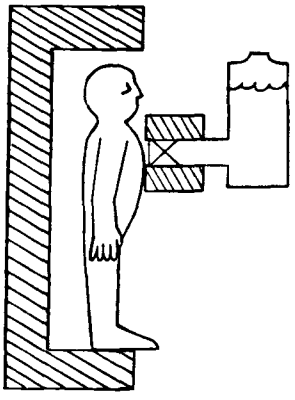
TOTAL BODY SCAN

- Full length scan
- Total body count
- Front/back count
- Diagnostic position for maximum accuracy
- Positional information for proper dose determinations



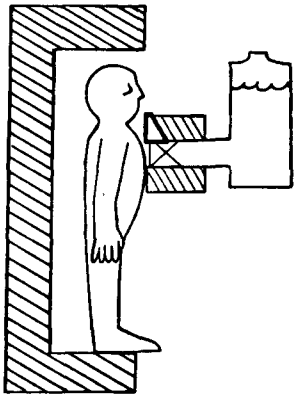
FIXED POSITION-DIAGNOSTIC

- Fixed position count or limited length scan
- Diagnostic mode for accuracy
- Can view thyroid, lung and GI tract
- 3X more sensitive than total body scan
- Can be repeated from back for maximum accuracy



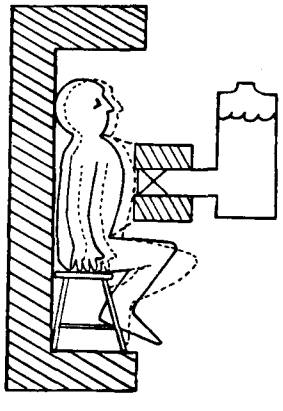
FIXED POSITION-SCREENING

- Fixed position count
- Screening mode for sensitivity
- Can view thyroid, lung or GI tract
- 2X more sensitive than diagnostic mode
- Can be repeated from back for maximum accuracy



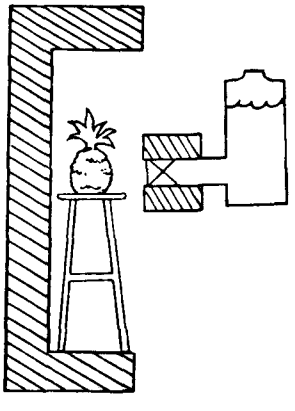
LUNG-THYROID SCREENING

- Lung/thyroid combination screening count
- Removable thyroid shield plug



SEATED COUNTING MODE

- Chair diagnostic mode (against wall)
- Chair screening mode (against detector)
- For longer count times



SAMPLE COUNTING

- Same system can be used to count a wide variety of sample sizes
- Low background shield construction
- Shadow shield construction—no door to open and close
- Virtually same background as full shield for energies > 250 keV

FIGURE 2 - VARIOUS COUNTING GEOMETRIES OF THE ACCUSCAN-II

For the computer based systems, operation with ABACOS-II is extremely simple for the pre-configured routine counting conditions. It is also very flexible when using the special operations mode. Extensive error checking is used to reduce operator input mistakes. The multi-user, multi-tasking software, and the use of a separate stand-alone MCA and computer, allows counting of the next subject while the previous spectrum is undergoing analysis. Furthermore, the separate components make troubleshooting easier, and allows operation using the MCA if the computer is inoperable.

ABACOS-II SOFTWARE

The key element to transform a group of detectors inside a shield to a complete, successfully operating WBC system capable of performing in a production environment is the software program. The software must efficiently convert the spectra into a well documented, reliable, legally defensible record. ABACOS-II has been designed specifically for the unique and demanding purpose of spectral analysis for whole body counting. It is menu driven, and therefore, very easy to learn and to operate. All operator responses are checked for validity, to minimize incorrect data entries. After starting the count, and answering the demographic entries, the entire sequence of STOP/TRANSFER/ANALYZE/REPORT is automatically performed by ABACOS-II. Because of the program structure, a new count can start after the transfer task is complete. The analysis and reporting tasks operate in parallel with the next acquisition.

The system can be pre-configured by the whole body counting manager in a wide variety of ways. Even if the system is pre-configured for a certain standard set of conditions (library, efficiency, energy range of analysis, etc.), these conditions can be easily changed on a case-by-case basis for further reanalysis of the same spectra, without recounting.

A drawback of most programs for analysis of low intensity peaks has been the peak search routine. It is a well recognized problem that searching for small peaks in low-count spectra is a difficult task. Sometimes the peak search routine is too sensitive, and finds many false positive peaks. Reducing the sensitivity to false positives also makes the false negatives increase, so that peaks near the calculated MDA will not reliably be found. Canberra has solved this probelm by implementing a unique library-driven peak search in ABACOS-II. This technique truly allows near-MDA peaks to be reliably found, even if they are only a few counts in area or on a zero-count background. Furthermore, this technique still allows corrections for interferences from adjacent peaks or from the underlying Compton continuum.

A critical element in an *in vivo* measurement system is Quality Assurance. ABACOS-II has an internal QA program that automatically tracks three types of QA counts (check sources, blanks, and duplicates). Each of these count types has user-adjustable predetermined limits of acceptance. If any of the results is outside its limits, then the operator is notified immediately. Twenty-five different parameters are tracked, and can have high and low warning levels. The results of the QA program can be printed out in a tabular form, and/or plotted on the printer for quick visual review.

To prevent unauthorized changes in parameters, data, or reports, there are six different user definable passwords within ABACOS-II. Each password protects a separate function, and can therefore be distributed to different user levels. ABACOS-II can print out the MDAs (user selectable) for peaks and/or nuclides searched for, but not found. The definition of MDA is also user definable (e.g. 1 sigma, 3 sigma, 4.66 sigma) and allows the use of the 2.71 constant to account for low-count statistics. ABACOS-II also provides spectral and efficiency plots from the computer. The CIMPA plotting package is used by ABACOS-II, and is optionally available, along with the necessary hardware.

PERFORMANCE RESULTS

Figure 3 demonstrates the typical LLDs of the ACCUSCAN-II. The LLD is calculated using 5% probability of type I and type II errors, i.e. the activity equal to $2.71 + 4.66\ (BKG)^{1/2}$. A 5 minute count time was used. Shorter count times can be used, but the LLD is increased accordingly. The efficiency was determined using the Livermore/Humanoid torso phantom. The background is from an average subject containing only ^{40}K. The gamma abundance was assumed to be 100% for the chart. Therefore, divide the LLD shown by the actual gamma abundance of the nuclide of interest. One or two germanium detectors are used, as noted, each 25% relative efficiency.

Curve A represents a total body scan, with the source in either the lung or the total body, and the subject in the diagnostic position. Curve B represents a lung scan, with the source in the lung, and the subject in the diagnostic position. Curve C is a stationary detector, with the source in the lung, and the subject in the screening position.

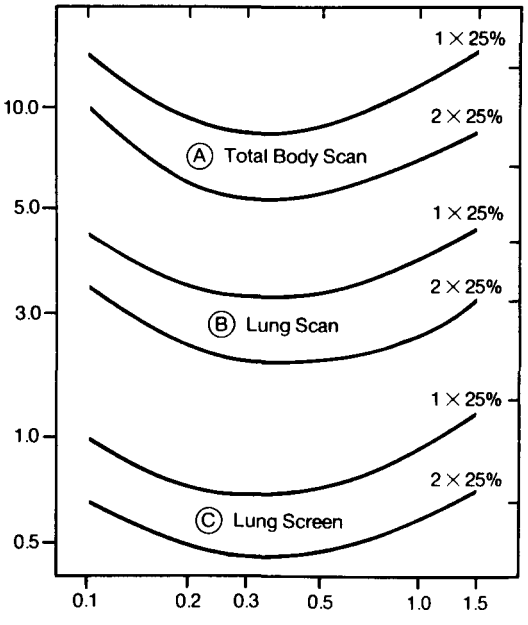


FIGURE 3 - TYPICAL ACCUSCAN-II LOWER LIMIT OF DETECTION

URANIUM/PLUTONIUM LUNG COUNTER USING MULTIPLE GERMANIUM DETECTORS

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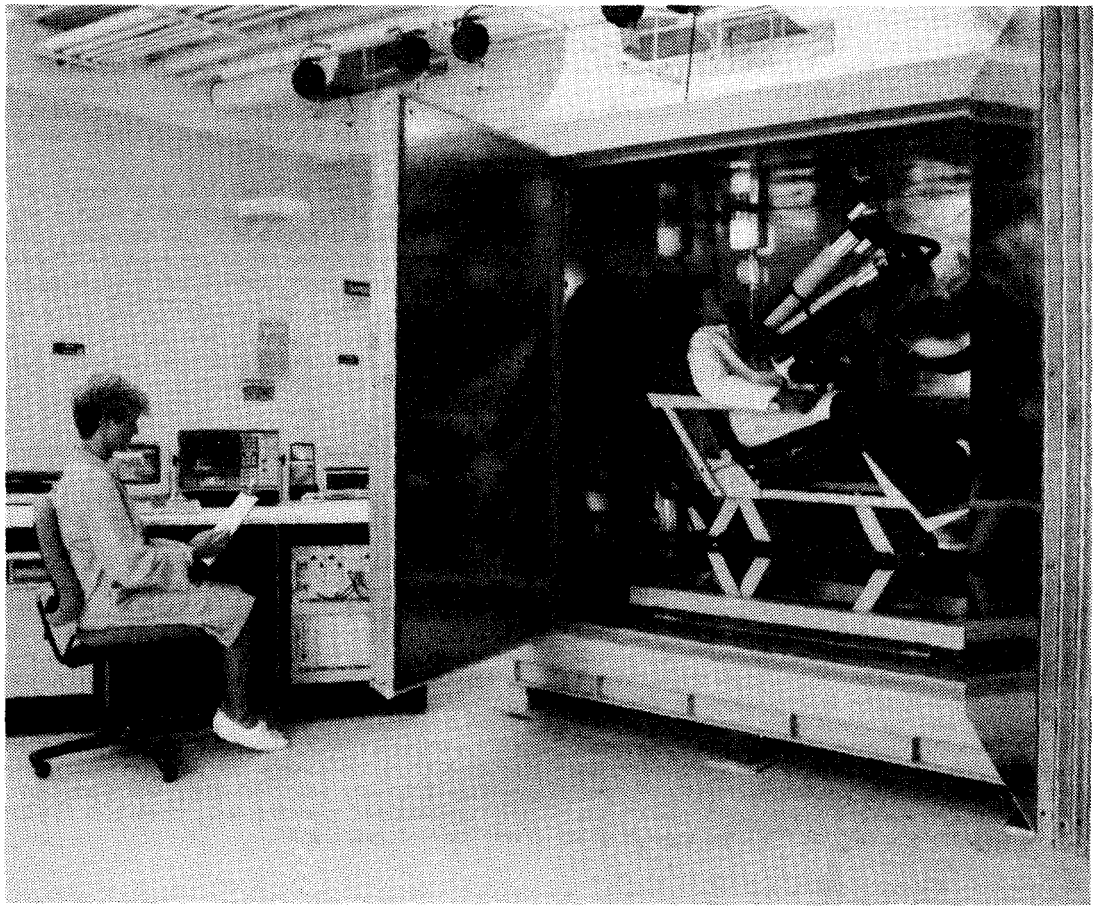
INTRODUCTION

The accurate measurement of the lung deposition of Actinides such as Uranium, Plutonium, and Americium is a critical need of the facilities dealing with these materials. Traditionally, NaI/CsI phoswich detectors have been used; however, in recent years extensive progress has been made using high purity germanium detectors. Their main advantages are a more positive identification of radionuclides, somewhat greater sensitivity, and less interference from higher energy gamma rays. While the phoswich detectors have higher efficiency, the much higher resolution of the germanium detector provides better photopeak-to-background ratios and a better background correction capability. The detection of U/Pu/Am is particularly a problem where there are concurrent depositions of higher energy nuclides (e.g. ^{137}Cs from Chernobyl, or ^{60}Co from reactor maintenance). The environmental ^{137}Cs from Chernobyl has essentially made phoswich detectors useless for plutonium on workers in Europe.

The move from the concept of using germanium detectors in a research facility to using them on a working system in a production environment, however, required a unique mechanical design for the detector and cryostat, special software, as well as a sophisticated approach to the shielding of the subject. This system incorporates the experience gained in designing, manufacturing and operating nearly 100 WBC systems, including six previous U/Pu Lung Burden systems. The result is a complete, turnkey system specifically tailored to the measurement and analysis of U/Pu lung burdens on a routine basis. The system recommended consists of a steel shield, 4 to 8 Intrinsic Germanium detectors and associated electronics, a multichannel pulse height analyzer, a DEC computer system for data analysis, and Canberra's proven ABACOS-II software.



SHIELD

The shield is 6" thick, and is constructed of certified low background steel, from a source that insures the absence of ^{60}Co. Lead is an undesirable material for low background shields. All lead (even so-called old, virgin, or low background lead) has radium and thorium, which have peaks that interfere with the very low level of U/Pu measurements to be made.

The conventional Pb/Cd/Cu liner is not beneficial in U/Pu lung systems where the detectors are in contact with the subject (a low Z material). This extra expense can be avoided. It is beneficial, however, to add a thin graded-Z liner around the portion of the detectors not in contact with the subject (sides and back). This lowers the background about 15%.

The typical shield has inside dimensions of 6' wide, 6' tall, × 4' deep; other sizes are available. If desired, the shield can be built from stacking interlocking modules, allowing assembly inside an existing structure.

The interior of the shield is covered with a low background wall lining and floor covering. The interior is lighted, ventilated, has an intercom, a music system and has a closed circuit TV camera to view the subject.

DETECTORS

The detectors are Canberra's LEGe type, a design now copied by other manufacturers. This is a low profile coaxial detector, with better resolution and background performance characteristics than planar detectors of comparable size. The recommended size is 20 cm^2 area, and 20 mm thick. The detector thickness is optimized for the minimum MDA of nuclides from 60-200 keV. Both 4, 6, and 8 detector configurations are available.

The detectors are mounted in Canberra's newly designed multi-attitude cryostat, the ACT-I. The ACT-I is a unique low background configuration designed for maximum density detector arrays. The small (2.75") diameter endcap and the 3.5" diameter cryostat are critical in obtaining the maximum efficiency from an array of detectors on the chest of a subject. All materials are selected or tested for low background, resulting in a background spectrum virtually free of peaks from contaminants in the detector package. The rear mounted preamplifier keeps cables away from the subject's face, and the printed circuit board away from the detector. The ACT-I is only 28" long, allowing smaller shield size than longer designs. The detector can rotate to 180 degrees without LN spillage on the subject, an important safety consideration. The holding time is 24 hours.

Important to the maintenance of the low background performance of the LEGe detector is separating and shielding radio-sensitive items (detector) from other potentially radioactive items (dewar, subject). Internal tungsten or lead shields are not desirable because of additional cooldown times, and because of the inherent radioactivity of tungsten. For low level counting they typically increase the background. Canberra's unique Remote Detector Chamber (RDC) allows a removable external detector shield to be placed surrounding the detector, thus shielding the detector from the cryostat, the preamplifier, and the scattered radiation background environment. With the RDC, the expense of a Pb/Cd/Cu graded lining on the inside of the shield can be avoided. The Canberra RDC allows shielding to be placed around the sides and back of the detector element, thereby using a much smaller amount of material.

SUBJECT-DETECTOR POSITIONING MECHANISM AND LIQUID NITROGEN FILL SYSTEM

The subject is seated in a semi-reclining position in a chair geometry. This geometry is the optimum for minimum chest wall thickness (especially females and large males) and also is more comfortable than fully reclined. The detectors slide out of the way, and the chair slides out of the shield for easy subject loading. The chair mass is minimized to avoid background contribution. All components used for the subject-detector positioning mechanism are tested and chosen for minimal radioactivity. When a subject containing only ^{40}K is counted for 60 minutes, no peaks interfering with Pu/U/Am will be statistically present (greater than the LLD).

A simple, (and therefore quite reliable) LN_2 filling system is supplied. This system is automatically enabled by means of a programmable timed controller system. There is also a manual fill capability. Typically, the programmer would be set to fill at times when the system is not being used (e.g. 6 AM) Each detector has a detector temperature monitor with visual indicator and audible alarm, and a bias disable circuit to remove HV if the detectors warm up.

ELECTRONICS

The electronics consist of independent bias supplies and amplifiers for each detector. The amplifier contains an ultra-fine gain control to allow precise matching of the detector spectra. Each detector's spectrum is stored separately in the MCA.

To provide the ability to function if the computer is inoperative, an independent standalone MCA is provided (the Canberra Series 35 PLUS, with 8192 channels of memory). Other MCAs are available offering more channels of memory (to 64K) and/or color display.

The MCA is controlled by either a DEC MicroPDP-11 RSX based computer system (least expensive) or, the recommended MicroVAX-II VMS based computer system (faster, more future expansion).

SOFTWARE

The key element to transform a group of detectors inside a shield to a complete successfully operating WBC system capable of performing in a production environment is the software program. The software must efficiently convert the spectra into a well documented, reliable, legally defensible record. ABACOS-II has been designed specifically for the unique and demanding purpose of spectral analysis for whole body counting. It is menu driven and therefore very easy to learn and to operate. All operator responses are checked for validity, to minimize incorrect data entries. After starting the count, and answering the demographic entries, the entire sequence of STOP/TRANSFER/ANALYZE/REPORT is automatically performed by ABACOS-II. Because of the program structure, a new count can start after the transfer task is complete. The analysis and reporting tasks operate in parallel with the next acquisition.

The system can be pre-configured by the whole body counting manager in a wide variety of ways. All spectra can be analyzed and reported separately, or as a single summed analysis of all detectors, or as two analyses of two detector groups each (left and right lungs). Even if the system is pre-configured for a certain standard set of conditions (library, efficiency, energy range of analysis, detectors to be summed), these conditions can be easily changed on a case-by-case basis for further reanalysis of the same spectra, without recounting.

A drawback of most programs for analysis of low intensity peaks has been the peak search routine. It is a well recognized problem that searching for small peaks in low-count spectra is a difficult task. Sometimes the peak search routine is too sensitive, and finds many false positive peaks. Reducing the sensitivity to false positives also makes the false negatives increase, so that peaks near the calculated MDA will not reliably be found. Canberra has solved this problem by implementing a unique library-driven peak search in ABACOS-II. This technique truly allows near-MDA peaks to be reliably found, even if they are only a few counts in area or on a zero-count background. Furthermore, this technique still allows corrections for interferences from adjacent peaks or from the underlying Compton continuum.

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PERFORMANCE

The LLD for a 4 detector system has been calculated as the activity equivalent to $2.71 + 4.66\,(BKG)^{1/2}$. Background is from a 30 minute count of standard size person (23 mm chest wall thickness) containing only ^{40}K. When comparing different systems, care should be exercised to assure a consistent definition of the LLD. More detectors, longer count times, less ^{40}K, thinner chest wall, ignoring the 2.71 factor, and different confidence level will all generate lower LLDs.

LLD	^{239}Pu	^{241}Am	^{234}Th	^{235}U
	75 nCi	0.10 nCi	1.0 nCi	0.10 nCi

A SAFE INFRA-RED Nd:YAG LASER FIBEROPTIC GUIDE SYSTEM

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Northrop Corporation
2301 West 120th Street, Hawthorne, CA. 90250, U.S.A.

ABSTRACT

Laser marking or engraving of serial numbers and identifications on strategic metals such as Beryllium is attractive as a non-contact and superior method. The laser system of choice is the Neodymium-YAG solid rod laser which produces a sharp infra-red beam at 1.064 μm wave-length. The design calls for a beam of 250 μm numerical diameter and power output of 75 - 100 Watts cw or Q-Switched. For the purpose of laser marking, it is not advisible to transport toxic metals such as Beryllium outside the 'controlled' environment. A 'remote' marking system was developed utilizing a fiberoptic delivery system. The laser beam is readily transmittable through a plastic clad pure SILICA CORE fiber 300 μm in diameter. Around the hard acrylic clad there is a TEFZEL buffer enclosed by Kevlar webbing. The outermost jacket is made of polyurethane. The overall diameter is 2-4 mm. The special cable is further sheathed in EMT light gauge flexible galvanized steel conduit ½" dia. passing above the false ceiling in an industrial building. The distance between the source laser and the workstation inside the Beryllium room is about 300 feet a distance across which there is a laser beam power loss of about 1 db or about 20% loss. The loss at each end of the fiber is about 5%. Thus a loss of 30% occurs making the 'delivered' laser beam little over 50 Watts. Galvanometer X & Y Mirrors move the beam under computer

A HIGH RELIABILITY PERSONAL ALARM DOSEMETER WITH A SEMICONDUCTOR DETECTOR

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Eisuke Okamoto and Yoshiteru Yoshida
Tokyo Factory, Fuji Electric Co., Ltd.
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SUMMARY

A personal radiation exposure dosemeter in nuclear power plants and other nuclear-related facilities was developed. This is equipped with a semiconductor detector and is provided with a wider measuring range and longer service life compared with a conventional one using a GM counter. This is also provided with data transmitting functions using an optical communication system.

FUNDAMENTAL COMPOSITION

Figure 1 shows a block diagram of the present dosemeter. The detecting part employs a-Si:H/C-Si heterojunction diode made by a plasma CVD (Chemical Vapor Deposition) process. An amplifying circuit, a discriminating circuit and a bias circuit (+15V) compose a hybrid integrated circuit with the diode. The whole circuit is controlled by a 4-bit one-chip micro-processor. Counting pulses from detecting part are arithmetically processed and displayed as an integrated dose by LED.

Alarm level setting can be made with a manual alarm setter. An electronic buzzer is put into operation when the integrated dose has reached the set alarm value. A data communication circuit connect the information with an external computer system through an optical communication system.

A power source using Ni-Cd chargeable batteries is normally monitored by a voltage drop detecting circuit.

CHARACTERISTICS

The present dosemeter is characterized as follows:

(1) Less than 2.8×10^{-2} %/10^3 hr in failure rate has been achieved.
(2) Two types of energy response are available. (See Figure 2)
(3) Continuous operation is ensured over 12 hours in the temperature range of 0 - 50°C
(4) All the information on the dosemeter conditions, including integrated dose, built-in battery voltage, operating status and set values, can be outputted to an external system.
(5) Accurate and stable measurement is ensured both in N-16 high-energy gamma-ray (6 MeV) atmosphere and in high dose rate conditions (up to 50 R/hr). (See figure 2, 3)
(6) The present detector is provided with self-diagnosing functions such as battery voltage drop detection as well as detector self-check in the natural background.

(7) The present detector, compact in thickness (see figure 4) and light in weight (approximately 200g), can be carried in a pocket of working clothes.

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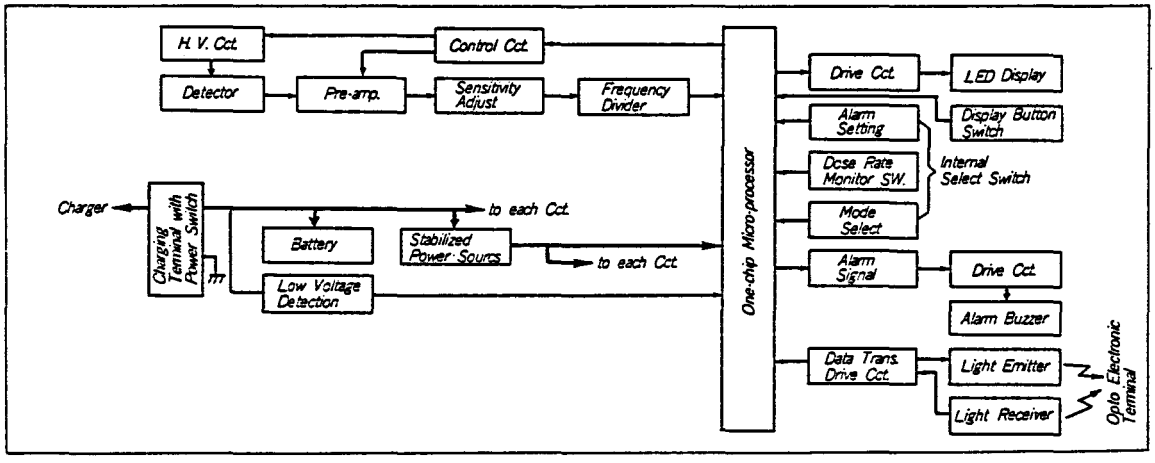

Fig.1. Block diagram

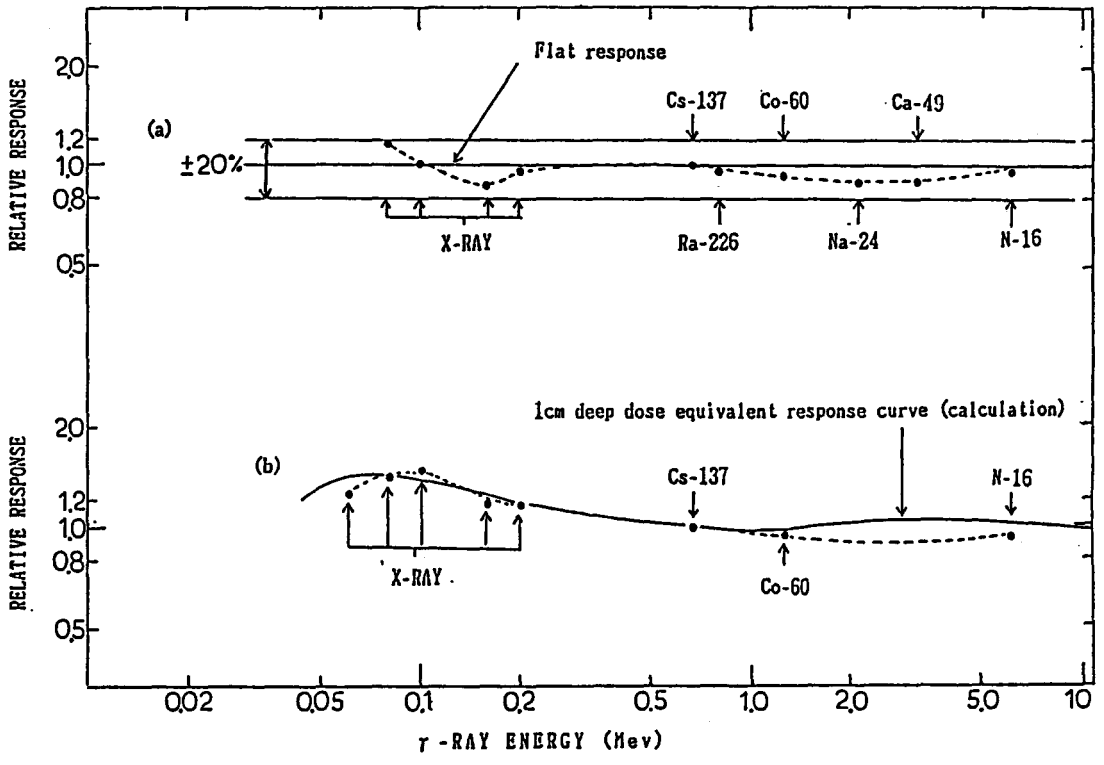

Fig-2 Energy Dependence of the present detectors normalized at Cs-137

(a) Flat response type detectors (b) 1cm deep dose equivalent response type detectors

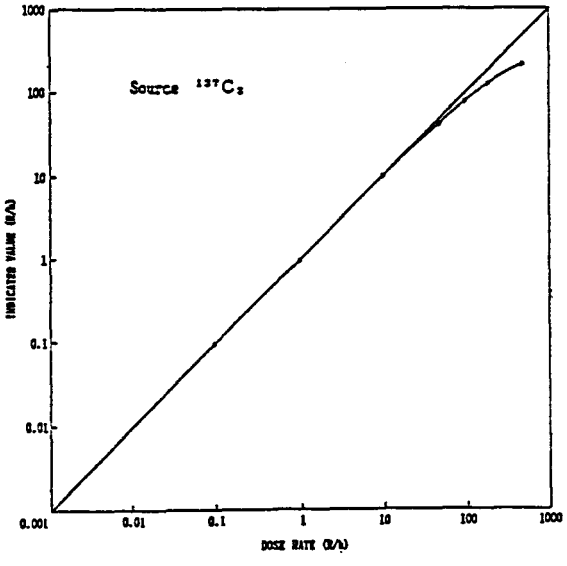

Fig.3. Dose rate dependency

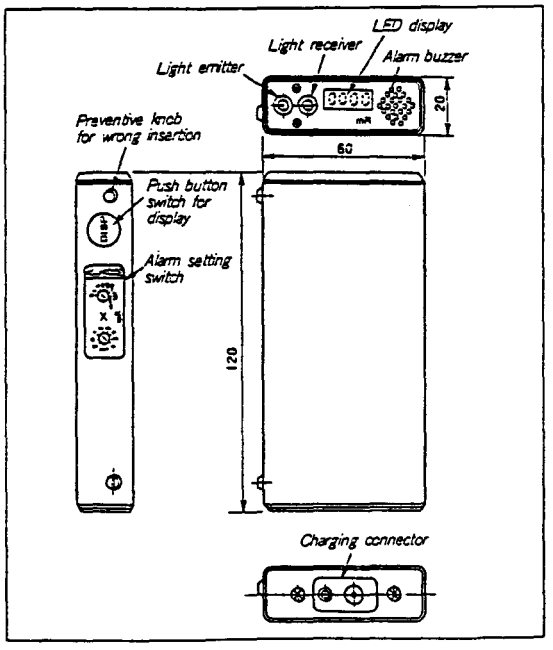

Fig.4. Dimensional drawing

TOWARDS AN AUTOMATED TLD SYSTEM THAT MEETS INTERNATIONAL REQUIREMENTS

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1. INTRODUCTION

The new recently introduced fully automated TLD system developed by Alnor OY on the basis of the Risø prototype (1), is intended to meet draft IEC/ISO proposals and ANSI requirements. Part of the system is a personal dosemeter badge and an environmental dosemeter package following ICRU recommendations.

The overall system consists of a software-controlled automated reader, a programable irradiator/calibrator, a computer, and dosemeters for environmental, whole body, extremity and clinical applications.

The personal TLD badge that contains four TLD pellets is designed to agree wigh ICRU $H_p(10)$ and $H_s(0.07)$ quantities for determining dose equivalent. The badge can accommodate a large variety of the most commonly used solid TL dosemeter products. A special effort was put into the evaluation of skin dose by considering the use of graphite-mixed hot-sintered LiF pellets (2).

The TLD system is described and results from a performance test that comprised measurements of photon energy response, angular dependence, and reproducibility are presented.

2. SYSTEM CONFIGURATION

2.1. The Alnor TLD reader and irradiator/calibrator

The automated Alnor TLD system consists of a software-controlled reader unit and a programable Sr/Y-90 irradiator/calibrator unit both of which can be loaded with identical cassettes containing dosemeter cards.

The read-out system is based on heating solid TL detectors with a hot nitrogen jet. The reader is equipped with a high-performance single photon counting system, and it measures and files raw glow-curve data by means of an on-board computer. Alternatively, data can be transferred to a host personal computer adapted to specific user requirements with software. The beta irradiator/calibrator unit can be programmed to give either equal doses for normalisation of individual TL detector sensitivities or stepwise increased individual doses to obtain response versus dose calibration curves.

2.2. The Alnor TL dosemeter card

The new Alnor TL dosemeter card consists of a two-part moulded plastic unit developed for automatic processing. The dosemeter slide contains four solid TL detectors kept in depressions and an identification number with corresponding hole code. The slide cover has four positions, three for interchangeable filters and a fourth as an open window position (3.5 mm diameter) with 45° shaped edges.

Fig. 1. The Alnor TLD personal badge, dosemeter card and finger dosemeter.

Any solid TL dosemeters with max. dimensions of 4.5 mm diameter x 0.9 mm can be used with the TL dosemeter card. Twenty dosemeter cards are contained in each cassette and ten cassettes may be loaded in an automatic cassette feeder system that is also provided.

The dosemeter card fits into the Alnor personal TLD badge made of plastic material (see Fig. 1).

For environmental monitoring purposes dosemeter cards containing four LiF dosemeters are kept in a welded plastic bag to withstand weather conditions. Each environmental dose is thus determined from the mean value of four individual dosemeter responses.

3. SYSTEM PERFORMANCE

A single photon counting technique in connection with the time-controlled multiscaling system of the Alnor reader produces glow curves as histograms over typically 120 counting channels. Channels at low- as well as high temperatures can easily be discriminated by the software resulting in a fraction of the glow curve being centred around the stable glow peak. A proper discrimination can thus exclude both unstable low-temperature peaks as well as thermal background signals at high temperatures.

Experience with repeated measurements has shown that as far as low dose measurements are concerned oven annealing of LiF materials can be replaced by annealing performed as a normal read-out in the reader when combined with a proper time discrimination of the TL glow curve, whereby even a higher reproducibility is obtained.

Reproducibility of the Alnor TL system including the Alnor beta doser was tested with repeated measurements of batches of different TL dosemeters exposed to a beta dose equal to 1 mSv Co-60 gamma irradiation. No annealing other than a normal read-out was applied. Low-temperature peaks and thermal background were removed by software time discrimination of the obtained glow curves.

Measurements were carried out each day over a 7-day period with LiF, TLD-700, hot-pressed round pellets, and $Li_2B_4O_7$:Mn, Alnor, hot-sintered round pellets. At the same time the reproducibility of the LED stability test facility of the Alnor reader was tested. The results are listed in Table 1. Figure 2 shows a typical glow curve of LiF, TLD-700 with the selected time discrimination indicated by the painted fraction.

Table 1. Reproducibility of the Alnor TLD system. Dose: Alnor beta doser (~1 mSv Co-60).

Day no.	$Li_2B_4O_7$:Mn (Alnor)	LiF-7 (Harshaw)	Light source (LED)
1	1.000	1.000	1.000
2	0.999	0.976	1.000
3	0.992	0.988	1.001
4	0.986	0.976	1.000
5	0.996	0.981	1.004
6	0.994	0.976	0.993
7	0.967	0.971	0.989
Mean	0.991	0.981	0.998
SD(%)	1.2	1.0	0.5

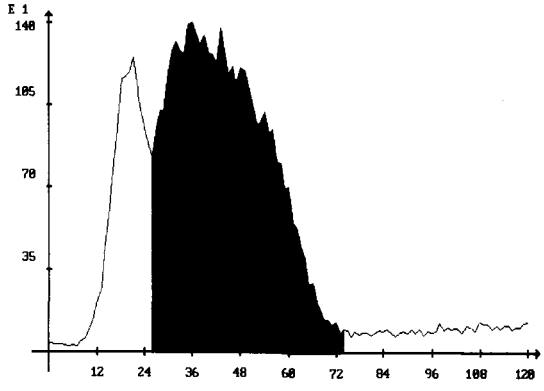

Fig. 2. Glow curve of LiF, TLD-700. The intregration area is indicated. (Dose ~ 1 mSv Co-60).

4. DOSIMETRIC PERFORMANCE OF THE ALNOR TLD BADGE

The dosimetric performance of the Alnor TLD badge was investigated for exposures to photon and beta radiations. For personal monitoring the ICRU quantities $H_p(10)$ and $H_s(0.07)$ are considered for photon radiations and $H_s(0.07)$ for beta radiations.

Photon energy response curves were obtained with different dosemeters contained in the new Alnor dosemeter card equipped with different filters (one window position) and attached to a cubic perspex phantom with dimensions 300 x 300 x 300 mm. Photons were ISO quality X-rays of 17 keV, 33 keV, 48 keV, 65 keV, 100 keV, 161 keV, and 248 keV in addition to gamma photons from Cs-137 and Co-60 sources. The measured values were converted from cubic to spherical shape by calculations following the procedure of Grosswendt et al. (3).

An example of the photon energy response of the depth dose and skin dose detectors of the Alnor TLD badge equipped with LiF, TLD-700 dosemeters, and 1 mm aluminium filters (one window position) compared with the energy dependence of the ICRU conversion coefficients that convert the dose in free air to $H_p(10)$ and $H_s(0.07)$ quantities is illustrated in Fig. 3.

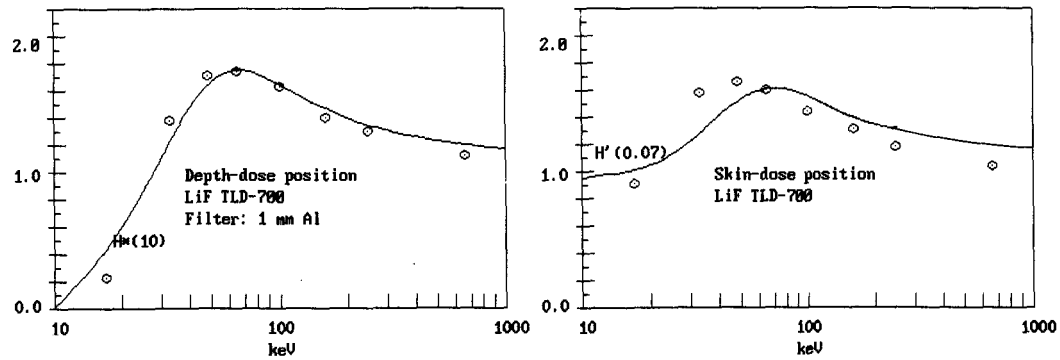

Fig. 3. The photon energy response of the Alnor TLD badge compared with the ICRU $H_p(10)$ and $H_s(0,07)$ quantities. (O = measured values.)

The angular dependence of the TLD badge was measured with LiF TLD-700 dosemeters. TLD badges were attached to a cubic perspex phantom that was turned in two planes and exposed to 65-keV X-rays. The results compared with conversion factor values given by Grosswendt et al. (3) are presented in Table 2.

Table 2. Angular dependence of the Alnor TLD badge (LiF, TLD-700). Irradiation: 65 keV X-rays.

	Horizontal plane				Vertical plane			
Angle(α)	0°	20°	40°	60°	0°	20°	40°	60°
Alnor (meas.)	1.00	0.98	0.93	0.85	1.00	0.98	0.97	0.87
PTB($H10,\alpha$)	1.00	0.99	0.94	0.83	1.00	0.99	0.94	0.83
Alnor/PTB	1.00	0.99	0.99	1.02	1.00	0.99	1.03	1.05

5. CONCLUSION

The new Alnor TL system has proved to have an excellent stability, and a high flexibility is achieved due to the ability of the dosemeter card to accommodate a large variety of different filters and TL dosemeter products. The measurement results demonstrate that the system satisfies international requirements well.

ACKNOWLEDGEMENT

We wish to express our gratitude to K. Ennow, Danish National Institute of Radiation Hygiene, for his valuable assistance in providing the ISO quality X-ray irradiation facilities.

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CONTAMINATION CONTROL: INSTRUMENTATION AND MEASUREMENT TRENDS IN THE USA

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ABSTRACT

Microcomputer-based systems using large area gas-flow proportional and scintillation detectors have greatly reduced the need for hand-probe frisking at most nuclear power plants in the USA. Instrumentation purchased in recent years for contamination control takes advantage of microcomputer-based electronics to automatically update background and set alarm points at statistically significant levels above background.

The driving force for trends toward improved instrumentation has been reduction in cost for contamination control and waste disposal. A side benefit has been improved performance, with more likelihood of contaminated material being detected. Replacement of hand-probe frisking with total body personnel contamination monitors has saved enough time during a single reactor shutdown to pay for the new equipment. Similar savings have been achieved using box-type tool contamination monitors.

One USA trend is to reduce the volume of low level waste (LLW) by using a combination of a sorting table and conveyorized contamination monitor, with final checking in a box-type bagged waste monitor. These waste monitors use gas-flow proportional and scintillation detectors to provide assurance that significant amounts of contaminated material will not be missed.

The latest generation of laundry monitors is similar to the conveyorized waste monitors. Gas-flow proportional detectors are being used in applications that require the detection of alpha and beta contamination. A combination of gas-flow proportional and scintillation detectors is being used at nuclear power plants to detect hot particles and distributed contamination.

Five years of almost daily use has proven the value of computerized contamination monitors installed since 1982 in nuclear power plant and U.S. Government facilities.

Recent Biological Developments of Importance in Radiation Protection

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In the decade since the last major recommendations of the ICRP (ICRP, 1977) developments have occurred in all phases of our knowledge of radiobiological effects. Many of these have a direct impact on radiological protection and must be accounted for in any revision of the ICRP's basic recommendations.

Among the many topics that ICRP, and especially its Committee I, has under consideration for the revisions of Publication 26 (ICRP, 1977), a few representative items are selected for discussion. These include the risk of cancer induction, the risk of the induction of mental retardation, effects on the skin, and quality factor.

THE RISK OF CANCER INDUCTION

Cancer induction has emerged as the primary stochastic effect of radiation at low doses which must be considered in radiation protection. (Genetic effects are also of great importance but require a separate discussion.)

Changes Since 1977

Additions have occurred in the data on cancer risks since the evaluations of UNSCEAR (UNSCEAR, 1977) and ICRP (ICRP, 1977). These include three cycles of data on the Japanese survivors of the A-bombs 1975-78, 1979-82, 1983-85 and further information on clinical studies such as the ankylosing spondylitics and carcinoma of the cervix cases treated by radiation.

New information on the age dependence of breast and thyroid tumors has become available and the importance of the age dependence of risk has been more generally realized. Younger people (<18y) are more sensitive to the induction of cancer than older people. Thus, the sensitivity of a worker population of 18-65 years is appreciably less than that of a population of all ages. This must be accounted for in a radiation protection system.

Revisions in the Japanese dosimetry have recently been completed. The effect of the new DS86 (Roesch, 1987) as compared with the former T65D is to increase risk for organs by a factor of 1 to 2x depending on whether the organ is deep in the body (~ 1x) or shallow (~ 2x) (and also on the RBE assigned to the neutrons) (Preston & Pierce, 1987). There have also been improvements in methods of analysis.

Recent Risk Estimates

In the latest interpretations of the Japanese data (Preston & Pierce, 1987) the risk coefficient for leukemia is approximately

doubled. An increase of about 2 1/2 is estimated for the risk of nonleukemic tumors due to the DS86, longer follow-up, and changes in statistical methodology and projection. The overall risk coefficient for a population of all ages is about 5.5%/Sv, assuming a dose reduction factor of 2.5, although linearity or nonlinearity in the dose response is not fully treated in this paper. These preliminary results indicate that risk coefficients will be higher than those estimated in the 1977-80 period.

Weighting Factors

In 1977, ICRP used a relatively simple weighting factor system involving the gonads, five tumor sites and a complex remainder group. Now the list of tumor sites, (NIH, 1985), has expanded to 12 and there is much more information on the risks of each, including their age dependence. Again, an important question is what to include concerning the genetic risk.

Comprehensive Evaluations of Risk

After ICRP in 1977, the US BEIR III Committee (BEIR, 1980) had preferred values of risk similar to UNSCEAR (UNSCEAR, 1977) but provided wider ranges, especially at the higher risk end. Later an NIH group (NIH, 1985) re-evaluated a number of tissues, provided a new list of those sites susceptible to radiation induction and chose linear dose response for thyroid and breast and linear quadratic for other sites. Estimates of risk based on these data for adults are about twice those used by ICRP (ICRP, 1977).

Major comprehensive re-evaluations of risk undertaken by UNSCEAR and by the new BEIR V committee are due to be completed early in 1989. Then groups such as a Task Group of ICRP Committee I, can determine which coefficients to use for radiation protection purposes.

MENTAL RETARDATION

The ICRP was advised by J. Schull in late 1981 of new evidence on periods of vulnerability to the induction of mental retardation by radiation in the human embryo-fetus (Schull, 1981). The most sensitive period is 8-15 weeks after fertilization and a second less sensitive period is 16-25 weeks after fertilization. The embryo-fetus is insensitive to this effect before and after these periods. ICRP set up a Task Group in 1982 to study this important problem.

These observations (Otake & Schull, 1984), also showed an approximately linear shape of the response with dose for the 8-15 week period. A recommendation was made by ICRP (ICRP, 1983) to the effect that, in addition to maintaining the overall limit, radiation received by the fetus should not be received irregularly. Thus, the dose that could be received in the most critical two-month period could not exceed 1 mSv, for which the risk was quite small.

The report of the ICRP's Task Group was published in 1986 (ICRP, 1986). The risk coefficients for the two sensitive periods (based on linearity of response) were about 40%/Gy and 10%/Gy

respectively. If the response were not linear the maximum possible values of threshold were 0.1 Gy for the first period and 0.5 Gy for the second. The Commission re-emphasized its recommendation concerning control of irregularities in the dose rate to avoid undue risk to the embryo-fetus in the sensitive periods.

A preliminary result of using DS86 instead of T65D indicates only a small difference in the risk based on a linear response but enlarges the possibility of a threshold.

SKIN

The skin has not been considered a very radiosensitive tissue in terms of cancer induction and skin tumors are relatively easily controlled. This may have contributed to radiation protection of the skin seeming less urgent than for some other tissues. However, exposure of the skin can sometimes be limiting, and lethal. Thus an overall appraisal of skin effects is needed which includes the relative importance of stochastic and nonstochastic effects, the depths in skin at which these occur and for which the dose is important, and the special problem of hot particles on the skin. ICRP Committee I has set up a Task Group to evaluate the situation. Hopefully its recommendations will be included in the revisions of Publication 26.

RADIATION QUALITY

A number of developments have taken place in matters relating to radiation quality since the publication of ICRP 26. Four aspects of this subject will be discussed.

Dose response relationships at low doses

The shapes of dose response curves for high-LET vs low-LET radiations yields maximum values of RBE, i.e., RBE_M, at low doses, viz. a few hundredths to tenths of a Gy (Sinclair 1985, 1986).

The larger values of RBE_M are not exclusively due to reduced effectiveness of the γ rays. Progressively lower neutron doses appear to give rise to steeper initial slopes which thus contribute to the increased RBE in the RBE_M region.

Difference between low-LET photons, 200 kVp x and γ rays

In the low dose region 200 kVp x rays are about twice as effective as ^{60}Co γ rays, according to measurements in a number of biological systems (ICRU, 1986). Consequently RBE values are affected by which radiation is used as the reference radiation. Thus, a question is whether to lump all low-LET radiations at Q=1 or instead to specify either x or γ rays as that radiation with Q=1. Also concerned is for which radiation the risk coefficients for human carcinogenesis are best known.

Values of RBE for neutrons and other high-LET radiations at low dose

Laboratory work has recently extended the evaluation of RBEs into the low dose region and in a number of systems values of RBE_M

have been determined. These values cover a wide range but are generally high, as we expect with RBE_M. A table given by ICRU (ICRU, 1986) indicates that for fission neutrons vs. fractionated γ rays, the average RBE_M is about 50 and presumably therefore about 25 vs. x rays. Furthermore, when α's and fission neutrons are compared directly, they often give rather similar results. Thus the recommendation already made by ICRP (ICRP, 1985) to increase the neutron Q from 10 to 20 seems well justified.

RBE for nonstochastic effects

For larger accidental or emergency exposures, Q does not apply and other values of RBE are needed to express the equivalent dose for high-LET radiations. These RBE values should be appreciably smaller than RBE_M and may depend on dose. A Task Group of Committee I of ICRP is studying this problem and hopefully its report will be incorporated into the revisions of Publication 26.

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ICRP AND THE RADON PROBLEM

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Radon, usually represented by radon-222, has long been recognized as a source of radiation risk. Lung cancer among miners in the Schneeberg-Jachymov region in the Erzgebirge was diagnosed in 1879. This was the "Schneeberger Krankheit" that is known already from the 15th-16th century. Its association with radon was suggested in the 1920's.

Over the last few decades, there has been a growing concern about the occupational radon problem in mines; not just in uranium mines but, as was found in the 1970's, also in other mines. For the public, new types of buildings, with reduced ventilation to reduce the heating cost and spare fuel, and with directly accessable basement space, exposed to emanation from the ground, have caused increased radon concentrations in indoor air. Values exceeding 10,000 Bq/m3 have been found and the problem has now been recognized in most countries with sufficiently cold climat to make limited ventilation a practical necessity.

The 1982 UNSCEAR report, with its presentation in terms of the effective dose equivalent, made it obvious that radon exposures, even under "normal" conditions, contribute substantially to the annual effective dose equivalent. The previous value of 1 mSv per year from natural background radiation was suddenly raised to 2 mSv per year. It is against this development that the work of ICRP in relation to radon and to the natural radiation background in general should be seen. A conceptual problem is involved: Can and should natural sources of radiation be controlled? How should this be done? What is "natural"?

RISK ASSESSMENTS

The risk from radon is almost exclusively a lung cancer risk. "Exposure" in this connection is usually taken to mean either the time integral of the potential alpha energy concentration in air inhaled by the exposed individual, or the time integral of the equilibrium equivalent concentration (EEC) of radon in that air.

The radon daughter exposure of miners is often expressed in the unit WLM (working level months), which, for radon-222 relates to the time integral of the potential alpha energy concentration and the activity concentration by: 1 WLM = 0.0035 Jh/m3 = 630,000 Bqh/m3

There are two different models for risk projection. The relative risk model is based on the assumption that the risk per unit time, after a latent period, will be some fraction of the age-specific lung cancer mortality, whatever that would have been without the radon exposure. For the absolute risk model it is assumed that the future increment of the age-specific lung cancer mortality is independent of the normal lung cancer mortality and only a function of the radiation dose and the age at exposure.

Once a model has been chosen, there are also two different ways of assessing the risk. One is by epidemiological studies, the other by calculating the effective dose equivalent and applying the usual risk factor (about 2 % per sievert).

The epidemiological approach is the most direct one and has been used in assessing the lung cancer risk for miners. In ICRP Publication 50 (1987), the risk-exposure coefficient is believed to be in the range of 5 - 15 cases per million personyears and WLM on the basis of the absolute risk model, and 0.5 - 1.5 % per WLM on the basis of the relative risk model.

For members of the public, the epidemiological method has not been available directly, because of the practical difficulties. However, the epidemiological findings from the studies on miners have been used to derive indirect risk estimates, taking into account factors such as differences in the breathing rate. This derivation of the public risk from the risk of miners is subject to a number of uncertainties because of obvious differences in the exposure situations.

The alternative method, the dosimetric method, however, also involves great uncertainties. One is the identification of the relevant tissues for which the dose should be calculated. In ICRP Publication 50, it has been assumed that the cells at risk are located in the basal and mucus cell layer of the tracheobronchial epithelium and the pulmonary region.

The influence of smoking habits is of particular importance. Some epidemiological results suggest a more than additive influence of smoking which can be approximated by a semi-multiplicative model in the case of heavy smoking and high radon exposures.

The conclusion of the Task Group preparing ICRP Publication 50 was that an appropriate reference value for the life-time risk of lung cancer from chronic indoor exposure to Rn-222 daughter products would be an <u>absolute</u> risk of

$$2 \times 10^{-8} \text{ per Bqh/m}^3 \text{ for each year of exposure}$$

based on the absolute risk model, and a <u>relative</u> risk of

$$0.8 \times 10^{-6} \text{ per Bqh/m}^3 \text{ for each year of exposure}$$

based on the relative risk model. The two estimates would be equal if the mean life-time risk of lung cancer (from other causes) were 2.5%.

OCCUPATIONAL LIMITS

Various exposure limits and "maximum permissible concentrations" of radon in air inhaled by miners have been recommended by ICRP since 1954. In ICRP Publication 24 (1977), the Commission made clear that it is not the radon equilibrium equivalent concentration (EEC) at any given time, but the average concentration over a year, that must not exceed the MPC-value. In ICRP Publication 32 (1981) new limits for radon concentrations and exposures were recommended, namely, for radon concentration (EEC):

$1500\ Bq/m^3$ or 0.4 WL

and for annual exposure to radon daughters:

$3 \times 10^6\ Bqh/m^3$ or 4.8 WLM

ICRP Publication 26 put the emphasis on optimization of radiation protection. In ICRP Publication 37 (1983) an Appendix dealt with variables associated with radon contamination and their influence on the optimization process, and algebraic examples were given of optimization procedures, e.g. the design of ventilation.

ICRP Publication 47 (1986) is a revision of Publication 24 on the basis of information in Publications 26 and 32. The annual limit of intake (ALI) is expressed as 0.02 joule or 3.6 MBq. The annual limit of exposure and the derived air concentration (DAC) are the same as in Publication 32.

RECOMMENDATIONS WITH REGARD TO RADON IN DWELLINGS

Since ICRP Publication 1 (1958), the Commission has assumed that the risk resulting from exposures received from natural background radiation should not affect the justification of an additional risk from man-made exposures, because, as long as the probability of stochastic harm is proportional to the dose, the risks due to different sources of exposure are simply additive.

It was thought at the time that it was possible to define a "normal" natural background and that the limits only referred to doses resulting "from technical practices that add to the natural background radiation", such as mining and flight at high altitudes.

However, evidence of very high indoor concentrations of radon in dwellings accumulated rapidly. It was found that, in many cases, people received, in their homes, lung doses that would not have been accepted even for uranium miners. Something had to be done, but if the background dose were to be included under the normal dose limit the problem was obvious; if not, there was no other recommendation on dose limitation.

In its Publication 39 (1983), the Commission concluded that the distinction between normal and enhanced levels of exposure is not helpful and decided to base its new advice on a different approach. A more useful distinction, the Commission said, is between existing exposure situations, where any action would have to be remedial, and future situations, which can be subject to control at the stages of decision and planning.

For existing situations the Commission recommended the use of an action level specific to the initiation of the remedial action being considered. The concept of "existing situation" would also be valid for emergency situations after any accident that causes environmental contamination, where, for example, remedial actions against food-stuffs may be appropriate.

"A well chosen action level", the Commission said, "will take account of the likely effectiveness of the action to which it will

lead and of the total detrimental consequences of the action, which may not be limited to financial costs. The persons at risk should also be put in a better position by the action.

The Commission found it impossible to suggest a generally applicable value of an action level, even one limited to the specific case of radon in houses. However, if the remedial action considered is fairly simple, the Commission said, an action level "in the region of 200 Bq/m3" for the radon-222 concentration (EEC) might be considered. The Commission estimated that the corresponding annual effective dose equivalent would be about 20 mSv (which may now be considered an overestimate since the recommended dose coefficient is 0.06 mSv/year per Bq/m3).

For future situations involving exposure to natural sources of radiation, "the position is closer to that recommended by the Commission for artificial sources of exposure" says the Commission in paragraph 24 of Publication 39. The basic principles of justification of practice and optimization of protection should therefore apply. The main question was whether the normal dose limits should apply or not.

In the case of workers, the dose limit applies to occupational exposures irrespective of source, although exposure from "normal" natural background is at present not included under the occupational dose limit. A current problem for the Commission is to decide whether such exposures should be included in the future if it becomes increasingly more difficult to identify what is "normal". At present, doses from natural sources in mines are included, but not doses from high indoor radon concentrations at other work places.

For members of the public, the limitation principle is different. Their exposures are not controlled individually but by requirements related to the sources. The primary dose limit may be seen as a limit of the contribution from a particular selection of sources, explicitly excluding natural sources and medical irradiation of patients.

Since the normal dose limits were not intended to cover the doses from natural sources, the Commission decided to introduce special constraints for the optimization of protection in the case of future sources, and recommended that competent national authorities should establish an appropriate upper bound of individual dose in optimization assessments. The need of this constraint was obvious. Indoor radon concentrations (EEC) of more than 10,000 Bq/m3 had been measured in dwellings in some countries, implying annual effective dose equivalents of more than 10 times the Commission's dose limit for workers.

The Commission believed that 100 Bq/m3 would be a reasonable upper bound for the radon concentration (EEC) in new buildings, and that in many countries, it would prevent radon from becoming a dominating source of risk in dwellings. It was expected that the use of this upper bound in the optimization assessment of radiation protection in new houses would influence building standards for constructure, manufacture, ventilation, etc.

ICRP ACTIVITIES IN MEDICINE - THE DEVELOPMENT OF COMPREHENSIVE GUIDES TO GOOD PRACTICE

Roger J Berry, International Commission on Radiological Protection.

The special relationship between the practice of medical radiology and the ICRP is no accident. The International Commission on Radiological Protection was established in 1928 under the name "International X-Ray and Radiation Protection Committee" by the Second International Congress of Radiology Meeting in Stockholm. It assumed its present name and organisational form in 1950 in order to cover more efficiently the expanding field of radiation protection. In its most recent recommendations, contained in Publication 26 (1977) a new system of dose limitation was elaborated, based on three interlocking elements. The Justification, Optimisation and Dose Limits interact to keep radiation doses to both occupationally exposed persons and to the general public "As Low As Reasonably Achievable, economic and social considerations being taken into account". Although medical uses of ionising radiation were excepted from dose limits because it was felt that the degree of safety achieved was now so high than an X-ray examination, recommended on the basis of qualified clinical judgement, generally brought a benefit to the patient which entirely outweighed the unavoidable radiation risk, the Commission felt that it would be helpful to disseminate in a more digestable form the specific application of its recommendations to modern medical practice.

In 1981 Committee 3, formerly the Committee on External Radiation, became a Committee on Radiation Protection in Medicine and in its new guise became more widely representative of practicing radiologists, nuclear medicine physicians and radiation oncologists. As a result of its activities, a series of ICRP publications have now appeared on protection of the patient - in Diagnostic Radiology (Publication 34), in Radiation Therapy (Publication 44) and in Nuclear Medicine (in Press). Unlike other

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ICRP publications, these are an attempt to provide readable but reasonably comprehensive guides to good practice, and are targetted at the practising radiologist and clinician and not at National Authorities.

Diagnostic X-ray procedures are the cause of the majority of man's exposure to ionising radiations from artifical sources. The Commission identified the need for all radiological examinations to be justified. On the principle that all radiation doses should be kept "As Low As Reasonably Achievable", the Commission has developed principles which allow the assessment of how far it is reasonable to go in cost and efforts to reduce radiation exposure. These methods apply to protection of the patient in diagnostic radiology no less than in any other planned exposure of man to radiation. The limitation of risk to the individual patient is usually implicit in the medical decision that a particular examination is in the interest of the patient. The optimum use of equipment and techniques then make any explicit limits of radiation dose for diagnostic radiology inapplicable. If each individual examination is properly justified, the collective risk is by necessity also justified. There is thus no reason to limit the total collective radiation dose from medical exposures below any value that would simply be the sum of the individual doses from appropriately performed examinations. However, it is still necessary to assess the collective doses from various medical procedures since this gives a useful indication of where protective measures related to design or choice of procedure might have a large impact in reducing overall population radiation exposure. Publication 34 outlines the background to the protection of the patient in diagnostic radiology, from clinical judgement through education and manpower requirements to the technical and physical factors, and gives a detailed method for the determination of organ doses from diagnostic X-ray examinations which can be used as a basis for optimisation of individual and collective doses.

For radiation therapy, the problem is more complex. Protection of the patient in radiation therapy requires, uniquely, not the avoidance of radiation exposure or even the avoidance of risk of severe damage to some tissues. Rather, once the choice is made that radiation is the appropriate treatment, it involves achieving

the optimal balance between the efficacy of irradiation in achieving sterilisation of the malignant growth and the production of the minimum treatment-related complications. This latter involves making unwanted radiation doses As Low As Reasonably Achievable. The process of protecting the patient requires optimisation of the treatment as a whole, optimisation of protection alone is not enough. This is detailed in Publication 44 which is not designed to be a text book of radiation therapy. Rather, it presents a broad picture which is readable by all involved in the care of cancer patients. This includes physicians, such as medical oncologists who will not themselves use radiation therapy but who need to know about its potential benefits and risks. It includes also the members of the radiation therapy "team"; medical physicists, radiographers, dosimetrists and also the administrators who need to understand the requirements of their clinical specialist colleagues. The Publication covers general principles of radiation therapy, the specific problems of external beam and brachytherapy, gives a guide to expected risk to specific organs/tissues from non-stochastic radiation injury, as well as discussing carcinogenesis, and gives guidance on doses to organs and tissues outside with the useful treatment beam for radiation field arrangements used to treat some of the most common tumours.

Once again in clinical nuclear medicine the problem is different in that although the technique can give unique information about the function of organs, the body tissue most heavily exposed to radiation is often not the organ being studied and is more likely to be in the path of either intake or removal of the specific radioactive tracer material from the body. Reduction of the dose may be associated with a loss of diagnostic information, and an optimisation procedure is required to maximise the amount of clinically relevant information obtained from each examination. Because radioactive material is administered to the patient, the patient represents a hazard to other members of the public, and both the design of clincal facilities and the organisation of departmental work should be tailored to minimise its potential hazard. This is discussed in detail in the publication.

In all these ICRP publications it is stressed that the establishment of measures for patient protection are not intended to impede the continued development of the clinical uses of radiation, both diagnostic and therapeutic. It is expected that such measures will actually contribute to higher standards of clinical practice, yet, it is made clear that where there are limitations on resource, both material and personnel, so that the ICRP recommendation cannot yet be met, patients should not be denied the necessary diagnostic examinations or radiation treatment.

For the future, Committee 3 is drafting a further simplified version of the Publication on Protection of the Patient in Diagnostic Radiology aimed specifically at non-radiologist clinicians, radiologists and radiographers, and it is hoped that both it and a single composite volume containing the three existing "Guides to Good Practice" on protection of the patient in diagnostic radiology, radiation therapy and nuclear medicine will be available to participants at the next International Congress of Radiology to be held in Paris in 1989.

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EVOLUTION OF THE ICRP RADIATION PROTECTION PHILOSOPHY

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ABSTRACT not available

BEAGLE DOG STUDIES IN RADIATION PROTECTION

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INTRODUCTION

Beagle dogs have been used as experimental subjects in studies sponsored by the U.S. Department of Energy (DOE) and its predecessor organizations, since the establishment of the Atomic Energy Commission in 1947. Major longterm studies began with two contracts let in 1950-51 to the University of California at Davis (UCD) and to the University of Utah (UU). In 1954 the Battelle Pacific Northwest Laboratories (PNL, then contracted under General Electric Company) began inhalation studies utilizing dogs, and in 1961 the Lovelace Foundation Fission Product Inhalation Laboratory (now the Inhalation Toxicology Research Institute, ITRI) was founded. The Argonne National Laboratory (ANL) did some limited studies using the Beagle dog in the 1950's, and its current longterm external radiation studies began in 1968. The first four studies have dealt primarily with radionuclides, except for early work at UCD. Early studies at ANL were with radionuclides but they are currently committed to external radiation. Various routes of entry of radionuclides have been used; the radionuclides have been primarily fission products and/or actinides; external radiation exposures have used neutrons and x- or gamma rays.

Almost all research with Beagle dogs in DOE programs has aimed at supplying data for the purpose of strengthening the bases for radiation protection guidelines. The studies have provided information ranging from the very basics of dosimetry to the definitive description of the resulting radiololgical effects. Thus, they are defined as dose-response studies, that explore the causes of radiation damage from the molecular to the whole body levels of investigation. What will be reported in the oral presentation is a more detailed review of the use of results from Beagle dog studies that have played an important role in radiation protection.

RESULTS AND DISCUSSION

The most recent compilation of radionuclide guidelines was in 1978 by the International Commission on Radiologiocal Protection (ICRP78). In reviewing the recommendations in ICRP-30 (ICRP78) a tally was made of the number of DOE Contractor publications used in establishing individual guidelines. Numerous publications cited were from the use of rodents in radionuclide metabolism, dosimetry, and toxicity studies. Of the total, 75% of the references were from DOE supported laboratories; most of the remaining 25% were publications from studies performed outside the United States. For 12 elements, dog data constituted the basis for the numerical value used for a critical parameter in derivation of Annual Limit of Intake or Derived Air Concentration. Most uses have been for organ distribution

and retention parameters or proper assignment of compounds to inhalation classes (D, W, or Y). For 33 elements, critical parameters were based on experimental animal data and/or human data that included comparative studies in dogs.

Respiratory tract particle deposition models (lung models) use many of the results obtained from studies with Beagle dogs to assign values of half lives for transport from lung to blood. Early lung models used 120 or 360 days for the retention halflife of insoluble particles in the deep lung (NBS53, TGLD65). Currently, the value used is 500 days, and the newer models may extend this. Present dog studies of the lung retention time for insoluble plutonium will doubtless indicate a longer half life. Only a species with a lifespan like the dog may be used for this type of simulation, as its respiratory physiological properties are so much like the human. The latest lung retention data for plutonium in the dog show some trends toward a power function retention, and this may well be enforced by recent findings in the human autopsy cases being analyzed. In the latter there is plutonium in lung tissue decades after exposure.

The sophistication that has been developed for characterizing aerosol properties to describe inhalation studies has advanced the field of aerosol physics. Use of the aerodynamic diameter to quantitate deposition in the respiratory tract has become commonplace. Mercer revolutionized the ideas concerning particle solubility in the respiratory tract by relating the half life of movement to blood in terms of the particle properties (Me67). His work has been applied to many studies in which chemical characteristics of the particle inhaled and its subsequent biological behavior could be predicted through in vitro solubility studies. Such studies show promise for use of on-the-job in vitro solubility measurements on aerosol samples collected at an operation, to predict the class (D, W, or Y) of inhaled compound being considered a potential health hazard.

Dosimetry in radionuclide studies is difficult and much progress has been made over the past few decades, that has direct implications on radiation protection standards. With radionuclides, the chief experiments aimed at dosimetry have been known as distribution-excretion or dose pattern studies. These generally require sacrifice of an animal after exposure so that organ and tissue distribution of the radionuclide can be determined as a function of time. When sacrifices cover a long period post-exposure, the pattern of dosage that will give rise to the biological effect is calculated.

In addition to information for the major organ systems, the parameters associated with the tracheobronchial lymph nodes play an important part in hazards evaluation following inhalation. ICRP-26 (ICRP77) considerations of guidelines for estimating the risks and exposure limits for the tracheobronchial lymph nodes have been largely based on data from dog studies that showed high concentration, long retention times, and therefore, very high radiation doses in the nodes of dogs that inhaled insoluble radionuclides. Limited human autopsy data support the dog data in that the tracheobronchial lymph nodes appear to receive the highest radiation dose

for many inhaled insoluble radionuclides. However, data from dog studies and limited human data at that time showed no neoplastic or life shortening effects of the radionuclides in the tracheobronchial lymph nodes. Based on this information, the Commission (ICRP77) decided that irradiation of the lung is likely to be more limiting than that of lymphoid tissue for inhaled insoluble radioactive particles and that it would be satisfactory to consider the pulmonary lymph nodes and lung as one composite organ.

Biological effects studies have assumed many different approaches over the years, but almost all have depended upon an investigation by the patholologist using the microscope to determine cause of death. Gross tumor incidence has been the important endpoint for estimating risk from radiation exposure, with newer sophisticated methods currently being used to more definitely determine cause of death. Because the ultimate endpoint of interest is carcinogenesis, however, this has been the prime target of the longterm studies, whether from external or internal radiation. With the availability of multipurpose blood testing techniques, clinical findings have also become an integral part of the biological effects studies. Each dog on experiment receives a periodic physical examination, the clinical results of which may bear a relationship to carcinogenesis. The ultimate goal in these studies is to derive a value for the number of cancers that may be expected for each man-year-dose of radiation exposure. These are the constants of importance in calculating new guidelines.

Although it might seem that risk parameters in standard-setting are derived almost entirely from human data, these parameters are supported by a great deal of animal data. The risk factor for lung cancer employed in arriving at a weighting factor in the ICRP system is derived from human experience, but without animal data it would not even be known that plutonium or other radionuclides, could produce lung tumors. Quality factors could be derived from stopping power calculations, but if these were not supported by experimental observations in animals there would be little confidence in them. It was easier to explain the dependence on animal data for bone-seeker standards in the old ICRP system where "distribution factors" were employed; now doses are calculated to bone surfaces to avoid the direct dependence on animal data, but the results are believed because they predict what was observed to happen in animals.

The risk estimates that are now being calculated from longterm dog experiments are likely to be the only ones available that incorporate well-documented exposures to radionuclides. Within the past few years, risk factors have been obtained for beta emitters deposited in Beagle dogs by inhalation, in longterm studies at ITRI. Some of these values will be used for these radionuclides when guidelines are again updated. Values for risk have been derived for alpha emitters at the University of Utah for bone-seeking radionuclikes in the Beagle dog. Completion of the lifespan dog studies in progress at PNL and ITRI will yield more accurate values for alpha-emitters in lung, and for combinations of effects in non-respiratory tract organs. Results from the ANL low level, chronic, whole body gamma ray exposures of Beagle dogs will doubtless play an important role in establishing such risk factors for external radiation.

CONCLUSION

This has been a general overview of the derivation of radiation protection guidelines from experimental data. Beagle dog studies have been utilized as a major data source for evolution of such guidelines, and this will continue to be the case until they are completed and the final data analyzed.

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LIFESPAN DOSE-EFFECT RELATIONSHIPS FOR INHALED $^{239}PuO_2$ IN BEAGLES*

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ABSTRACT

The objective of this research was to obtain, in an experimental animal, dose-effect relationship data for inhaled plutonium, that could be used to decrease the uncertainties for predicting health effects risks in humans. Young-adult beagle dogs were given a single inhalation exposure to a $^{239}PuO_2$ aerosol. Dose-level groups of ~20 dogs received mean initial plutonium pulmonary depositions corresponding to radiation doses that were approximately 1, 10, 40, 150, 550, or 2900 times the maximum permissible lung dose for a plutonium worker. The dogs have been observed for 16 years after exposure. Plutonium was cleared from the lungs with a half-time of about 1000 days, compared to 500 days for the model used by the International Commission on Radiological Protection (ICRP) Publication 30. In dogs with initial pulmonary depositions of less than 3 kBq, 40% of the plutonium was excreted, 55% was translocated to the pulmonary lymph nodes (nodes draining the lung), and 5% was translocated to the liver and skeleton. These values compare with 80%, 15% and 5%, respectively, in the ICRP model. There was no indication of clearance from the lymph nodes, compared to 90% clearance in the ICRP model. There was a dose-related increase in incidence of lung cancer, severity of chronic lymphocytopenia, and severity of lymph-node damage in the five highest dose-level groups. The relationship of damage to the lymph nodes and lymphocytopenia with the plutonium content of the lymph nodes suggests that nonstochastic effects in the lymph nodes should be considered in estimating the risk of inhaled $^{239}PuO_2$. It further suggests that the ICRP procedure of including pulmonary lymph nodes with the lung for radiation protection models needs further evaluation.

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* This work was supported by the U.S. Department of Energy under Contract DE-AC06-76RLO 1830

THE RELATIVE EFFECTIVENESS OF INHALED ALPHA- AND BETA-EMITTING RADIONUCLIDES IN PRODUCING LUNG CANCER[1]

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Proper assessment of long-term human health risks associated with inhaled radionuclides requires knowledge of dose to critical cells and tissues and relationships between dose and effect for different biological end points. Results from epidemiological studies of exposed human populations provide important information for such assessments. However, because the types of exposures are limited, these results need to be supplemented with more detailed information on dosimetry and biological effects available through studies in laboratory animals and *in vitro* systems. To provide health risk 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). Results of two life-span studies in dogs involving inhalation of the beta emitter ^{91}Y in fused aluminosilicate particles or the alpha emitter $^{239}PuO_2$ are reported here (1).

MATERIALS AND METHODS

The dogs used were young adult (12-14 months of age), purebred beagles from the ITRI colony that weighed 6 to 13 kg at exposure. Each exposed dog received a single, brief (< 70 min), nose-only inhalation exposure to either a polydisperse aerosol of ^{91}Y in fused aluminosilicate particles (AMAD = 1.5 - 2.6 μm or one of three sizes of monodisperse aerosols of $^{239}PuO_2$ (AD = 0.75, 1.5 or 3.0 μm) labeled with ^{169}Yb. The corresponding control dogs inhaled a non-radioactive vector aerosol. Periodic whole-body counts in the early post-exposure period and excreta collections at selected intervals throughout the study were used to determine the deposition and retention of these radionuclides. 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, at which time, complete necropsies and histopathological examinations were performed. Reviews of the clinical and pathological findings were used to categorize the major biological findings related to the cause(s) of death and incidental findings.

[1]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.

Calculations of absorbed beta or alpha doses were based on whole-body retention, tissue distribution and excreta data obtained from dogs in the life-span studies and from dogs sacrificed in parallel studies involving inhalation exposure to aerosols having the same characteristics. The relative risks of lung cancer in both studies were computed from the times to tumor and dose as a time-varying function using a proportional hazards formulation similar to those described by Kalbfleisch and Prentice (2).

RESULTS AND DISCUSSION

Results from the parallel serial sacrifice studies demonstrated that once the early clearance phase associated with an inhalation exposure was completed (3-4d), the remaining ^{91}Y or ^{239}Pu was in the pulmonary region where the longest-term retention components had effective half-lives of about 50 and 1500 days, respectively (3,4). These retention results were used to compute organ average absorbed alpha or beta doses using each dog's individual initial lung burden and an estimated lung weight with blood equal to 0.011 times the initial whole-body weight. Because of the relatively short effective half-life of ^{91}Y, over 90% of the committed beta dose to lung was received within the first six months after the inhalation exposure. Conversely, the long effective half-life of the ^{239}Pu produced chronic alpha irradiation of the lung throughout the dogs' lives.

As of 30 September 1987, all dogs in the ^{91}Y study were dead and the dogs surviving in the ^{239}Pu studies were 8.6 to 10.7 yr after exposure. Table 1 summarizes the biological effects seen in dogs that inhaled ^{91}Y FAP or 0.75 μm particles of $^{239}PuO_2$. The biological effects seen in the dogs exposed to 1.5 or 3.0 μm particles of $^{239}PuO_2$ were qualitatively similar to those listed here for 0.75 μm particles except the highest exposure levels were deleted in the study with 0.75 μm particles, thus producing fewer early deaths than seen in the studies with 1.5 or 3.0 μm particles. For both ^{91}Y and ^{239}Pu, dogs exposed at the highest levels died of radiation pneumonitis and pulmonary fibrosis within the first 3 years. The lung doses to death in these dogs ranged from 150 to 600 Gy in the ^{91}Y study and from 6 to 82 Gy in the three ^{239}Pu studies. Beyond three years after exposure, lung cancer was the most prominent finding at death, being found in 32 of 56 later deaths in the ^{91}Y-exposed dogs with absorbed beta doses to lung ranging from 35 to 250 Gy. For the ^{239}Pu study, 28 of 36 later deaths involved lung cancer and the corresponding alpha doses to death were 3.1 to 80 Gy. All of the lung cancers seen in both studies were carcinomas except that three dogs in these studies had both primary sarcomas and carcinomas of the lung.

Relative risk functions in the proportional hazards formulations for lung cancer were estimated assuming that the baseline functions for lung tumor incidence at 0 dose were the same for all four studies. Functions of the following forms were used to estimate the relative risk:

TABLE 1. Biological Effects in Dogs that Inhaled Aerosols of ^{91}Y in Fused Aluminosilicate Particles or $^{239}PuO_2$

	^{91}Y FAP		$^{239}PuO_2$ (0.75 μm)	
Diagnosis	Number of Dogs	Death Days PE	Number of Dogs	Death Days PE
Exposed Dogs	96		48	
Early Effects (<3 yr.)				
Pulmonary Injury	40	113-1011	2	891,1035
Late Effects (>3 yr.)				
Pulmonary Injury	1	2890	14[b]	1181-1588
Lung Cancer	32	1115-5624	28[c]	1467-3867
Other Cancers	14[a]	1435-5052	5[d]	1961-3626
Non-Cancer	11	3843-5721	1	2007
Control Dogs	12		12	
Late Effects				
Lung Cancer	0		0	
Other Cancers	5	3363-5942	1	3609
Non-Cancer	7	3029-5386	2	1893,3006

[a]One dog had both a lung cancer and a TBLN cancer and another dog had both a lung cancer and a mammary cancer.
[b]Ten of these dogs also had lung cancer and are included in the "lung cancer" total.
[c]One of these dogs had both a pulmonary carcinoma and a sarcoma.
[d]Four of these dogs also had lung cancer and are included in the "lung cancer" total.

$$\text{Relative Risk } (^{91}Y) = e^{\beta_1 D(t)} \quad (1)$$

$$\text{Relative Risk } (^{239}Pu) = e^{(\beta_2 D(t) + \beta_3 D^2(t) + \beta_4 D^3(t))} \quad (2)$$

Figure 1 illustrates the datum points for individual dogs in the ^{91}Y study along with contours of the 10, 50 and 90% probabilities that a dog develops a tumor by the time and at the dose where they are plotted. These probabilities are conditional on the dogs surviving to the times shown. Similar plots were obtained for each of the ^{239}Pu studies. At low doses, the linear coefficients account for most (>90%) of the risk incurred. Comparison of these linear risk coefficients among the four studies indicates that all three $^{239}PuO_2$ exposure regimens were more effective in producing lung cancer than was ^{91}Y. The $^{239}Pu/^{91}Y$ ratios for the linear risk coefficients were 18, 15 and 10 for the studies involving 0.75, 1.5 and 3.0 μm, respectively. The ^{239}Pu studies are still in progress and the final results may result in some changes in these numbers.

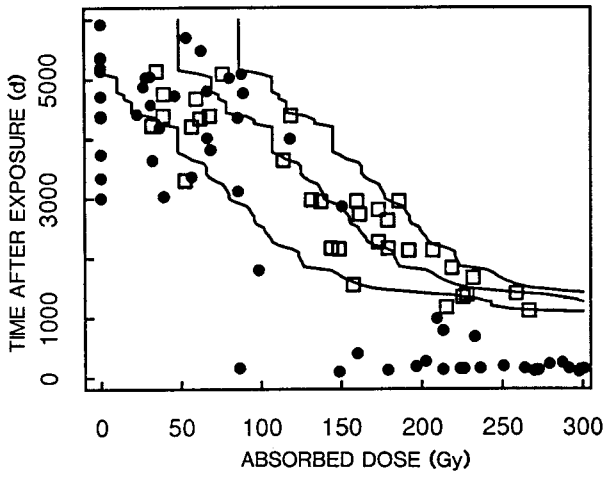

Figure 1. Plot of 10, 50 and 90% conditional probabilities of developing a lung cancer using a Proportional Tumor Incidence Rate Model for dogs that inhaled ^{91}Y FAP. Squares = lung cancer, circles = other deaths.

These long-term studies with ^{239}Pu and ^{91}Y in a long-lived species indicate that QF = 20 for internally deposited alpha-emitting radionuclides if a value of QF = 1 is used for beta emitters. Also, the differences in effectiveness among the three particle sizes of ^{239}Pu indicate that, at this stage in the completion of these studies, the tumorigenic effectiveness is directly related to the uniformity of lung irradiation. As predicted by many national and international bodies, a more uniform distribution of alpha irradiation of the lung appears to be more effective in producing cancer than more nonuniform distributions. Such experimental results, if borne out at the completion of these studies, provide direct refutation to theories that ascribe unusually potent carcinogenic properties to non-uniform distributions of alpha-emitting radionuclides in the lung.

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AGE EFFECTS ON THE INDUCTION OF RADIATION-INDUCED LUNG DISEASE IN BEAGLES THAT INHALED $^{239}PUO_2$ AEROSOLS: STATUS REPORT[1]

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Exposure of the general population to airborne radioactive materials would involve persons of widely differing ages. It is important, therefore, to understand the role of age at exposure in modifying both the biological effects from and the dosimetry of inhaled radionuclides. Due to the lack of adequate human data, we are studying the effects of age on both the organ doses received and the biological responses observed after single brief inhalation exposures. The results reported here focus on the biological effects being observed in studies in which immature, young adult or aged dogs inhaled monodisperse aerosols of $^{239}PuO_2$. The effect of age on dosimetric considerations has been previously described for the dogs of these studies (1).

MATERIALS AND METHODS

All dogs used in these experiments were purebred beagles in good health and derived from the Institute's colony. All dogs assigned to the experimental portion of the study received a single brief (< one hour) pernasal exposure to a monodisperse aerosol of $^{239}PuO_2$ (1.5 μm activity median aerodynamic diameter) and are being maintained for life-span observation. There were 96 dogs exposed at 90 ± 10 days of age (Immature) to achieve initial pulmonary burdens (IPB) ranging from 0.0085 to 21 kBq/kg body mass, a similar group of 96 dogs exposed at 13 ± 1 month of age (Young Adult) to levels from 0.0085 to 21 kBq/kg, and 48 dogs exposed at 8-10.5 years of age (Aged) to levels of 1.1 - 10 kBq/kg. Equal numbers of males and females were used; 36 dogs were exposed once to the diluent aerosol and serve as controls for the studies. 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 receives daily observation, annual physical and radiographic examination and semiannual blood cell counts and serum chemistry tests. 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 excreta samples were analyzed radiochemically for ^{239}Pu content by either alpha liquid scintillation counting or alpha spectrometry, depending on activity level of the individual sample.

[1]Research conducted for the Office of Health and Environmental Research of the Department of Energy under Contract No. DE-AC04-76EV01013 in facilities fully accredited by the American Association for the Accreditation of Laboratory Animal Care.

RESULTS AND DISCUSSION

As of 30 September 1987, all dogs had been on study at least 4.8 years for the immature dogs, 8.6 years for the young adults and 5.3 years for the aged dogs. The current status of these studies in terms of numbers of surviving animals and major biological effects noted to this date are summarized in Table 1.

Table 1
Status of Age-Effects Studies for Inhaled $^{239}PuO_2$ in Dogs

Study	Dogs on Study	Survivors	Pneumonitis*+ Fibrosis	Pulmonary* Carcinoma
Immature				
Experimental	96	72	5	15
Control	12	11	0	0
Young Adult				
Experimental	96	34	47	20
Control	12	12	0	0
Aged				
Experimental	48	2	34	4
Control	12	1	0	1

*Dogs that had both pneumonitis and/or fibrosis and carcinoma were counted twice.
+Cases of radiation pneumonitis and pulmonary fibrosis in which the disease contributed significantly to the death of the animal.

For the experimental dogs on study, the only radiogenic effects noted to date have been radiation pneumonitis, pulmonary fibrosis, pulmonary carcinoma, and fibrosis and atrophy of the tracheobronchial, mediastinal and sternal lymph nodes. These effects were expected given that the $^{239}PuO_2$ aerosols used have proven thus far to be insoluble in vivo and that the lungs and lung-associated lymph nodes have incurred most of the alpha- radiation dose (4). With regard to these tissues, few differences in the cumulative radiation doses have been found for the three age groups. For example, the cumulative radiation dose to lung to 1100 days after exposure for the inhalation of 1 Bq $^{239}PuO_2$ was 0.097 mGy for the immature dogs, 0.11 mGy for the young adults and 0.11 mGy for the aged dogs. Similarly, the doses to the tracheobronchial lymph nodes were 2.2, 3.3, and 2.4 mGy/Bq inhaled for the immature, young adult and aged dogs, respectively.

With regard to the biological effects of ^{239}Pu on the lungs of dogs exposed at different ages, there have been some significant differences noted thus far in terms of the relative incidence of inflammatory disease and lung cancer. For the aged dogs, all of the experimental animals that died of effects ascribable to the radiation exposure succumbed to radiation pneumonitis and pulmonary fibrosis. The pulmonary incidental findings.

In comparison, the dogs exposed as young adults also had a significant incidence of inflammatory disease (47 dogs); however, there also has been a significant number of cases of pulmonary carcinoma (20), which have included both incidentally occurring as well as fatal tumors. There also has been a difference in the temporal distribution of these two biological effects for the young adults, at least from the point of view of severity of the disease. Radiation pneumonitis was the dominant cause of death from about 100 to 2000 days after exposure. The first pulmonary carcinomas were observed as incidental findings at 1100 days after exposure. Thereafter, lung tumors became more important such that, during the period beyond 2000 days after exposure, both carcinomas and inflammatory diseases were important findings, with the former appearing to be the dominant effect at the latest survival times and the lowest exposure levels.

For the immature dogs, there have not been very many deaths associated with alpha radiation exposure (20). Of the small number, however, it is noteworthy that only 5 animals have died of radiation pneumonitis and pulmonary fibrosis, compared to 15 dogs that have died with pulmonary carcinoma. Additionally, the earliest occurrence of death from either cause was essentially the same, 1352 days after exposure for lung cancer and 1386 days for radiation pneumonitis. Thus far, both diseases appear to be continuing to occur; however lung cancer is already predominating.

Figure 1 (a and b) shows the incidence of radiation pneumonitis and pulmonary fibrosis for all of the dogs from the three age studies that died from this disease. When expressed in terms of kBq/kg body mass initial pulmonary burden (IPB), there is an apparent age relationship of survival with IPB. For similar IPBs, the aged dogs died sooner than the young adults, who appeared to die sooner than the immature dogs. If the same survival data are reexpressed in terms of kBq IPB, the aged dogs still appear to die sooner than the young adults, but the immature dogs no longer appear to be different from the latter group. Since there were no differences in the retention of ^{239}Pu in the lungs of the immature and young adult dogs, it would appear that the growth of the lungs of the immature dogs subsequent to their exposure (which amounted to a two- to threefold increase in mass) played an important role in mitigating the inflammatory response in these dogs. Hence, local dose may be more important than average dose relative to the production of lethal inflammatory lung disease, and incidence would be directly related to the uniformity of the alpha radiation of the lung.

In summary, there are biological effects of ^{239}Pu on the lung that appear to be related to the age of the animal at exposure. Sensitivity to development of radiation pneumonitis and fibrosis from an inhaled alpha emitter appears to be inversely related to age at exposure. The role of age in the development of lung cancer cannot yet be evaluated, but is under continuing study.

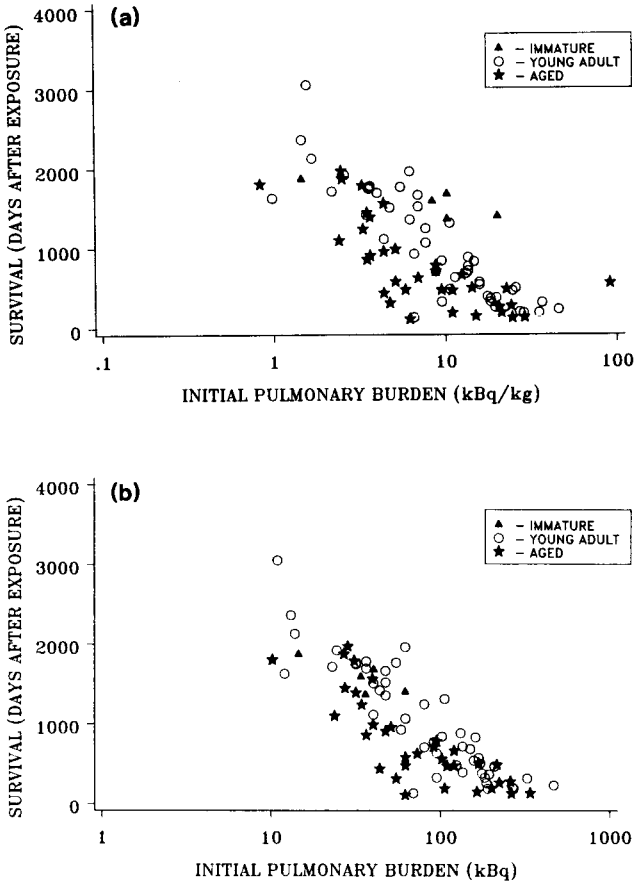

Figure 1. Dose-survival relationship for immature, young adult and aged dogs that died of radiation pneumonitis and/or pulmonary fibrosis. Initial pulmonary burden expressed as kBq/kg body mass (a) or kBq (b).

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HEALTH HAZARDS FROM RADIOCAESIUM FOLLOWING THE CHERNOBYL ACCIDENT

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The WHO Regional Office for Europe has organized a series of meetings to assess the health impact of the Chernobyl nuclear accident. The most recent meeting in the Federal Republic of Germany reviewed the principal long-lived radionuclides emitted from the accident and concluded that ^{134}Cs and ^{137}Cs had the greatest potential for contributing to the human dose because they are still present, the dose will be delivered over a long term, and because of the accumulation in some edible plants and animal products. The observed contribution of radionuclides to the collective effective dose-equivalent in the first year is about 60-80% from ingestion, 30-40% from external irradiation, and 2-20% from inhalation.

ENVIRONMENTAL PATHWAYS

Ingestion of contaminated food and water can occur by direct and rapid, or indirect and long-term pathways. The most immediate are the drinking of contaminated milk and the ingestion of fresh, leafy vegetables and fruits contaminated by direct deposition. Other pathways involve the movement of radionuclides from soil to plants and on to animals.

The amount of direct contamination on plants depends on how the deposition occurs, the density and type of crop, and the weathering that occurs before the crop is harvested. Initial retention is typically 25%, but may vary between 5 and 40% depending on vegetation density and rainfall rate. Approximately half of the deposit may be lost in 2 weeks due to weathering and growth [1]. Uptake of radiocaesium by plants from soil depends on the physicochemical conditions of the soil and the type of plants. For example, soils with high clay content retain radiocaesium much better than certain acid, sandy and organic rich soils, and plant uptake from clays will be very low. The radiocaesium uptake from acid soils may be significantly higher, and contamination of plants grown in these soils will continue for long periods of time [2].

Transfer from plants to food products of animal origin, such as milk and meat, depends on how each radionuclide is metabolized by individual species. Radiocaesium is easily transferred from fodder to milk and meat. For humans, approximately 20-30% of the daily intake is secreted in human milk [3]. Of particular interest is the lichen-reindeer (caribou)-human pathway. The effective half-life of caesium in lichen is 8-10 years, and approximately 25% of the caesium contained in lichen is absorbed by reindeer [3]. The biological half-life (T_b) of radiocaesium in animals depends on the body mass, and on whether the animal is ruminant or monogastric, domestic or wild.

Ingestion of radionuclides through the aquatic food chain also contributes to the dose, although the impact is often more limited

than through the terrestrial pathways. Radiocaesium is soluble in water. The uptake in aquatic life is influenced by factors, such as dilution, feeding habits and water salinity. Marine fish show steady-state concentration factors of the order of 100 [3]. Freshwater fish may have concentrations which are greater than those of marine fish by one order of magnitude or more, although large differences occur between species [4,5].

EXTERNAL AND INTERNAL EXPOSURES

The external radiation dose received by individuals in urban and rural areas will vary markedly for a given amount of deposition, as the external dose from deposited radionuclides is influenced by the type and amount of deposited material, and the extent and time of exposure. In rural areas, compared to urban areas, a larger fraction of the airborne particles near the surface will be deposited as a consequence of differences in surface characteristics, but the dose rate will decline progressively due to the gradual penetration of caesium into the soil. In urban areas, deposited material will be washed away and weathered more rapidly; also, the buildings will provide a significant shielding. Moreover, the population in urban areas spends more time indoors than one in rural areas. Overall, therefore, the urban external dose rate will be considerably less than the rural one.

Seasonal variations strongly influence the level of radioactivity in different foods, such as cereal crops, fodder, and pasture grass. The state of grass growth and availability of pasture at the time of the accident exerted a strong influence on the contamination observed in some animal products.

Once radiocaesium has entered the food chain, dietary habits of different populations will exert a great influence on exposure of individuals to internal radiation because of food quantities eaten and selection of dietary items. Food processing may alter the degree of radioactive contamination. For example, it decreases during the preparation of fruit juices or bread, or increases due to drying or concentration of milk.

DOSE FACTORS

Whatever the entry route into the body may be, uptake of caesium to blood is generally quick and almost complete. Once absorbed, caesium distribution is diffuse and relatively homogeneous, and its concentration is approximately the same in all the internal organs and soft tissues. Consequently, the dose factors are virtually independent of the organ or tissue considered. When considering age, two parameters count: mass of organs and tissues, and retention. In the case of radiocaesium, the effect on the dose factors of their variation with age are opposite, and compensate each other. Indeed, going from the infant to the adult, dose factors should decrease because the mass of organs and tissues increases. However, this trend is counteracted as, at the same time, the long-term retention increases with age. The retention fraction and biological half-life (T_b) of the long-term

component of retention vary with age. An infant will retain a fraction of 0.4 (T_b = 20 d), whereas an adult will retain a fraction of 0.9 (T_b = 110 d). Consequently, dose factors are virtually independent of age, although the use of an age-independent dose factor may result in a small overestimate of the effect on children.

HEALTH RISKS

The possible health effects from radiocaesium after the accident may be expressed in terms of stochastic consequences. These are fatal cancers, serious genetically related ill health and possible teratogenic effects. They are added to the incidence of these ill effects which is normally present in the population. The calculations, based on a linear no-threshold model, will usually represent an upper or conservative limit of risk expectations. The radiation doses to individuals and population for the first year after the accident, as well as the committed doses projected for 50 years, have been considered. For fatal cancers induced by radiation, a risk factor of $2 \times 10^{-2} Sv^{-1}$ is used. For serious genetic health effects, the appropriate risk factor is $4 \times 10^{-3} Sv^{-1}$ for the first two generations. Teratogenic effects may be seen following high radiation doses, and present knowledge does not exclude the possibility of a threshold, especially for low doses, in the range of 1 mSv. At higher doses the risk of severe mental retardation is 0.4 Sv^{-1} for a fetus exposed in the period between the 8th and 15th weeks of gestation, and 0.1 Sv^{-1} for a fetus exposed between the 16th and 25th weeks.

The range of individual 50-year dose increments from the accident is estimated at 0-6 mSv. The dose estimates are based on the measurements and assessments made within individual countries from methods they have considered most suitable. No attempt has been made to "normalize" such estimates. Standardization of dietary habits or other patterns of behaviour that will vary widely even within countries, is not appropriate. However, the overall uncertainty will be of the order of no more than a factor of 2.

Concerning the health effects from all radionuclides contributing to population dose, the 50-year collective dose is about 0.33×10^6 person-Sv. Thus, the upper estimate of associated fatal cancer cases in Europe (excluding the USSR) is about $(0.02)\ (0.33 \times 10^6) = 7000$, which would be added to a normal expectation of about 110 million fatal cancers (assuming 20% is the fraction of overall spontaneous mortality due to cancer which is applied to the cohort of 550 million Europeans, excluding the USSR). This is equivalent to an additional incidence of up to about 0.006%, which is an added risk to the population of about 0.001% $(7000/(550 \times 10^6))$. The first-year dose is considered for both genetic and teratogenic health effects. Assuming 28×10^6 cases as a nominal incidence of serious genetic effects, the additional radiation-induced cases in the first generation are estimated at up to 700 cases (an extra 0.003%). Teratogenic effects, such as severe mental retardation, have been seen in children who were irradiated during their 8th to 25th weeks of fetal development. The expected number of live births in the year following the accident in the

550 million people in Europe is 12 000 per million, or a total of 6.6 million. The average individual dose in the first year is 0.3 mSv, and this is combined with risk factors of 0.4 for 8 weeks plus 0.1 for 10 weeks. Up to an additional 200 cases might therefore be added to the 50 000 cases which would be expected spontaneously - (extra 0.4%) or an additional risk of 0.003% in the newborn population. The up to some 7000 anticipated radiation-induced cancer fatalities will not be evenly distributed throughout Europe (excluding the USSR). The risk of such fatalities will be higher in areas where deposition was higher or where received doses were greater. Because of differences in population density in these areas, radiation-induced cancer fatalities may be a higher proportion of the normal incidence than is given for Europe as a whole. Clearly, the reverse also is true. However, the estimated range of radiation-induced increments of possible stochastic health effects risk is very low and probably could not be detected, even by the most careful study.

ACKNOWLEDGEMENT

The final, complete report was issued as a document in Environmental Health 24 by the WHO Regional Office for Europe, Copenhagen, 1987. This paper is based on that report. The views expressed are those of the author and the WHO Group of Experts and do not necessarily represent the decision or stated policy of WHO.

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IMPLICATIONS FROM THE CHERNOBYL-ACCIDENT FOR OFF-SITE RESPONSE TO TRANSBOUNDARY CONTAMINATION

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ABSTRACT

The paper reviews the performance of radiation monitoring- and meteorological systems, communication problems, and the suitability of countermeasures taken. Socio-economic consequences and legal aspects are discussed, together with actions needed in order to improve crisis management in a similar accident in the future.

THE NEED FOR PREPAREDNESS

Since the nuclear reactor accident at Chernobyl, USSR, in April 1986, it is evident that a certain level of preparedness is required also at off-site distances hitherto considered too remote to be affected. For instance, the Austrian Province of Salzburg (population: 452 000) located about 1300 km from Chernobyl, is among the most heavily contaminated areas of Western Europe due to wash-out effects during the passage of the plume (WH86).

By 1990 the 382 nuclear power plants presently operating world-wide will probably be increased by another 118 new plants . Therefore the probability of a major accident will increase due to the combined effect of aging old plants, larger total number of plants and the lack of international standards for operator training and quality control for hardware components. The following results summarize the experience in the Province of Salzburg, Austria and involvement with international agencies in order to identify potential areas of improvement with regard to cost-effective crisis management (St88).

PERFORMANCE OF RADIATION MONITORING- AND METEOROLOGICAL SYSTEMS

The nation-wide installed radiation monitoring-system provided most valuable information on the temporal and local distribution of the external gamma dose rate. Differences in

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The views expressed in this paper represent the personal opinion of the authors and not necessarily those of the Provincial Government of Salzburg or the IAEA.

mounting the probes caused in some instances a spread of data and misinformation about the actual situation at ground-level. Since no institutions had previously been designated as regional radiation laboratories, different institutions had to provide ad hoc-analytical capacity for the large number of samples to be measured. Due to the lack of official guidelines for sampling and insufficient centralized dissimination of results, costly duplication was unavoidable. Differences in sampling methods, hardware counting equipment and analytical methods resulted in some problems of comparability among individual results.

Complex topographical conditions, together with unusual weather phenomena added to the uncertainty of meteorological predictions of localized wash-out effects. Consequently, large gradients of nuclide surface deposition over short distances, typical for the present situation, were not foreseen.

Action needed: calibration of automatic radiation monitoring network to assure description of the situation at ground level; establishment of a national inventory of radiation monitoring systems (hard- and software); regular national radiation monitoring intercomparison exercises; definition of graded sampling plans for different scenarios of environmental radioactive contamination (e.g. for different seasons, urban vs. agricultural vs. forest areas, mountainous regions vs. planes); improved meteorological models predicting rain-out effects as well as the trajectory of dispersing contamination. The International Atomic Energy Agency (IAEA) and the World Meterological Organization are conducting a joint program on atmospheric transport modelling to improve national predictions of transboundary releases.

COMMUNICATION PROBLEMS

At the regional level communication worked most efficiently in the form of a "coordination committee", comprised of local public health authorities, radiation protection experts and representatives of farmers and food industries. Close collaboration with the local media, the installation of a 24 hour-telephone service and numerous public lectures provided the basic information to the public.

At the national level communication functioned best through personal contacts between individual scientists. Official inter-office information exchange was subject to sometimes serious deficiencies due to overburdening of the system (e.g. insufficient number of telephone lines, unidentifiable individual responsibilities, significant time-delays in urgently needed decisions). Since the existing national "Radiation Protection Commission" was not called into session, occassionally interpretation of national regulatory directives was varied from province to province, e.g. concerning the use of fresh grass as animal fodder or the method of waste-disposal of contaminated whey.

At the international level special trade connections with

neighbouring West Germany were reflected in general decisions, e.g. Austria also excluded East Germany from the import restrictions concerning food from all other COMECON-countries but included Jugoslavia although it is not a COMECON-member; or: food limits for cheese and beef were adjusted to EC-levels to faciliate trading. While detailed information on quantity and nature of the release was ultimately communicated to affected States, the initial delay reduced confidence in the reliability of the data made available. The lack of prompt and credible information resulted in the implementation of some counter-measures which may not have been considered necessary in the face of the actual levels of contamination.

Action needed: installation of a national system for retrieval and dissimination of radiation data with telex/telefax-capability; design of institutional mechanisms capable of making policy decisions on national radiation protection measures; internationally, adherence to and compliance with the convention on Eary Notification of a Nuclear Accident in co-ordination with the IAEA.

SUITABILITY OF COUNTERMEASURES

The information about the arrival of the plume was transmitted rapidly to the public via the broadcasting network. During the following ten hours only small amounts of dry deposition occurred. The official advice given at that time emphasized reduced contact with the ground, and limitations on the consumption of certain food stuffs grown outdoors. The resulting economic damages could have been minimized by providing group-specific advice on countermeasures, such as: intensified harvest of food stuff or animal fodder; covering of unripe vegetation or food stored outdoors with protective foils; closing of all openings of green-houses and stables; prohibition of the use of contaminated rain water or compost; advice to wear protective masks during dust-creating work.

Action needed: collection of international experiences concerning practically applicable profilactic measures to minimize impact of fallout-contamination for different population groups (consumers, farmers, trade, industry); development of agreed levels of intervention; implementation of the Convention on Assistance in the Event of a Nuclear Accident or Radiological Emergency, including the development of a data bank on experts, materials, and equipment available in the event of an emergency.

SOCIO-ECONOMIC CONSEQUENCES

The fallout contamination affected most of all food producers (farmers, industries) and consumers. The milk industry suffered the heaviest losses (54% of the total direct damages), followed by stock farming (16%) and vegetable farming (5%). Based on the Austrian upper limits of radionuclide concentration in food the total costs due to damages amounted in Salzburg to about USS 13/person, resulting in a collective dose reduction of at least 50% had no action been taken. These costs were due to

the need for additional uncontaminated animal fodder for cattle and sheep, the destruction of contaminated vegetables and fruit, the blending of milk with varying degrees of contamination, the changing of the production ratio of milk, butter, and cheese, the disposal of contaminated whey, the use of contaminated game, pork and beef as powdered animal fodder and the cost of food additives to lower cesium-uptake by cattle. The supply of suitable raw materials was particularly difficult for producers of bottled food for babies, which had to rely to a large extent on imports.

The consumers reacted with a change of their dietary characteristics, e.g. less demand for milk products, sheep and game, but increased demand for food products from knowingly uncontaminated areas, such as Spain, Israel and Africa.

Action needed: legislative measures to provide for compensation of losses to food producers complying with national regulatory standards; provision for emergency fodder storage; development of techniques for waste disposal or re-cycling of contaminated agricultural products; increase of analytical capacity of existing laboratories (e.g. automatic sample changers).

LEGAL ASPECTS

Immediately following the accident, review of the question of compensation for damages revealed the lack of any multilateral agreement to which the Soviet Union was party which would provide a mechanism for compensating non-nationals for deleterious transboundary effects of a radiological release. While the Paris Convention on Third Party Liability in the Field of Nuclear Energy and the Vienna Convention on Civil Liability for Nuclear Damage both address this issue, neither are comprehensive, insofar as they deal only with civil liability and not State responsibility and limit the types of damage for which recovery is available, and neither are widely adhered to (14 parties to the geographically-limited Paris Convention; 10 parties to the Vienna Convention).

Action needed: harmonization of the existing compensation conventions; increased adherence to bilateral and multilateral agreements covering both civil liability and State responsibility.

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EFFECTIVE DOSE EQUIVALENT TO BREAST FED INFANTS DUE TO ARTIFICIAL AND NATURAL RADIOACTIVITY

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1. Introduction

Since many years health organisations have paid attention to the contamination of human milk caused by the exposure of pregnant women to toxic substances. In particular, several chemical pollutants, which undergo very slow metabolisation processes in the human body, can be accumulated in wet tissues and released to the milk. It has been observed (1) that in the case of some chemical substances (like DDT or PCB) the levels in human milk are higher than those in cow's milk. It is now well known that human milk is a biological gauge of environmental pollution. However, in the case of radioactive contaminations more experimental work and theoretical interpretation of data are yet needed. Generally speaking, available information seems to indicate that relative levels of some radionuclides in human milk may be different from those present in the maternal diet.

After the Chernobyl accident, the Italian National Institute of Health (ISS), with the collaboration of the Epidemiological Unit of Latium District, started a research devoted to the study of the radioactive contamination of human milk in Latium. The motivation is three-fold, namely: i) to measure the contamination of a food product utilised for infants in order to assess the corresponding dose to breast-fed infants; ii) to clarify any possible correlations between radioactive levels in human milk and those in the total body; iii) to correlate the levels in human milk with those in the maternal diet.

2. Main features of the collected samples

Samples of human milk have been collected from one Milk Bank (which gathers milk from more than ten clinics and hospitals in Rome) and from the two greatest Universitary Polyclinics in Rome. It should be pointed out that such health units receive patients from the whole Latium District. Each sample is a mixing of milk from 5-10 women, collected within the first week after the child birth.

In order to check if the collected samples are representative of the situation in the District, a comparison among the distributions of some parameters relative to milk donors and to the whole class of pregnant women in Latium has been carried out. Indeed several factors (like the age of the mother, the weight of the child, the parity, the job of the mother and her food habits) may affect the representativeness of the samples. Such a comparison (see ref. 2) has clearly shown that the samples are really representative of the present situation in the Latium District.

3. Radioactivity in human milk samples

Gamma activities of human milk samples of about 450 ml have been measured with two coaxial intrinsic HpGe detectors (efficiency 38.2% and 26.6%, resolution 1.95 keV and 1.73 keV, respectively). The values of I-131, Cs-134 and Cs-137 concentrations have been published in refs. 2 and 3. The I-131 concentration has never exceeded 4 Bq/l and the Cs-134 + Cs-137 concentration has always been lower than 9 Bq/l. The corresponding effective dose equivalents have been evaluated for breast-fed infants born in three different periods (namely, at the end of April '86, October '86 and March '87), assuming the NRPB (4) conversion factors and the conservative hypoteses of 0.9 l/d milk ingestion and of a six-month breast feeding period. The calculated doses turned out to be lower than 15 μSv (3). It must be stressed that the highest doses were received in the first period due to the I-131 contribution, even if the ban of consumption of leafy vegetables for the whole population and that of cow's milk for pregnant women and children under ten lowered the intake of I-131 during May '86.

In this paper attention is paid mainly to natural radioactivity and to a comparison between cesium and potassium concentrations in human milk. In Fig. 1 the monthly averages of K-40 concentration found in the samples are shown and compared with the corresponding Cs-137 mean concentrations. The uncertainties, represented by one standard deviation in Fig. 1, depend on the number of samples measured each month and are higher for potassium concentration due to background subtraction. The mean value of K-40 concentration in the period May '86 - November '87 was (19.9 ± 2.6) Bq/l, whereas the corresponding value observed (6) in cow's milk from the Central Dairy of Rome was (57 ± 10) Bq/l.

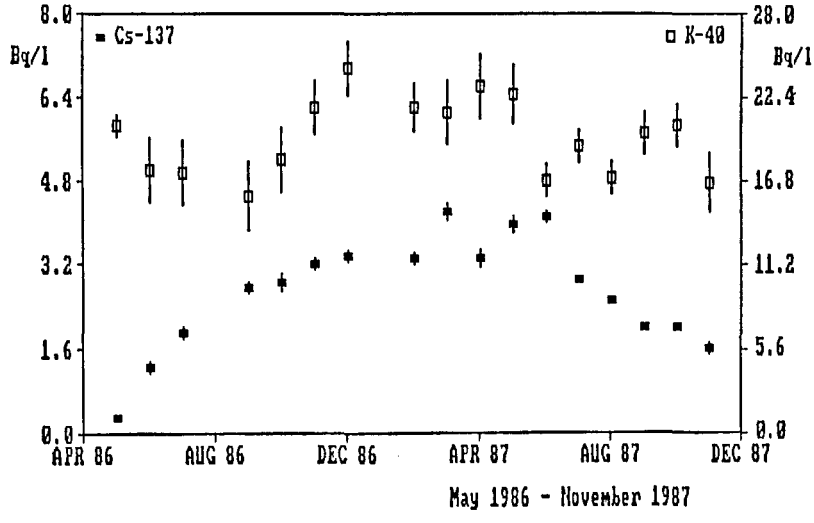

Fig. 1. Monthly averages of Cs-137 and K-40 concentration measured in human milk samples.

In Fig. 2 the monthly averages of Cs-137 concentration in human milk and cow's milk (from the Central Diary of Rome) are

compared. It can be seen that the Cs-137 level was much higher in cow's milk than in human milk during the whole '86 and the first half of '87.

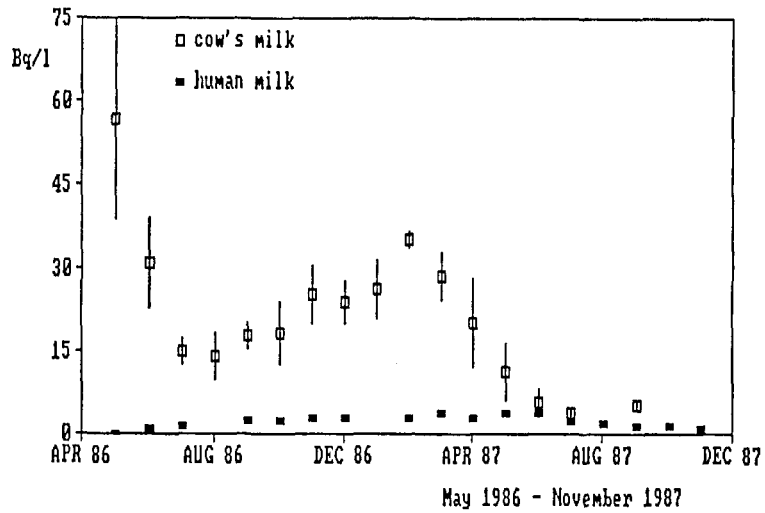

Fig. 2. Monthly averages of Cs-137 concentration in human milk and in cow's milk (6) from the Central Diary of Rome (errors are one standard deviation and represent the spread of the data within each month).

Total potassium concentration has been measured in some human milk samples and - for comparison - in some cow's and sheep's milk ones (see ref. 5 for the radioactivity measurements). The potassium mean content was found to be 0.75 g/l, 1.47 g/l and 1.20 g/l for human, cow's and sheep's milk respectively. The value for human milk is higher than some values found in literature (see ref. 7), but it is well known that it is highly dependent on mother's diet and other factors. The ratio between the K-40 and total K concentrations turns out to be in the range (28 - 36) $Bq \cdot l^{-1}/g \cdot l^{-1}$, in good agreement with the isotopic ratio of natural potassium (0.0118 %). This fact makes it possible to use directely the K-40 concentration measured in the samples instead of the total potassium concentration in evaluating the Cs-137 to potassium ratio. In Fig. 3 the monthly averages of the Cs-137/K-40 ratio observed in human milk samples are shown. It can be seen that the ratio is always lower than 25%, whereas the corresponding ratio in cow's milk (6) is higher than 30% up to May '87.

Unfortunately, up to now available experimental data cannot allow any meaningful assessment of the Cs-137/K-40 ratio in the total diet of pregnant women in Latium. However it is interesting to note that the Cs-137/K-40 ratio measured (5) in wheat samples from Latium was found to be higher than 30%.

The results presented clearly shows that the Cs-137 contamination in human milk has been very low both in an absolute sense and in comparison with cow's milk. Furthermore, the Cs-137/K-40 ratio has been lower than the corresponding ratio in cow's milk

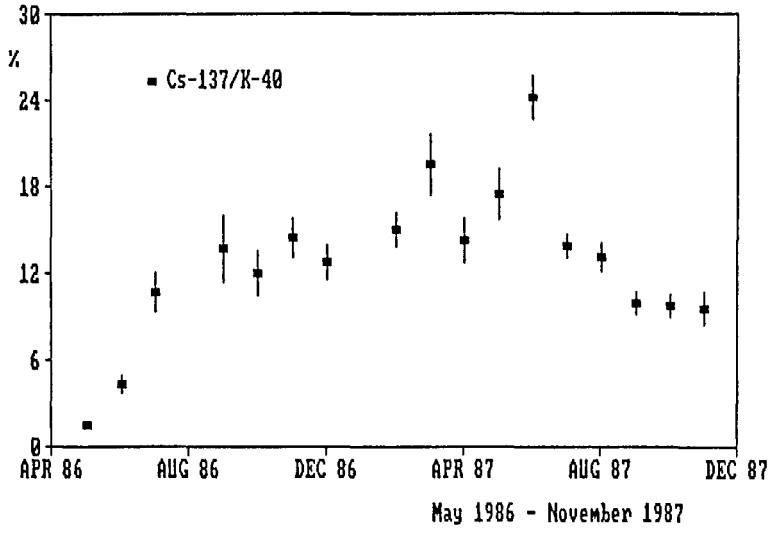

Fig. 3. Values of the Cs-137/K-40 ratio found in human milk.

and, probably, in mother's diet, unlike the case found by other authors in the past (8).

ACKNOWLEDGEMENTS. The authors are very grateful to Dr. S. Caroli, Dr E. Beccaloni and Dr. E.Coni - Lab. Tossicologia Applicata ISS - for the determination of total potassium concentrations.

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PRELIMINARY DOSE ASSESSMENT IN TURKEY AFTER THE CHERNOBYL ACCIDENT

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ABSTRACT

A major reactor accident occurred at unit 4 of the Chernobyl Nuclear Power Station located about 60 miles north of Kiev in the U.S.S.R. on April 26, 1986 at 1:23 a.m. Turkey's coasts on the Black Sea are about 1200 miles south of Kiev. The radioactive cloud reached northern and western Turkey about 48 and 72 hours after the accident, respectively. Most of the activity in the clouds reached the ground with localized rain; therefore the ground contamination showed great variation. For example, at several different locations, milk samples taken between May 5-9, 1986 showed average activity of 399 Bq/l, and a standard deviation of ± 222 Bq/l. Cs-137 activity at one of these locations measured 108 Bq/l on May 5, 1986. The air sample activity on the same location was 0.4 Bq/lt. The doses to individuals calculated from these figures are insignificant. For example, thyroid dose of an individual from one liter of ingested milk is about 0.19 mSv. Inhalation dose to an individual is about 0.0001 mSv/l air inhaled. Short-term and long-term doses from the cloud itself, from inhalation and the deposited radioactivity on food, will be presented.

The fallout information was not released to the public by the authorities, which created a great deal of confusion and anxiety in the population.

WEATHERING AND MIGRATION OF CHERNOBYL FALLOUT IN SWEDEN

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ABSTRACT

In situ spectrometer measurements of the Chernobyl fallout in Sweden have been made in the Gävle and Studsvik areas. The main objective of the measurements is to study the weathering effects on typical urban surfaces and migration on porous surfaces.

Measurements were made in May, July and September. The average remaining fraction after 5 months is around 0.6 (excluding decay). No significant differences between typical urban surfaces were found, except for paving stones, which showed a remaining fraction of 0.8.

The soil profiles have a half value depth of a few centimeters. Activity was found below 20 cm, which indcates other transport mechanisms than pure diffusion.

A similar behavior of all nuclides with respect to both weathering and migration, indicates that the main part of the activity is bound to carrier particles.

SWEDISH RADIATION PROTECTION RESEARCH FOLLOWING CHERNOBYL: DISPERSION, DEPOSITION, MIGRATION, UPTAKE, INTAKE, DOSE

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ABSTRACT

Immediately after the Chernobyl accident, a massive research effort in the radiation protection field was started in Sweden. Most projects were initiated by the National Institute of Radiation Protection (NIRP) and were carried out by a number of different Swedish research groups, including NIRP itself.

The investigations cover fields such as: plume transport over Sweden, deposition of radioactive particles, transport of the activity in aquatic and terrestrial systems, food chain pathways including uptake in vegetables and intake of activity in animals (and some efforts to reduce this intake), intake in man and dose calculations for Swedish individuals and populations.

A number of sociological studies of the reactions of the public to the accident have also been initiated.

This paper summarizes and evaluates this national research effort.

RADIONUCLIDE CONCENTRATION FROM PEAT BURNING AFTER THE CHERNOBYL ACCIDENT

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ABSTRACT

We have studied the radionuclide concentrations in byproducts and releases from a 30 MW peat-burning power plant in central Sweden. The plant is located in an area that received high levels of radioactive fall-out from the Chernobyl accident.

After the accident at Chernobyl, the plant carried out a test run before the beginning of the normal running season. Samples of peat and ash were collected during a 2 month period and were studied in order to ascertain whether radiation protection was necessary for workers handling the peat and byproducts.

In spite of the high ground contamination of radionuclides (20-80 kBq/m^2) of the peat, the radionuclide concentration in the peat was only about 1 kBq/kg (and half of this one year later). This is due to the process in which the top 50 cm layer of peat is continuosly mixed and turned over.

Samples of fly ash from different parts of the plant, analysed using gamma-ray spectroscopy, were found to have activity concentrations of 10-50 kBq/kg Cs-137, while the activity concentration of bottom ash was 4-10 kBq/kg. During the winter of 1984-85 the average level of Cs-137 in the flyash was 340 Bq/kg.

Condensed water from the chimney did not contain any measurable amounts of Cs-137. Emission measurements of the gases in the chimney gave rather high activity concentrations of Cs-137. The maximum value of 70 kBq/kg was probably due to the ease with which caesium escapes during heating.

No special radiation protection steps were found to be necessary.

INTRODUCTION

After Chernobyl Cs-137 was one of the major contributors to external radiation in areas with high fall-out. Peat in these areas will therefore also contain Cs-137 in large amounts.

Gamma-spectrometric analyses of radioactivity concentrations in peat and different ash products have therefore been determined in a district heating plant in Sandviken in the central part of Sweden.

During this period 15000 tonnes of peat were used, producing 470 tonnes of flyash and 70 tonnes of bottom ash. Only about 1-3 tonnes of flyash passed the filters.

SAMPLING

During a two-month period, from the end of November 1986 till the end of January 1987, grab samples were taken at the plant and were sent to us where they were homogenized and dried and packed into 123 ml airtight tubs.

Grab samples were taken at 5 stages, as shown in figure 1. Peat (1), flyash from after the cyclone (2), particle size 10-50 µm, after the textile filter (3), less than 10 µm and a mixture of total flyash immediately before the waste container (5), and bottom ash (4), consisting of mainly sand, were collected at least once a week during the period of normal running.

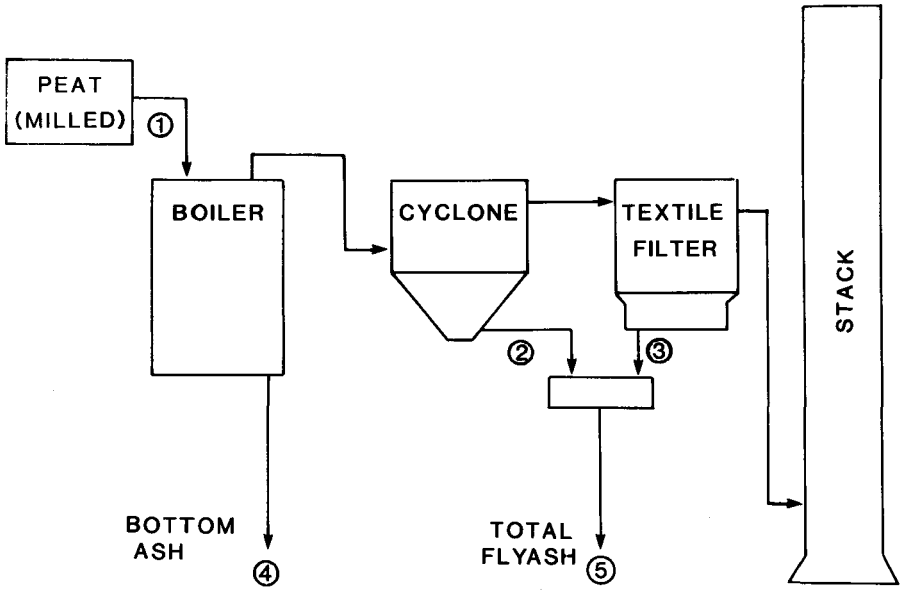

Fig 1. Schematic diagram of the Sandviken peat plant. The numbers 1-5 show at which stages samples were taken.

DETECTOR

We have used a germanium lithium detector for the determination of Cs-137. The detection efficiency was obtained using a reference solution in the same geometry and of the same density as the samples to be investigated.

RESULTS

Cs-137 concentration of fall-out was between 20 and 80 kBq/m^2 in the area with several peat mosses used at the plant (1).

In spite of the high ground contamination of mainly caesium but also other radionuclides, of the peat moss, the radionuclide concentration in the used peat was fairly low because of the process in which the top 50 cm layer of peat is continuously mixed and turned over with less contaminated peat. One year after this sampling the activity concentration was only half of that in table 1.

Table 1. ^{137}Cs concentrations in peat and ash products from Sandviken.

Type of sample	Arithm ave s.d.	Median
	Bq/kg dry weight	
Peat (16)	1150+/-155	1154
Bottom ash (6)	6031+/-1691	5354
Flyash 10-50 µm (16)	9565+/-2195	9537
Total flyash (8)	16792+/-3560	17270
Flyash <10 µm (13)	29934+/-14628	28830

The number of samples is shown in parentheses.

It is evident that the highest activity concentrations are found on the smallest particles. Emission measurements of the gases in the chimney gave values of about 70 kBq/kg, with great uncertainty because of small and only 2 samples.

If we try to check the balance of in- and outcoming particulate matter for Cs-137 we will find that only about 15% is missing. This is probably due to the fact that Cs-137 is volatile.

During the winter of 1984/85 the average level of Cs-137 in the flyash was 340 Bq/kg (2).

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COLLECTIVE DOSE COMMITMENT IN SWEDEN AFTER THE CHERNOBYL ACCIDENT
CALCULATION FOR INHALATION AN EXTERNAL IRRADIATION

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ABSTRACT

The collective committed dose to the Swedish population from the Chernobyl accident has been estimated from a large number of measurements of radionuclides in air and on the ground. The calculation is, besides a number of teoretical assumptions, based on three types of measurements carried out during the first year after the accident.

1. Particulate air activity from six high volume air filter sampling stations.

2. In situ high resolution gamma spectrometry at selected sites representing the different fallout compositions.

3. Gamma spectrometry of total Sweden by aircraft carried sodium-iodide spectrometer.

The collective committed dose to the Swedish population has been estimated between 4000 and 6000 manSv, foodstuffs excluded. Around 20 % of the collective dose is received during the first year. The contribution to the collective dose commitment from inhalation and external cloud radiation is small.

The collective dose commitment is dominated by the ground deposited cesium-134 and cesium-137 isotopes. In wet deposition areas up to 80 % of the dose equivalent during the first year is due to cesium isotopes. In dry deposition areas the relative composition of radionuclides is different from the wet deposition areas and the cesium contribution is lower.

The average dose commitment to the Swedish population is around 0.1 - 0.5 mSv during the first year and 0.7 - 4 mSv during the first 50 years.

The dose commitment from Chernobyl can be compared with the dose commitment from ground deposited cesium-fallout from nuclear atmospheric tests during the next 50-year period. The fallout dose is 0.26 mSv, which corresponds to a collective dose of 2170 manSv in Sweden.

RESULTS OF A HONEY MONITORING PROGRAM IN ITALY AFTER THE CHERNOBYL ACCIDENT

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INTRODUCTION

Honey bees (Apis Mellifera) as bioindicators of environmental pollution have been studied for a long time to analyse their ability to detect air and water pollution (1). Bees forage over a large area (around 7 km^2) and seem to act as biomagnifiers of the contamination of the environment.

Honey bees and their products, mainly honey, were used in the U.S.A. during the seventies to monitor potential releases from nuclear waste disposal sites. Honey samples were examined to detect contaminated areas surrounding a nuclear facility at Cattaraugus Country (NY) (2); the presence of radionuclides from fallout of nuclear weapon tests in honey samples was also investigated (3).

As a consequence of the Chernobyl accident, the radioactive contamination spread all over the Italian soil allowed the analysis of the radioactivity transfer from the environment to honey.

MATERIALS AND METHODS

Soon after the Chernobyl accident few hundreds of honey samples from various parts of Italy were collected.

The samples originated from two administrative districts (Regioni) located in the North-West (Piemonte) and in the North-East of Italy (Friuli-Venezia Giulia) were singled out. These two districts were heavily affected by the radioactive contamination.

Mainly two different types of honey were sampled: honey from flowers (Multiflora) and from Robinia pseudoacacia.

The samples, weighing around 500-1000 grams, were analysed, without any pretreatment, by gamma spectrometry using HPGe detectors. An intercalibration was carried out among the laboratories performing the measurements.

To study the contamination trend, honey samples of the same type were also collected in the same districts during 1987.

EXPERIMENTAL RESULTS AND ANALYSIS

The concentration of Cs-137 detected in the honey samples ranges from 3.4 Bq/kg to 606.8 Bq/kg for the Piemonte and Friuli Venezia Giulia (FVG) districts in 1986. First of all an attempt was made to correlate the Cs-137 honey contamination with Cs-137 soil deposition averaged over an area comparable with that used by the bees to forage.

In the case of FVG district detailed deposition data were available (4), enabling us to perform a correlation analysis. The results show that correlation exists (Pearson product-moment correlation = 0.85) between Multiflora honey contamination and soil deposition averaged over circular areas having radius $\leq$ 8 km. Conversely, if samples of all different types of honey were considered, a very low correlation was found. The results of measurements performed on all the samples collected in FVG during 1986 and 1987 are shown in TABLE 1. The district was divided into three geographycal areas (A = mountain, B = hill, C = open plain). For each area the average value of the Cs-137 honey contamination is indicated as well as the standard deviation and the number of samples. For each area the range of 1986 soil deposition is also shown.

For the Piemonte district the soil deposition data did not allow us to carry out a detailed analysis as for FVG. However a rough calculation of the ratio (Bq/kg) honey contamination and (Bq/m^2) soil deposition shows for Multiflora honey a value of 10^{-2} which is in agreeement with that evaluated from the FVG data. The experimental results are shown in TABLE 2. Only two areas have been considered in this case (B = hill, C = open plain).

In a previous study (5) it was noted that in Piemonte and in FVG the Cs-137 contamination detected in the Multiflora honey collected in 1986 was significantly higher than that of Robinia honey. This could be due to two reasons: 1) the activity intercepted by Multiflora is sensibly higher than that intercepted by Robinia and 2) the Multiflora blossom was at maximum at the end of April-beginning of May, while Robinia blossomed around half of May when the concentration of Cs-137 in air was rather low.

TABLE 1
FRIULI VENEZIA GIULIA

AREA	Cs-137 SOIL 1986 (kBq/m^2)		Cs-137 HONEY CONTAMINATION (Bq/kg) 1986						1987					
			MULTIFLORA			ROBINIA			MULTIFLORA			ROBINIA		
	MIN	MAX	n	$\bar{x}$	s	n	$\bar{x}$	s	n	$\bar{x}$	s	n	$\bar{x}$	s
A	18.3	27.7	10	153.3	88.1	1	67.0		4	17.6	6.6			
B	4.5	20.5	15	149.3	139.2	11	39.4	35.8	3	16.7	12.4	4	3.2	1.0
C	1.2	14.2	21	63.8	45.6	9	29.9	14.6	2	5.7	.7	4	5.9	5.2

TABLE 2
PIEMONTE

AREA	Cs-137 SOIL 1986 (kBq/m^2)		Cs-137 HONEY CONTAMINATION (Bq/kg) 1986						1987					
			MULTIFLORA			ROBINIA			MULTIFLORA			ROBINIA		
	MIN	MAX	n	$\bar{x}$	s	n	$\bar{x}$	s	n	$\bar{x}$	s	n	$\bar{x}$	s
B	5.5	18.2	3	128.5	53.8	13	13.6	7.9	6	5.7	2.4	13	1.5	.7
C	8.8	9.4	4	109.0	54.6	5	6.8	2.2	2	1.4	.1	5	1.2	.3

CONCLUSIONS

The analysis of data collected in 1987 shows that the honey contamination is around one order of magnitude lower than in 1986. While in 1986 the honey contamination can be attributed mainly to direct deposition, in 1987 there are several factors which might affect the plant uptake of radionuclides: direct deposition, resuspension, translocation and it is impossible to discriminate, at the moment, their relative contribution to the honey contamination.

An exception seems to be represented by some Chestnut honey samples. For the FVG district, in fact, the average value of Cs-137 contamination in the 1986 Chesnut honey is 96.3 Bq/kg (n = 6, s = 89.0) and in 1987 is 43.3 Bq/kg (n = 4, s = 6.8). For the Piemonte district only one sample was collected in 1986 showing a Cs-137 contamination value of 29.5 Bq/kg, while the average value for 1987 is 38.5 Bq/kg (n = 6, s = 20.9). Due to the scarcity of data, more analysis is required to study Chesnut honey contamination trend.

It appears also that no discrimination should take place between the transfer of Cs-137 and that of other radionuclides to honey, as the ratio Cs-137/Ru-103 in air is the same as detected in honey samples (1).

With regard to the use of honey as bioindicator, we pointed out that Multiflora honey can be useful as contamination indicator in case of elevated contamination, during flowering time, around areas of few km^2. Nevertheless it seems to us that, in case of large contamination, environmental samples exist that can be better and perhaps less expensively used as bioindicators (milk, grass, soil). However the problem is still open and more analysis is in progress.

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RADIATION MONITORING OF IMPORTED FOOD TO SAUDI ARABIA AFTER CHERNOBYL ACCIDENT

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INTRODUCTION:

Following Chernobyl reactor accident, King Abdulaziz University (KAU) was assigned the responsibility of monitoring food imports reaching the western ports of Saudi Arabia. This includes the three western seaports of Jeddah, Yanbu and Jizan and the airport of Jeddah. Through the seaport of Jeddah, the largest in Saudi Arabia,essentially all kinds of foodstuffs are entering. Chilled meat, fresh vegetables and other items that can not be stored for long time are coming through Jeddah airport, while Jizan and Yanbu handle mianly barley and animal feed. The monitoring program started in the middle of June. This is the time when pilgrimage season starts and about one million persons come from different parts of the world to the city of Mecca. Food imports drastically increases during this time and large number of live sheep and cows are imported for religious sacrifice.

MONITORING PROCEDURE:

The monitoring process includes the use of 5.08-x 5.08-cm (2-x2-in) NaI(Tℓ)scintillation detector used with counter that gives count rate only. Several samples of few kilograms each are brought from different parts of each shipment for radiation inspection. In this process the detector was either surrounded by the sample material or put on its surface to obtain better counting efficiency. In many cases and whenever possible inspection was done inside the food container where usually better counting is obtained, since more food is present and gamma photons are coming from all directions. An increase of 10 to 20% in the count rate is observed when the sample is containing few hundered Bq/Kg of total radioactivity.

If any increase in the count rate is observed over the background, larger samples were then inspected by a 12.7-x 10.16-cm (5-x4-in) scintillation detector provided with a discriminator and a counter-timer where discrimination was made only against lower level background radiation. This detector was not used to identify radionuclides in the food or for measuring their concentration, but rather to confirm the observation made by the survey meter since the counting time can be extended to any desirable value. Usually 5 minutes is long enough time for such confirmation.

Samples showing increased radioactivity above background are brought to the radiation detection laboratory for radionuclides identification and concentration measurements. The detection systems include a Ge(Li) detector, a pure Ge detector, a 1024 multichannel analyzer, a PC-IBM computer with special electronic board equivalent

to 8192 channels along with the associated electronics. The detectors are shielded by lead bricks to reduce as mush as possible the background radiation. Meat samples were cut into small pieces to fill a plastic container where their activity was compared with water (filling similar container) having 1000 Bq/Kg of each of different fission products. Milk powder radioactivity was compared with the activity of another milk powder originally free from radioactivity and then mixed with known amount of fission products radionuclides to give the same concentration as that of water. Radioactivity of other food samples, such as fruits, were compared with water, otherwise different reference sources were made for them. In the measuring system used it was possible to measure concentrations of few Bq/Kg. The Kingdom of Saudi Arabia has initially adopted the acceptable levels of imported food taken by E.E.C. countries namely 600 Bq/Kg and 370 Bq/Kg of total Cs activities for adult and dairy and children food respectively.

RESULTS AND DISCUSSION:

From the middle of June, when the monitoring program was effectively started, untill early September essentially all samples analyzed in the KAU laboratory were meat and milk powder. These food items came from cattle and sheep grazing on contaminated grass. Mainly Cs-137 and Cs-134 were observed, although in the early days other fission products were found as well. During the same period no observation of radioactive contamination of fruits and vegetables was registered. Presumably surface contamination of these items, if any, was removed by washing. In the month of September, radioactivity were found in biscuits, mushrooms and fertilizers in addition to meat and milk powder. About early October, radioactive contamination was detected in nuts, fruits, grain products, macaroni, amber and some indoor plants.

Table 1 shows the type of food analyzed by the gamma spectroscopy system, their total number, the number of those having radiation concentration above the acceptable level and their percentages during the period from the first of July 1986 untill the end of February 1987. During this period large numbers of meat and milk samples were brought for radioactivity analysis. Furthermore a large percentage of them showed levels above acceptable. Wheat and wheat products such as Macaroni, biscuits also showed noticeable amounts of contamination. Clearly the right conclusion on tendency of certain food products to accumulate radioactivity can not be easily concluded from this table because this will depend on the total amount imported of the specific type of foodstuff.

Figure 1 shows the variation of total number of samples analyzed and the number of those showing radiation above the acceptable level with time.From July to September 1986.The samples sent for analysis were decreasing in number. This may be due to the fact that radiation contamination was actually decreasing with time in meat and milk and/or possibly due to recognition by exporting countries of radiation in food and better inspection procedures being applied. After the month of October the number of samples brought for analysis started to increase due to the fact that contamination was found in other varieties of foodstuff in addition to meat and milk. Generally smaller contamination concentration was found.

Figure 2 shows the ratio of Cs-134 to Cs-137 concentration in meat after August 16. Clearly this ratio is decreasing due to the fact that the half-life of Cs-134(2.06y) is much shorter than that of Cs-137 (30y).

Table 1: Types of samples analyzed by gamma spectroscopy method, their total number and the number having radiation level above acceptable.

Sl. No.	Sample Type.	No.of samples received	Percentage out of total	Above acceptable level	
				No.of samples	Percentage
1	Meat	66	29.07	22	33.33
2	Lentils	25	11.01	4	16.00
3	Milk Powder	23	10.13	13	56.52
4	Wheat	17	7.49	1	5.88
5	Macaroni	14	6.16	5	35.71
6	Fig	9	3.96	0	0
7	Biscuit	8	3.52	3	37.5
8	'Mahlab' (*cerasus mahaleb*)	7	3.08	6	85.71
9	Semolina	6	2.64	0	0
10	Plant soil	6	2.64	4	66.67
11	Veg.soup	5	2.20	0	0
12	Dried grass (for animal)	4	1.76	4	100
13	Chocolate	4	1.76	0	0
14	Hazelnuts	4	1.76	2	50
15	Apricot	4	1.76	0	0
16	Apple	3	1.32	0	0
17	Walnut	3	1.32	0	0
18	Chestnuts	3	1.32	0	0
19	Honey	2	0.88	0	0
20	Animal feed	2	0.88	0	0
21	Ambergris	2	0.88	2	100
22	Cumin	2	0.88	0	0
23	Hazelnuts with honey	2	0.88	1	50
24	Fertilizer	2	0.88	2	100
25	Mushroom	1	0.44	1	100
26	Cheese	1	0.44	1	100
27	Almond	1	0.44	0	0
28	Orange	1	0.44	0	0

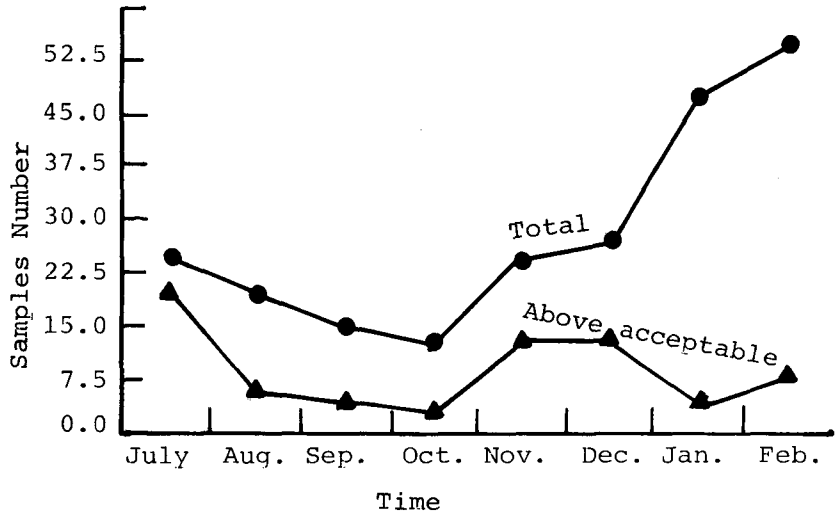

Fig.1: The variation of total number of samples analyzed and the number showing radiation level above acceptable with time starting from July 1986 until February 1987.

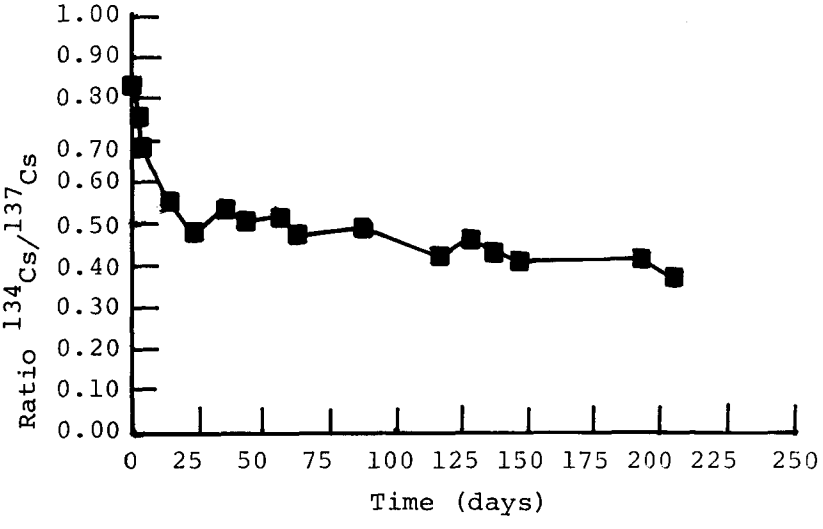

Fig.2: The variation of the ratio of ^{134}Cs to ^{137}Cs concentrations in meat with time starting August 16, 1986.

MONITORING OF RADIOACTIVITY IN IMPORTED FOODSTUFFS EXPERIENCE GAINED AND RECOMMENDATIONS

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INTRODUCTION:

The Chernobyl reactor accident released large amounts of radioactivity that was carried by winds across international boundaries. Soils and surface waters as well as vegetation and grazing animals in neighbouring countries to the Soviet Union were subject to radioactive contamination in various degrees. Saudi Arabia imports much of its foodstuffs from European countries and about 75% of the imported foodstuffs come through western ports of the country. Hence the Government of Saudi Arabia took appropriate measures to ensure safety of imported foodstuffs.

King Abdulaziz University (KAU) was directed by the government at about mid-June 1986, to carry out radioactivity inspections of food items reaching Jeddah, Yanbu and Jizan seaports as well as food arrivals at King Abdulaziz International Airport at Jeddah. Parallel responsibilities were assigned to other universities and King Abdulaziz City for Science and Technology, for radioactivity inspections of imported foodstuffs reaching ports or inlets in their respective areas. In applying the measures and controls enacted by the government against importation of radioactively contaminated foodstuffs, much management experience was acquired and some of it is rather unique to Saudi Arabia.

RADIOACTIVITY INSPECTION MANAGEMENT:

1. Partition of the monitoring process into two parts was adopted. Gross radioactivity inspection was done at the inlets by the use of scintillation survey meters and qualitative and quantitative analysis for radionuclides was done at laboratories of the nuclear engineering department of King Abdulaziz University. Food samples that showed any increase of radioactivity above background raised sufficient concern that the parent shipment might have radioactive contamination resulting from Chernobyl reactor accident. The shipment was then held at the respective inlet, pending detailed analysis of representative samples at (KAU). Should that analysis show unacceptable levels of reactor fission products, the decision was then taken by the authorities to bar that shipment from entry into Saudi Arabia and its local agent was instructed to reship it to its port of origin.

2. The inspection team applied the temporary foodstuffs radioactivity limits adopted by the government of Saudi Arabia and

were similar to those practiced by the European Economic Council. These limits were a maximum of 600 Bq/Kg or liter for adult food and 370 Bq/Kg or liter for all infant food, for total Cs-137 and Cs-134. Recently however, the limits were officially set at 75 Bq/Kg or liter of total Cs-137 and Cs-134, for adult food, 30 Bq/Kg or liter for infant food and 10 Bq/liter for water, while animal feed limit was set at 300 Bq/Kg.

3. The assignment of radiological inspection of imported foodstuffs to the different scientific institutions was on a temporary basis, pending completion of training of the technical staff at the government quality control laboratories at the different ports. Hence an appropriate program of technical training for the quality control employees, in the western ports, on radioactivity gross inspection was embarked upon by (KAU). Lectures on the relevant principles of atomic and nuclear physics, radioisotopes, and foodstuffs' contamination by radioactivity were included. Furthermore experiments and laboratory demonstrations covering the different aspects and sources of error in gross radioactivity assay in foodstuffs were also included.

4. Acquisition of various radioactivity monitoring equipment from the various companies suffered much delay after Chernobyl accident. There was much demand for these equipment and the inventories were down. It is important to be prepared for such emergencies and back-up equipment should always be available at the various inspection sites.

5. There was a need for quick yet reliable sampling technique. In some situations, the sample heterogenities were not only in single shipments, but also inside individual containers noticeable variations in radioactivity were observed. One sample might be practically free from fission radionuclides side by side with samples containing unacceptable radioactivity levels. This could be due to packaging of food items from different sources in one shipment. An efficient and not time consuming sampling technique should be devised so as to permit truely representative sampling in emergency situations such as at pilgrimmage times.

6. The quantity of imported foodstuff is greatly increased around the time of Muslim pilgrimmage to Holy Mecca. A great influx of food items, live animals (mainly sheep and cows) and of course people takes place over a short period of time. In 1986, pilgrimmage time was about the middle of August, i.e. after the time of Chernobyl accident, and food items and live animals were subjected to radioactivity inspection. The different ports to the country lacked animal whole body counters. When live animals are imported for religious sacrifice on a certain day, sampling meat from hundreds of animals becomes a problem. Inspection by survey meters is not enough and installation of animal whole body counters at all inlets to the country would facilitate the inspection process tremendously.

7. The Quality Control laboratories at the ports carry out several inspection processes on imported foodstuffs. Radioactivity inspection was a newcomer to the battery of tests usually

conducted and a position in the sequence of these tests has to be marked for radioactivity examination. The decision was taken by the Quality Control Department to conduct the radioactivity test at least concurrently with other tests, if not before,since presence of unacceptable levels would render other test unwarranted.

8. At the beginning of the inspection process, an unanticipated situation of great worry and concern evolved among seaport and airport workers handling foodstuffs found to be radioactively unacceptable, and ordered to be reshipped to its port of origin. This is not unusual anxiety on the part of these workers, who have no technical knowledge concerning radiation. However the problem was alleviated after oral communications with the workers and administrators in charge, and after distributing pamphlets written in nontechnical language explaning facts about radiations.

9. The inspection process for radioactivity levels in imported foodstuffs became widely known and had a tremendous impact on the public as well as on different governmental departments. As regards the public, this new awareness was definitely a positive outcome but it had also, in some cases, a negative aspect. Some members of the public became overly worried especially mothers of newborn babies and pregnant women. This matter was handled by having local experts in the field of radiation and health use the different news media to explain in simple language the relevant facts about radioactivity contamination of foodstuffs. This information campaign, so to speak, helped to ameliorate the unnecessary anxiety of some members of our public.

OBSERVATIONS AND RECOMMENDATIONS:

1. A lot of work needs to be done regarding setting internationally agreed upon maximum limits of radioactive contamination of adult and infant food items. The various situations of food mixes and eating habits should be taken in consideration.

2. Certificates accompanying food shipments from various countries declaring the food items to be"radioactively safe for human consumption" need to be issued by officially designated technical organization of the country shipping the foodstuffs.

3. The techniques used by the different countries for radioactive monitoring of food shipments should be standardized and agreed upon: Some certificates for example claimed that the accompanying food shipment was free from radioactivity as checked by Geiger survey meter. This of course is unacceptable technique at these very low radioactivity levels and was indeed found in error in several shipments.

4. Because another Chernobyl-like accident is not impossible , it would be extremely helpful to have publications written in the official languages of the United Nations and covering in simple non technical language the facts or rather the"true or false" about radioactive contamination of foodstuffs. Having such

publications distributed to member countries will be of great value in time of need.

5. The IAEA or WHO or some similar international organization may arrange a workshop or training program for administrators and public relation employees at seaports and airports covering simple facts about radiations and health and food contamination with radioactivity. Intelligent cooperation from these people is very important to the success of the inspection process.

6. The radioactivity inspection process has alerted several concerned scientific institutions in this country to the insufficiency of the present data on the radioactivity of our own natural environment and foodstuffs. This may actually be the situation with many other developing countries. Complete data on the radioactivity of locally grown food crops, meat from local cows and sheep as well as milk and milk products would have formed valuable baseline data for the food radioactivity inspection process.

7. In our management of the food inspection process, priority was given to perishable foodstuffs such as chilled meat and small live chicks...etc., But when these items show in the gross radioactivity check at the ports, increase in activity above background, the shipments will have to wait for the completion of the qualitative and quantitative tests. It is therefore advisable that the suppliers of these perishable foodstuffs be doubly sure that their shipment is free from radioactive contamination, so as to avoid possible financial losses.

POST-CHERNOBYL WHOLE-BODY COUNTING MEASUREMENTS IN THE FEDERAL REPUBLIC OF GERMANY

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Introduction: The Federal Republic of Germany (F.R.G.) is situated about 1500 km west of Chernobyl, has a borderline approximately parallel with that of the Soviet Union, is densely populated, and was the first contaminated Central European country outside the Eastern Block. Furthermore, a network of whole-body counting laboratories extends over the entire F.R.G.'s territory for radiation protection purposes. These institutions started measurements immediately after the radioactive cloud reached Germany (April 30, 1986). The paper summarizes the measurements of Cs-134 and Cs-137 body burdens of reference groups under both the regional aspect and the aspect of body burden development from May 1986 till October 1987.

Materials and Methods: For the feeding areas of the institutions with their respective reference groups, see fig. 1 which also shows the average Cs-137 surface contamination as of May, 1986. Note that two installations are located in Munich (München) and that the reference group of South-East Bavaria has been monitored by the BGA Munich counter.
In table 1, the main technical data of the 9 monitors are summarized. They all use

	Detector d/h	No.	Meas. Time (min)	Efficiency* Cs-134 (/min/Bq)	Efficiency* Cs-137 (/min/Bq)	Background Cs-134 (/min)	Background Cs-137 (/min)	Detector Geometry
Saarland	8"Ø 4"	2	10	0.3	0.5	100	150	scanning, variable speed
Karlsruhe	8"Ø 4"	4	5	0.63	0.72	244	412	fixed, stretcher
Frankfurt	8"Ø 4"	1	20	0.294	0.334	53	124	fixed, tilted chair
Mainz	5"Ø 4"	12	5	1.07	1.01	270	680	fixed, stretcher
Berlin	5"Ø 4"	4	20	0.43	0.67	90	140	fixed, stretcher
Duesseldorf	11.5" 4"	1	10	0.92	0.80	315		fixed, tilted chair
Juelich	4"Ø 2"	4	10	0.12	0.14	100	200	fixed, stretcher
Munich BGA	5"Ø 4"	4	20	0.30	0.34	70	100	fixed, stretcher
Munich GSF	5"Ø 4"	4	25	0.190	0.202	123	210	fixed, stretcher

Table 1; see text.

* 70kg standard man

sodium iodide detector(s), at least for the results reported in this study.
The number of persons measured differs from installation to installation as well as from month to month but is normally between 10 and 20 for each reference group and month.
Thanks to a program being started about 4 years ago and still in progress, all monitors have been subjected to calibration intercomparison runs and, if necessary, recalibrated. Thus, we may assume that the average deviations of mean specific radiocesium acivities due to differences in monitor calibration do not exceed 20%.

Results: The results of the counter measurements of the reference groups in the different areas and for the first 18 months after the Chernobyl accident are shown in figs. 2 (males) and 3 (females), respectively.

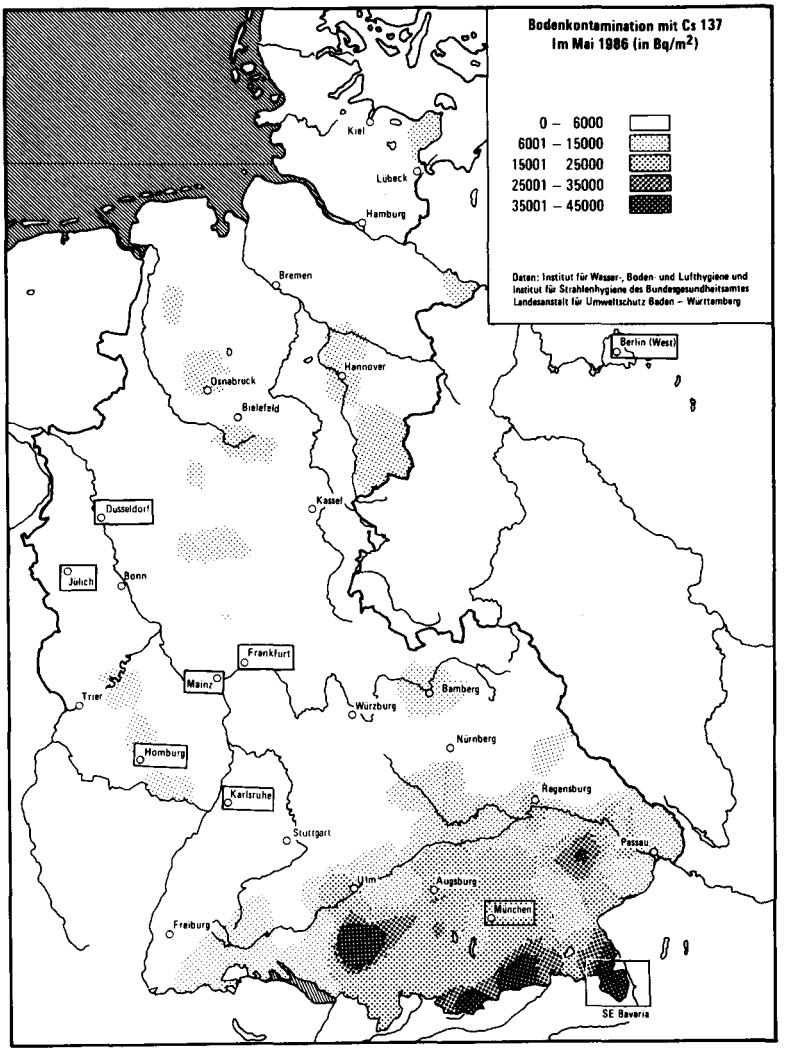

Fig. 1: Map of the Federal Republic of Germany including the locations of the reference groups. The Cs-137 surface contamination as of May, 1986, is also shown.

Owing to lack of space, only the sum of Cs-137 and Cs-134 is given. To make the diagrams readable and since the relation is nearly constant (2 : 1 to 3 : 1), we did not diffentiate graphically between the values for both nuclides.
From the activities measured, we calculated the dose equivalents (a) for the first year after the accident (i.e. from May, 1986, until April, 1987), (b) for the year 1986, and (c) for the year 1987 (November and December estimated); see table 2.

1987 (November and December estimated); see table 2.
The calculations were performed under the assumption that the monthly values for the Cs-137 and Cs-134 content measured in each reference group can be considered as being maintained, to a good approximation, at almost a constant level for such a relatively short period of time.

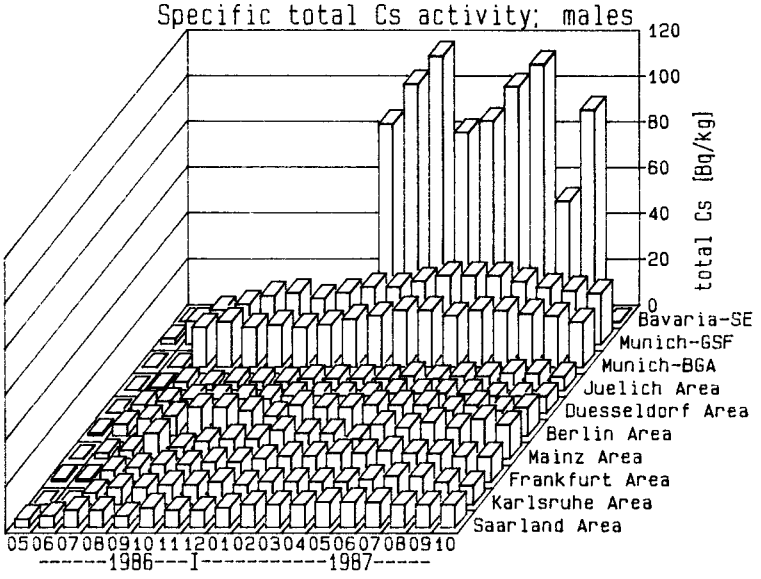

Fig. 2: Total cesium body burden; **males**

We used the dose value per decay and took into account the age dependence as recommended by ICRP.

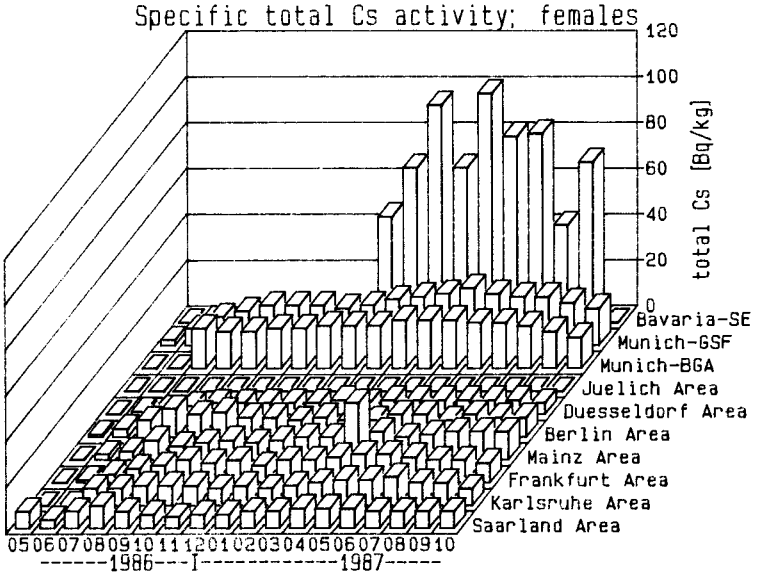

Fig. 3: Total cesium body burden; **females**

Discussion: Corresponding to the surface contamination (fig. 1), the incorporated activities are higher in the southern part of Germany, with a pronounced maximum in South-East Bavaria.
As an important result, it should be noted that their is at least no further increase in radiocesium body burden since the mid-1987; some institutions already report a decrease.
Nevertheless, the retrograde "12 months' doses" are still on the increase for most of the year 1987. Starting September, a light decrease is observed.
In general, the dose equivalents for 1987 are about twice those in 1986. However, the absolute dose equivalents for the majority of the population of the Federal Republic of Germany is well below 100 (50) microSv/year for males (females) and therefore negligible in comparison with the value due to natural sources (about 2000 microSv/year). This is valid even for South-East Bavaria.

Dose Equivalent (microSv/year)

Area	(a) First year		(b) 1986		(c) 1987	
	male	female	male	female	male	female
Saar-land	27	19	15	12	34	21
Karls-ruhe	28	21	18	13	32	25
Frank-furt	25	20	13	11	37	27
Mainz	24	21	13	10	44	31
Berlin	33	22	19	15	40	22
Duessel-dorf	20	11	11	6	24	15
Juelich	12	n.a.	6	n.a.	19	n.a.
Munich BGA	62	44	35	26	75	48
Munich GSF	65	41	35	24	85	48
Bavaria-SE (BGA)	230	150	130	90	260	160

Table 2; see text n.a.: not available

Acknowledgments: In particular during the first months following the Chernobyl accident, the counter installations were inundated with requests for measurement from the local population as well as with calls from both the government and the press. The authors would like to thank their respective staff members for their indefatigable cooperation. Special thanks are due to Dr. I. Gans for the pattern used for fig. 1.

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HANDLING OF RADIOACTIVE FALLOUT PROBLEMS AT CHERNOBYL ACCIDENT (1986) AS COMPARED WITH THAT OF BIKINI ACCIDENT (1954)

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I BIKINI ACCIDENT (1954)

We conducted an analysis in Japan of the highly radioactive fall-out on the Japanese fishing boat No.5 Fukuryu Maru that was engaged in fishing about 150 km east of Bikini at the time of the thermonuclear test conducted early in the morning of 1 March 1954, and which returned to Japan in the middle of the same month. According to the statements of some of the crew, a few hours after the thermonuclear detonation in Bikini the whitish dust began to fall on the boat so heavily that for a period they could hardly bear to open their eyes and mouth. It continued to fall for several hours. Some of the crew apparently tasted it, to see what it was, without knowing that it was highly radioactive. Owing to the difficulty of dose estimation without more accurate information on the initial condition, the radioactive fall-out conditions on the boat were experimentally reproduced by M. Miyoshi, the chief physician in charge of treatment of the exposed crew at the Tokyo University Hospital,using pulverized coral reef. This experiment was carried out in the presence of the crew as witnesses of the actual amount of ash which had fallen on the boat, This amount was then estimated to be about 3.38-8.52 mg/cm^2. The radioactivity of the ash was estimated by extrapolation to be about 1 Ci/g at the time it fell on the boat. Taking into consideration various possible exposure conditions of the crew during the voyage,the probable gamma dose was estimated to be in the range 170-600 rads. The degree of uncertainty was far greater for the internal dose. The long-lived radionuclides detected in organs such as the liver many weeks later could not be considered the only sources of internal exposure. Depending on the assumed degree of initial incorporation of short-lived radionuclides, a wide range of estimates was possible: for the liver, a few rads to a few tens of thousands of rads, the probable dose range being 10-10^4 rads; and for bone and bone marrow, a few rads to about 60 rads. If we assume a non-uniformity factor of five for bone, the local dose could be five times higher.The thyroid dose was estimated to be about 10-10^3 rads. Radiation syndromes such as radio-dermatitis, epilation, decrease of leucocytes, decrease of spermatozoa, etc. were observed in the exposed.Some of the larger aggregates of the Bikini dusts collected from the fishing boat 'Fukuryu Maru' were found to have a size of about 0.1-0.5 mm. (0.3 mm in average). However these granules were found to consist of finer unit particles of the size 0.1-3 μm with cubic or spindle shapes. Some of the fine particles of indefinite shapes were found to have a size less than 0.1 μm on electron microscopic examination. From electron micro diffraction and X-ray diffraction studies,the Bikini dust was confirmed to have the crystal structure of calcite while the coral reef is aragonite. From these findings it may be inferred that the coral reef evaporated at the time of the H-bomb explosion and recrystallized in the air into calcite with the inclusion of radioactive nuclides produced by the explosion. Double-coil magnetic-lens type beta-ray spectrometer was used to identify some of the radionuclides. The beta-ray activity of the rare earth elements mixture was about 30-60% of the total beta-activity of the original Bikini ash while that of uranium 237amounted to as much as 10-20% at the end of March 1954. This suggests the existence of

a large amount of uranium 238 in the March 1st bomb, if we assume uranium 237 were produced by (n-2n) reaction from uranium 238. The unexpectedly large amount of radioactivity release by Bikini test was officially announced by USAEC on January 15, 1955.

II CHERNOBYL ACCIDENT (1986)

The accident of Chernobyl Unit 4 took place on April 26, 1986. The Soviet experts calculated that the first power peak reached 100 times the nominal power within 4 seconds. Energy released in the fuel from the power excursion (>300 cal/g) suddenly disrupted part of the fuel into minute pieces. This disruption mechanism is known from experiments in safety research programmes. Small hot fuel particles may have caused a steam explosion. The energy release shifted the 1000 ton reactor cover plate and cut all cooling channels on both sides of the reactor cover. After 2 to 3 seconds a second explosion was heard and hot pieces of the reactor were ejected from the destroyed reactor building. It may be assumed that steam-zirconium and other exothermic reactions occur. Hydrogen and CO is produced and may explode. The destruction of the reactor allowed the ingress of air which led subsequently to graphite burning. Destruction of the Chernobyl containment and core structures led to release of radioactivity from the plant. The USSR experts estimated 100% of the noble gas radionuclides escaped the plant. Of the remaining, condensible, radionuclides the release amounted to about 5×10^7 curies or about 3-4% of the core inventory of radioactivity. This release was composed of about 10-20% of the Cs, I and Te inventories and about 3-6% of the inventories of other radionuclides.

The release of radionuclides from the Chernobyl plant did not occur as a single acute event. Rather, there was initial, intense release associated with the destructive events in the accident. Release rates decreased over the next few days probably as a result of accident management activities undertaken. Release rates were about 2×10^6 Ci/day five days after the accident initiation. At that point, the release rates began to increase and reached 8×10^6 Ci/day about nine days after the accident initiation. When UO_2 is further oxydized to U_3O_8, most of the fission products contained in UO_2 may be released. There was, then, a drop in the radionuclide release to 10^3 Ci/day. Release rates have continued to decline since that time. Radioactive releases corrected to 6 May and have an uncertainty range of $\pm$ 50%. In case of a nuclear bomb explosion, all the radioactive materials are instantaneously released to the environment. In case of Chernobyl Accident, volatile radionuclides such as I and Cs were more predominant.

On 3-4 May 1986, the radioactivity included in the surface air was observed to increase suddenly at the central part of Japan. The radioactivity detected in different parts of Japan was as follows: I-131, Te-132, Ru-103, Rh-106, Cs-134, Cs-137, Ba-140, La-140, Mo-99, Tc-99m. The radioactivity ratio Cs-137/Cs-134 observed in the dust and rainwater sampled by Morishima, et al. at Kinki University, Osaka, Japan, was about 2.0 ± 0.2 during the period 4-11 May, in good agreement with the ratio 2.0 ± 0.3 reported by Aoyama, et al. The air concentration of I-131 was observed to reach about 90 mBq/m^3 on 9 May at Kinki University, Osaka, and the percentage of I-131 was estimated to be over 50% of the total beta activity on 10 May. Kr-85 was estimated by cryogenic separation from air samples collected in an iron cylinder followed by gas chromatographic purification at the Geochemical Laboratory of Meteorological

Research Institute. Before Chernobyl Accident, Kr-85 was 0.9Bq/m^3, but increased to 1.04 Bq/m^3 on May 6, falling steadily to the normal. There is a possibility that a small fraction of Chernobyl radioactivity went up to the stratosphere, but this fraction would be very small as compared with that of Bikini H-bomb test. Because of the large explosion power of H-bomb, the global fallout of Bikini Accident was larger, but the local and regional fallout of Chernobyl Accident would be more significant.

III RADIOACTIVE FALLOUT PROBLEMS AND INTERVENTION LEVELS

According to the USSR-Report, the released radioactivity of Cs-137 at the Chernobyl Accident (1986) is about one megacurie ($\pm$ 50%). The fraction of Cs-137 released is estimated to be about 13% of core inventory. The released radioactivity of Cs-137 at the Bikini Accident (1954) may be estimated to be about the same order of magnitude. Assuming 15 megaton TNT equivalent with fission and fusion energy about 50;50 and assuming U-238 fission spectrum the total Cs-137 released at the time of explosion is also estimated about one MCi. However, the local (30 km) close-in fallout of Cs-137 is estimated about 100-1000 times higher with the Chernobyl Accident and the fallout of Cs-137 in an area equivalent to Central Europe region is about 20 times higher with the Chernobyl Accident.

The fraction of radionuclides of refractory elements was more predominant in the thermonuclear test conducted at Bikini on March 1, 1954. The fission yield of Cs-134 is much lower than that of Cs-137, and it may be difficult to identify this nuclide in the fallout due to nuclear weapons test. However, fission yield of Cs-133 is much larger, and Cs-134 is produced by neutron capture of Cs-133 in the nuclear reactor. In case of Chernobyl accident, Cs-134/Cs-137 was about 0.5 which corresponds to the burnout of about 10,000 Mwd/t of nuclear fuel. A large number of highly contaminated tuna fish brought back by the fishing boat showered by strongly radioactive fallout of Bikini test were distributed in the market. Sometime later, milk and vegetables were also contaminated by the fallout all over Japan. A confusion was created about the handling of these radioactive fallout problems in Japan in 1954. The situation encountered in Japan very much resembles that in Europe after Chernobyl accident in 1986. The intervention level of foodstuffs was one of the most important items of discussion. The intervention level under initial emergency condition and that under more or less stabilized chronic situation should be distinguished. In the latter case we may have more time to discuss the matter. However, in the first case, it is not known at the beginning whether the contamination may increase or not, and the extent of increase of external dose and that of inhalation dose, etc. are all fuzzy. Under such circumstances the derived intervention level based on maximum permissible dose or body burden may not be used to the full. Whether we take 1/5, 1/10 or 1/100 is up to the subjective judgement based on intuitive optimization of the decision maker. This constitutes a most difficult case of optimization under fuzzy environment and decision making with fuzzy information. ICRP optimization technique may not be applicable in such cased. When non- or less-contaminated alternative foodstuffs are abundantly available, one may take very strict measures and very low intervention levels to compensate external and inhalation dose. On the other hand, when all available foodstuffs are contaminated, one may be obliged to take, at least temporarily, a highly intervention level for survival under emergency condition.

The first contaminated tuna fish which were brought back to Japan by No.5 Fukuryu Maru in the middle of March 1954 were found to be emitting much stronger radiation from the surface (0.1-1.0 $\mu Ci/cm^2$) than from the inside,and the government set the tentative discarding level of radioactive contaminated fish at 100 cpm as measured at 10 cm from the wet surface with the beta-ray counter with 3.5 mg/cm^2 mica window plus 2.5 mg/cm^2 plastic cover to protect the counter window. The natural counts of this counter were about 30 cpm. Geometry of this counting condition was estimated to be 1/400 and the beta-ray absorption by mica window and plastic cover and water and scale on the surface of the fish about 2.5. The overall efficiency of beta-counting may be estimated about 1/1000. Since the first contaminated fish had very strong radioactivity on the surface, the above emergency intervention level was adopted tentatively to screen the high surface-contaminated fish. The fish caught later had much weaker radioactivity mostly inside and much lower intervention levels based on the then ICRP recommended action for respective radionuclide identified in the fish. The tuna fish is an expensive fish and a balance between the risk due to economic loss, the risk due to consumption of contaminated food, the pressure of public opinion, the psychological and political effects, the effects on international relation and international trade, etc. may have to be considered in setting an intervention level. If the contaminated food constitutes a significant fraction of staple food of the people, nutritional problem must be considered. A simple cost-benefit analysis may create a confusion and socio-political problems. According to an USSR expert (1987), the initial emergency intervention level at Chernobyl (1986) was 5 rem for external and 5 rem for internal irradiation, but later the level was reduced to a lower level. When no alternate food is available, one must assume a higher level under emergency condition. Through these two accidents the necessity of internationalization of radiation protection and nuclear safety was strongly felt.

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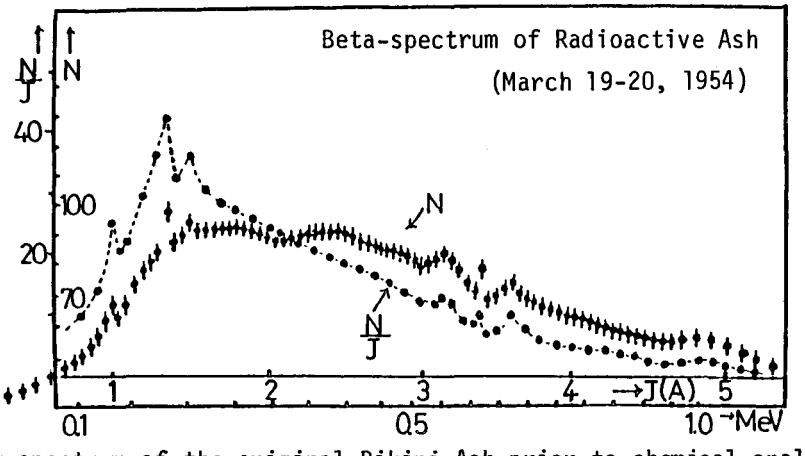

Beta-ray spectrum of the original Bikini Ash prior to chemical analysis March 1954. Double-Coil Magnetic-Lens type beta-ray spectrometer was used.

FOLLOW UP MEASUREMENTS OF ^{131}I THYROID ACTIVITY IN 54 GERMAN CHILDREN AFTER THE CHERNOBYL ACCIDENT

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INTRODUCTION

In the first days after the Chernobyl accident the uptake of ^{131}I was a primary matter of concern with respect to the internal radiation exposure. In the body, nearly all iodine is stored in the thyroid. Especially in children the incorporation of radioisotopes of iodine may result in high organ doses, because of the small thyroid masses.

The thyroid dose can only roughly be estimated from a single measurement of the thyroid activity or from the activity in the air or in foodstuffs. The accurate calculation of the dose, however, requires follow up measurements of the time course of the ^{131}I activity during the whole period of incorporation. In order to get detailed information on the radiation dose to the thyroid in Germany after the Chernobyl accident, the time course of ^{131}I activity was measured in 54 children during May and June 1986 and the individual organ doses were calculated.

SUBJECTS AND METHODS

In 53 healthy children (29 boys, 24 girls; age: 1 - 16 years) and in an eight years old girl with a hyperfunction of the thyroid gland, the time course of the ^{131}I activity in the thyroid was followed by up to 6 measurements performed in May and June 1986. The counting time ranged between 1 and 5 minutes. A 3" by 3" NaI(Tl)-crystal in a lead shielding was applied as thyroid monitor. The distance between the crystal and the neck surface is fixed by an adjustment device to 12.5 cm. A multichannel pulse height analyzer was applied for the spectrum collection. The count rate in the energy window from 0.31 to 0.41 MeV was used as a measure of ^{131}I activity. The transformation of the obtained count rates into activity requires a careful calibration of the system with several thyroid phantoms (1). These measurements showed that the volume of the thyroid has only little influence on the counting efficiency, but there is a strong dependence of the measured count rate on the distance from the detector to the thyroid. The calibration factor which was used to calculate the ^{131}I activity in the thyroid of children corresponds to a thickness of the tissue over the thyroid of 10 mm (2). After the decay of ^{131}I the individual background count rates were measured. Net count rates were converted into thyroid activities A_T and individual retention functions $A_T(t)$ were evaluated to calculate the cumulated activity $\tilde{A}_T$ in the thyroid:

$$\tilde{A}_T = \int_0^\infty A_T(t)\, dt$$

It was assumed that the earliest uptake of ^{131}I in the thyroid was on the 3rd of May, because no or only very low ^{131}I activity was measured in four adults on the 2nd of May 1986 (1). The thyroid dose D_T is calculated from

$$D_T = \frac{\tilde{A}_T}{m_T} \left(\bar{E}_\beta + a(m_T,E)\ \bar{E}_x + b(m_T,E)\ \bar{E}_\gamma \right)$$

with:
- m_T: mass of the thyroid
- $\bar{E}_\beta$: mean emitted energy of ß-particles per decay
- $\bar{E}_x$: mean emitted energy of X-rays per decay
- $\bar{E}_\gamma$: mean emitted energy of gammarays per decay
- a,b: absorbed fractions of photon energy

Absorbed fractions of the photon energy were determined from values published in MIRD-Pamphlet No 8 (3). In 31 children the thyroid volumes were determined sonographically, for all other children the values were interpolated corresponding to their age.

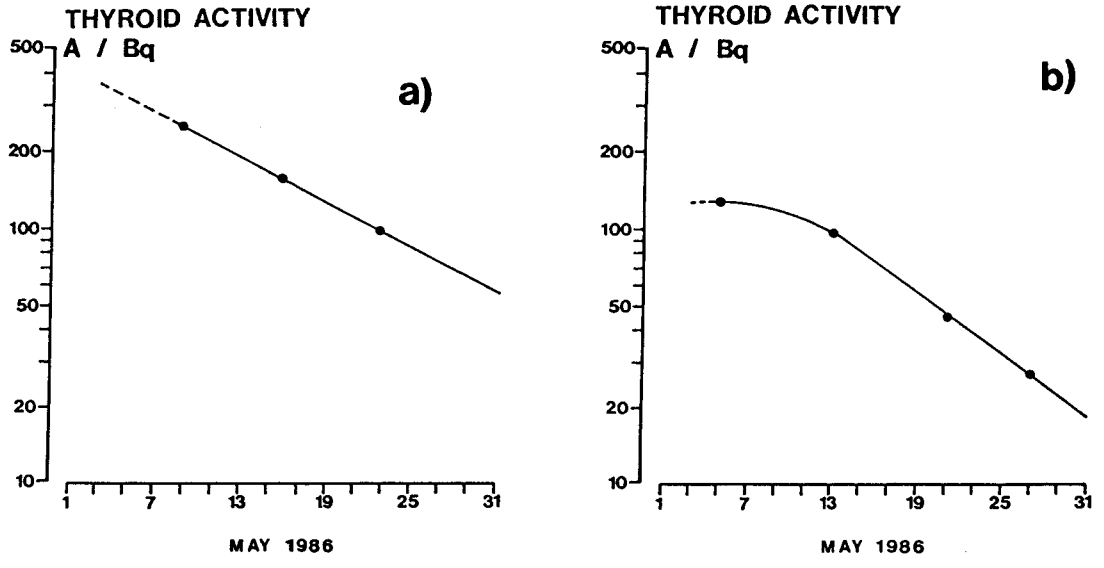

Figure 1: Time courses of ^{131}I thyroid activity in an 11 years old boy (a) and in a 5 years old girl (b)

RESULTS AND DISCUSSION

As examples, the figure 1 shows the time courses of ^{131}I thyroid activity of an 11 years old boy and of a five years old girl. For the calculation of the cumulated activity this time course could mostly be described as a monoexponential function. In those cases where no monoexponential decrease was observed, $\tilde{A}_T$ was determined by linear interpolation of the activity values. Since in the Rhein-Main-region it was not raining before May, 5th, most of the radioactive iodine was inhaled during the first days in May. But part of the ^{131}I must have been ingested, because the half life of the monoexponential decrease of the activity was higher than the physical half life time of ^{131}I.

The sonographically determined thyroid volumes ranged between 2 and 21 ml. Values of the calculated doses are given in figure 2. Radiation doses to the thyroid were between 0.3 and 2.1 mSv. According to the age dependent thyroid volumes, a decrease

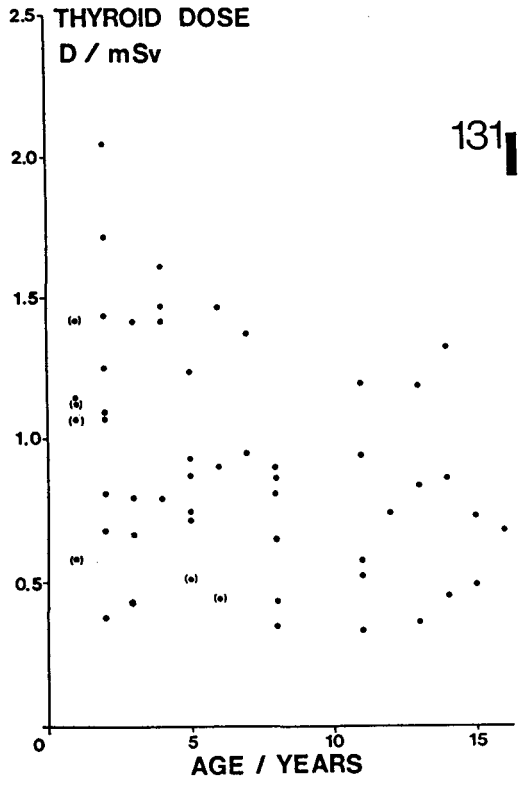

Figure 2: Thyroid doses in 54 German children

of the thyroid dose with increasing age was observed. The girl with the hyperfunction of the thyroid showed no deviation in thyroid activity and dose from the other children of comparable age.

These values are representative for nearly all parts of Germany except Southern Bavaria. For that region higher values have been reported (4). Thyroid doses can be assumed to increase to the south east corner of Germany (Berchtesgaden), corresponding to the higher fallout and washout. Nevertheless these data show that even about 1500 km from the site of the accident significant thyroid doses due to radioactive iodine were observed.

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ENVIRONMENTAL ACTIVITY LEVELS MEASURED AT THE NIJMEGEN UNIVERSITY AFTER THE CHERNOBYL ACCIDENT

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As a consequence of the Chernobyl nuclear reactor accident on April 26, 1986 large areas of Europe became contaminated with fission products. Since then many measuring programs have been started in order to determine the radionuclides and the activity levels in air, water, soil and vegetables. They served to determine which countermeasures had to be taken to keep the dose equivalent to the public as low as reasonably achievable. At the University of Nijmegen rainwater was collected during the first three weeks of May 1986 and the samples were measured with a Germanium detector coupled to a 4096-channel MCA. From quantitative analysis of the γ-spectra thus obtained and the amount of rainwater, we reported 55 GBq as the total activity precipitated per square kilometer in the Nijmegen area (Be87). People all over Europe were very concerned about the radiation risk when consuming contaminated foodstuffs. This caused us to measure a variety of samples we obtained from different countries.
Table 1 shows measured activity levels in such samples.

Sample	Activity ($Bq.kg^{-1}$)		
	^{134}Cs	^{137}Cs	Total
Mushrooms (The Netherlands)			
a) Paxillus involutus (Sept. 1986)	110	240	350
b) Chantarelles (Oct. 1986)	80	220	300
Cherries (Greece, Aug. 1986)	190	410	600
Hay (Serfaus, Austria, July 1986)	1400	2800	4200

No detectable activity was found in samples of milk (Krakow, Poland, June 1986), drinking water (Krakow, Poland, June 1986 and Leningrad, USSR, Sept. 1986), peaches (Nijmegen, The Netherlands, Sept. 1986) and several other fruits picked in the autumn of 1986 in The Netherlands.

Particular attention was given to the air inlet filters of large air conditioning systems. Because of the enormous flow rates of air of such systems the total accumulated activity in these filters could reach considerable values. Table 2 presents the measured activities per radionuclide of a representative sample from an air inlet filter from one of the buildings of the University of Nijmegen. This sample was taken on May 16, 1986, at which time the total filter was due to be changed.

Table 2. Measured activities in an air inlet filter at the University of Nijmegen after the Chernobyl accident.

Nuclide	Specific Activity ($kBq.kg^{-1}$) 16 May, 1986
$^{95}Zr/^{95}Nb$	50
^{103}Ru	56
$^{106}Ru/^{106}Rh$	360
$^{129m}Te/^{129}Te$	1000
^{131}I	730
$^{132}Te/^{132}I$	300
^{134}Cs	350
^{137}Cs	630
$^{140}Ba/^{140}La$	800
^{141}Ce	50
^{144}Ce	50
Total	4880

The adsorption coefficient for atmospheric dust for this type of filter is 0.35 and the total volume of air which passed through this section of the filter from May 2 until May 6 amounted to $1.6 * 10^6$ m^3. However, care must be taken in calculating the air activity concentration of the different radionuclides from the values given in table 2, since the adsorbed quantities may not be indicative of the concentration, for instance in the case of iodine. Worried technicians who had to change these filters were reassured by monitoring during the exchange. They were advised to wear dust masks to prevent internal contamination by inhalation.

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COMPARATIVE EVALUATION OF INCORPORATED ACTIVITY BY FOUR DIFFERENT WHOLE BODY COUNTERS

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INTRODUCTION

The assessment of internal radiation exposure requires the determination of the activity of incorporated radioactive substances and the knowledge of their biokinetic behaviour. If possible, the total body activity should be determined by direct measurements applying whole body or partial body counting. Thereby, serious problems can arise from differences in body size and shape, as well as from variations in the distribution of radioactivity within the subject. To overcome these problems, careful calibration of whole body counters is necessary. Long-term reproducibility of the calibration may be achieved by regular measurements of phantoms or activity sources in standard position. The accuracy of determinations of activity can only be realized by intercomparisons with the results obtained with other such devices. Several intercomparisons among whole body counters (1-4) showed that improper calibration may result in errors of estimated body activities of up to a factor of two or even more. With respect to the variations in the fall out of fission products from the Chernobyl accident, consistency of total body measurements throughout Europe for the estimation of the consequences due to ingestion of radioisotopes is required. The present study was aimed to compare the results obtained by four different whole body counters in Budapest/Hungary, Seibersdorf/Austria, and Frankfurt/Germany.

INSTRUMENTS

The whole body counter of the Central Research Institute for Physics, Budapest, is housed in a steel chamber with inner lead and copper lining. The routinely applied measuring arrangement is a modified single detector (6" x 4" NaI(Tl)-crystal) scanning-end-stop geometry with a scan length of 126 cm (5). The measured spectra are transferred to a PDP 11/34 computer and evaluated by least square fitting procedure using a preselectable set of calibration spectra stored in the spectrum library. For calibration, a BOMAB type standard phantom was used assuming uniform activity distribution. The measuring time applied for the persons of this intercomparison was 1294 seconds. The expected overall uncertainty for the determinations of the whole body activities of ^{134}Cs, ^{137}Cs, and ^{40}K is estimated as less than 11%.

The whole body counter operated by the Ges. für Strahlen- und Umweltforschung, Inst. für Biophysikalische Strahlenforschung, Frankfurt am Main, is housed in a basement with concrete walls of 2 metres thickness. The device is installed in a steel chamber (16 cm wall thickness). It applies the tilted chair geometry with a single 8" x 4" NaI(Tl)-detector, equipped with four 3"-photomultipliers. Gamma ray spectra are collected on a 0.5 K multichannel analyzer (5). Calibration of the counter for the measurement of ^{134}Cs, ^{137}Cs, and ^{40}K was performed using plastic bottles of 1 l and 2 l volumes, by which different body weights (between 10 kg and 100 kg) as well as different body shapes could be simulated. For this study, the subjects were measured for 20 minutes. The expected overall uncertainty for the determination of total body activities of ^{134}Cs, ^{137}Cs, and ^{40}K was estimated to be less than 10%.

The body counter at the Austrian Research Center Seibersdorf, Institute for Radiation Protection, employs a shadow shield tilted chair geometry. The outer shield consists of low activity 30 cm thick concrete walls lined by 1 cm steelplates. 4 NaI(Tl)-detectors are used. The backside of the chair is shielded by 5 cm of lead with copper lining and two detectors are integrated in the backside. The other two detectors mounted in front of the subject are also shielded with lead of 2-4cm thickness with copper lining. Thus, minimum detectable activities (3σ of background and 100% yield) for 1000 s measuring time are in the order of 100 Bq. Calibration of the counter was performed with a selfmade bottle-phantom (PTB-standard-solution with 7 radionuclides) by which different body weights and sizes could be simulated. Additionally, a high purity Germanium detector of 30% relative counting efficiency is installed in the same device for measurements with high energy resolution. The measured spectra are transferred to a PDP 11/34 computer and evaluated by a modified SAMPO-80 programme. The expected uncertainty for Caesium and Potassium is estimated as less than 10% (without statistical error).

The whole body counter at the laboratories of the International Atomic Energy Agency, Forschungszentrum Seibersdorf, is housed in a 249x188x233 cm (lengthxwidthxheight) steel chamber. The shielding is composed of 19 cm mill steel and 3 mm virgin lead. Also various counting methods and geometries are available. Only the tilting chair geometry is used for routine measurements. The measuring arrangement consists of a single 8" x 4" NaI(Tl)-detector with 3 photomultipliers, 1 K multichannel analyzer (Canberra S 35 plus), and for controlling and calculation a commodore 4032 computer. The measuring time for gamma ray counting is 600 seconds. The expected uncertainty without statistical error for the determinations of the whole body activities of ^{137}Cs and ^{40}K is estimated as less than 6%. The detection limit for ^{137}Cs is about 40 Bq.

SUBJECTS

The intercomparison measurements were performed on 8 healthy adult subjects (3 females, 5 males). Their body weights ranged

from 55 kg to 94 kg. Since the activities of the cesium isotopes varied slightly during the investigation period in spring 1987, all measurements for a particular person were carried out within one week. Only one subject (S.F.) underwent two courses of measurements, the first of which comparing the results as determined in Seibersdorf with those of Budapest and the second comparing Seibersdorf with Frankfurt. Subjects were measured at their home whole body counters immediately before and after the visit to all the other laboratories and the means of these two measurements were used as reference values.

Table 1: Total body activities of ^{134}Cs, ^{137}Cs, and ^{40}K as measured with the whole body counters in Budapest (KFKI), Seibersdorf (FZS and IAEA), and Frankfurt (GSF) in 8 healthy adult subjects.

Subject	Nuclide	Budapest KFKI (Bq)	Seibersdorf FZS (Bq)	Seibersdorf IAEA (Bq)	Frankfurt GSF (Bq)
H.Ch.	^{137}Cs	480	600	442	547
f, 55kg	^{134}Cs	187	280	-	204
	^{40}K	3224	3036	2817	3020
R.P	^{137}Cs	734	740	887	762
m, 93kg	^{134}Cs	286	310	-	304
	^{40}K	6385	4883	6291	5930
R.V.	^{137}Cs	466	520	479	503
f, 63kg	^{134}Cs	182	260	-	178
	^{40}K	4038	2441	2911	3285
W.E.	^{137}Cs	1005	1070	1096	1073
m, 84kg	^{134}Cs	392	490	-	420
	^{40}K	5321	4100	5603	5005
A.A.	^{137}Cs	1849	1830	1626	1786
m, 71kg	^{134}Cs	660	723	-	591
	^{40}K	4538	4539	3913	4223
B.E.	^{137}Cs	852	740	712	866
f, 68kg	^{134}Cs	305	286	-	259
	^{40}K	3318	2817	2661	3051
F.I.	^{137}Cs	1186	930	1177	1182
m, 94kg	^{134}Cs	424	374	-	375
	^{40}K	4570	4696	5133	4132
S.F.	^{137}Cs	3432	3310	3646	
m, 80kg	^{134}Cs	1338	1290	-	
	^{40}K	4132	3787	5133	
	^{137}Cs		3800	3752	3742
	^{134}Cs		1540	-	1276
	^{40}K		3975	4194	3963

RESULTS

The data obtained from the four whole body monitor laboratories are summarized in table 1. Since the calibration for ^{134}Cs had not been completed for the instrument operated by the IAEA, no body activities of that radionuclide could be evaluated for this place. In general, the results obtained show a good agreement. For the radionuclide ^{137}Cs, there are no systematic deviations between any two of the four laboratories. Although particular measurements even on the same day show differences of activity determination of up to about 25%, the sums of ^{137}Cs-activities of all measurements coincide within 5%. Since in the calculation of ^{137}Cs body activity the ^{134}Cs contribution in the photopeak region of ^{137}Cs has to be considered but not vice versa, even better agreement could be expected for the evaluation of ^{134}Cs activities. Actually, the observed deviations are greater for ^{134}Cs than for ^{137}Cs. This may be due to the poorer statistics and the different evaluation methods applied. Similar differences of up to 15% are also seen for the mean values of potassium (^{40}K).

CONCLUSIONS

The mean values of all three isotopes show no systematic deviations between the four laboratories, whereas for a particular person the differences may exceed those that could be expected from counting statistics. This intercomparison again demonstrates that in whole body counting systematic errors in reproducibility of subject and detector positioning together with the evaluation technique applied are predominating. During the preceeding years separate comparative measurements have been carried out in Eastern Europe including Budapest (4) as well as in Western Europe including Frankfurt and Seibersdorf (1,3). The data presented here close the gap between both parts of Europe, showing that there are no systematic deviations in the assessment of total body activity by direct measurements applying whole body counting.

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SELECTION OF SUITABLE LIQUIDS AND SOLIDS FOR A PHANTOM FOR INTERNAL DOSIMETRY MEASUREMENTS

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1. INTRODUCTION

The absorbed dose to various body organs from internally deposited radionuclides has two components, the β-/γ-dose from activity deposited in the organ itself and the γ-doses from activity deposited in all the neighbouring organs. Average dose equivalents in different target organs per transformation in a source organ can be determined either by Monte Carlo photon transport calculations or by point kernel calculations. Limitations of the computational methods as well as of the mathematical model of the human body call for experimental validation of internal dose calculations. Uncertainties of physical data such as tissue compositions may be of minor importance compared to the overall uncertainty but need to be experimentally demonstrated.

2. TISSUE EQUIVALENT MATERIALS

For internal dose measurements in phantoms, the phantom material and the tissue it simulates should have identical responses to both photons and electrons. Therefore, the photon mass-attenuation and mass-energy absorption coefficients, μ/ρ and μ_{en}/ρ, as well as the electron mass-stopping power, S/ρ, should all be identical for the two materials over a wide energy range [1]. As equal volumes of the two materials should have the same mass, the mass density, ρ, should also be identical.

New tissue substitutes for 9 tissues have been evaluated, both theoretically and experimentally [1]. One of these substitutes, "total soft tissue", is close to the composition of the total soft tissue given by the ICRP [2]. The mass density is 1.03 g/ml and the composition in per cent by weight is: H(10.49%), C(23.33%), N (2.58%), O(62.80%), Na(0.11%), Mg(0.01%), P(0.13%), S(0.20%), Cl (0.13%), K(0.20%), Ca(0.02%). The ratios of μ/ρ, μ_{en}/ρ and S/ρ for the substitute to those of real tissue are equal to 1 in the energy range of 0.01-100 MeV, and the tissue is recommended as a suitable phantom liquid in which the organs could be submerged. Consequently, the "total soft tissue" liquid was used in these experiments.

Because of the low density of a human lung (0.25-0.30 g/ml in the median respiratory state) a liquid cannot be used as lung substitute. For these experiments, a preliminary substitute was prepared from two different types of porous granules, VERMICULITE and LECA, both produced from geological minerals. The grain sizes ranged from 0.3-3 mm and their mass densities were measured to be 0.10 and 0.40 g/ml, respectively. The two materials were mixed in

an approximately equal volume ratio giving a mass density of 0.26 g/ml. Unfortunately, the content of H, C, and N is negligible; a chemical analysis gave the following elemental composition: H (0.3%), O(46.7%), Na(0.68%), Mg(4.4%), Al(10.1%), Si(25.0%), K (2.9%), Ca(0.72%), Ti(0.54%), Fe(5.8%). The effective atomic number $\bar{Z}$ has been calculated as 13.8, which is a factor of 2 higher than for both soft tissue and air.

In the energy range of 0.1-2.0 MeV the ratio of μ/ρ, μ_{en}/ρ and S/ρ for the lung tissue substitute to those of soft tissue were calculated to be 1.10, 1.12 and 1.20, respectively. As the elemental composition of lung tissue + blood + air is similar to soft tissue [2], the absorption and scattering properties of the lung tissue substitute for photons and electrons are rather poor compared to real lung tissue, especially for the electrons.

3. EXPERIMENTAL WORK

An experimental programme for determining the radiation doses absorbed in target organs from radionuclides deposited in different source organs will be carried out with a newly constructed phantom [3]. To determine the importance of tissue substitute compositions in the phantom, experiments with both γ- and β-emitting nuclides have been made.

The γ-dose rates were measured in cylindrical "organs" submerged

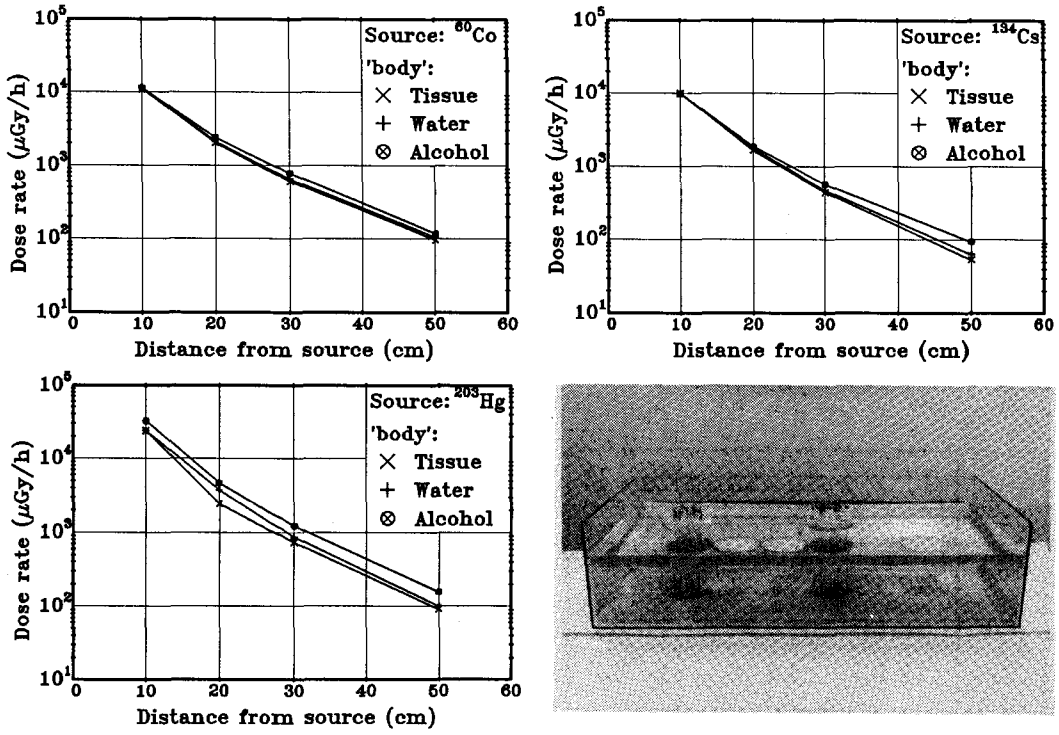

Figure 1. Measured γ-dose rates and the experimental setup.

in three different "body" liquids, namely alcohol, water, and the tissue equivalent liquid. An aquarium of dimensions 80 cm x 30 cm x 20 cm contained the liquid and represented a body. Two water filled cylinders with 10 cm diameter and 13 cm height represened a source and a target organ. The target cylinder was supplied with 13 vertically mounted tubes containing a total of 39 LiF TLD-700 dose meters. The centerline distance between the target and source could be varied from 10 to 50 cm. The target organ dose rates were represented by the average value of the readings from the 39 target dose meters.

Three different nuclides, ^{60}Co, ^{134}Cs, and ^{203}Hg, representing a photon energy range from 279 keV to 1332 keV were successively dissolved in the water in the source cylinder. The activities of the nuclides were chosen so that the dose rate was approximately 100 μGy/h at a source-target distance of 50 cm.

Figure 1 shows the results of the target dose rate measurements and the experimental set-up. As can be seen, the results for alcohol differ considerably, up to 80%, from those of the water and tissue for all three nuclides. At large source-target distances (50 cm) the difference between the target dose rate in water and in "total soft tissue" is 9% for ^{60}Co, 17% for ^{134}Cs, and 10% for ^{203}Hg.

Pure β-emitters ^{32}P ($E_{\beta max}$ = 1.71 MeV), ^{204}Tl ($E_{\beta max}$ = 0.763 MeV), and ^{35}S ($E_{\beta max}$ = 0.167 MeV) were successively dissolved in water, the soft tissue equivalent liquid, and the lung tissue granule. The lung granule was soaked with water containing the nuclide, and after drying the absorbed activity was distributed homogeneously in the granule.

The β-dose rates were measured in cubic "organs". Glass bowls of dimensions 12 cm x 10 cm x 8 cm were filled with the three organ materials. TL dose meters were placed in a special holder and covered with mylar foil of 0.8 mg/cm^2 thickness to avoid contamination of the dose meters. MgB_4O_7:Dy dose meters in which 3% graphite was imbedded were used, representing a nearly infinite thin dose meter, so that correction for attenuation in the dose meter was necessary only for ^{35}S [4]. The correction factor was 1.7. A correction for attenuation of β-particles in the mylar foil was also made for ^{35}S.

Nuclide	Dose rate (mGy/h * l)		
	Water	Tissue	Lung
^{35}S	27.13	24.21	146.71
^{204}Tl	16.52	14.92	79.28
^{35}P	19.02	18.02	116.32

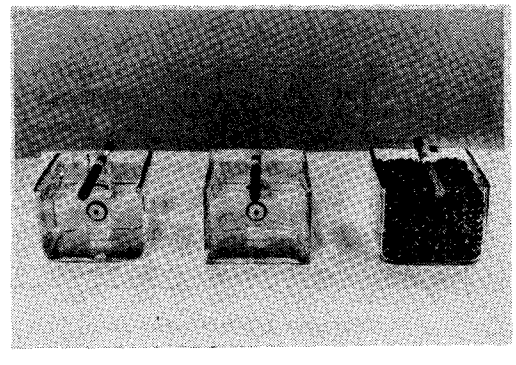

Figure 2. Measured β-dose rates and the experimental setup.

Figure 2 shows the result of the measurements and of the experimental set-up. The dose rate ratio $\dot{D}_{tissue}/\dot{D}_{water}$ normalised to source volume varied between 0.89 and 0.94, and the dose rate ratio $\dot{D}_{lung}/\dot{D}_{water}$ between 4.8 and 6.1.

4. DISCUSSION AND CONCLUSION

With 1 liter target and source "organs" submerged in a 50-liter "body" containing different liquids as substitutes for soft body tissue, γ-dose rate ratios $\dot{D}_{liquid}/\dot{D}_{water}$ were measured at different source-target distances up to 50 cm. For the total soft tissue equivalent liquid this ratio was measured at the largest distance to 0.91 (0.88) for ^{203}Hg, 0.85 (0.93) for ^{134}Cs and 0.92 (0.95) for ^{60}Co, respectively. For alcohol the ratio varied from 1.2 to 1.6. The figures in parentheses are the corresponding ratios calculated by the point kernel method. The measurements were carried out in two identical arrangements, and the difference between the two sets of results were in all situations less than 3%. Based on these measurements, it is concluded that for γ-radiation water is a rather good substitute for soft tissue, and that internal γ-doses for a given source-target geometry can be measured with an uncertainty less than 10% compared to a more correct soft tissue composition.

A homogeneously distributed β-emitting nuclide in a source organ will result in a β-dose rate to all the organ tissue that is proportional to the volume concentration (Bq/l) divided by the organ mass density if edging effects are neglected and the organ dimensions are greater than the β-particle range in the organ material. The β-dose rate ratio for two organs with the same nuclide concentration but with different densities will accordingly be equal to the reciprocal mass density ratio. The β-dose rate ratios $\dot{D}_{tissue}/\dot{D}_{water}$ and $\dot{D}_{lung}/\dot{D}_{water}$ were measured within 1-liter source organs containing soft tissue liquid, lung tissue granule and water to be 0.89 (0.93) and 5.4 (3.8) for ^{35}S, 0.91 (0.93) and 4.8 (3.8) for ^{204}Tl and 0.94 (0.93) and 6.1 (3.8) for ^{32}P, respectively. The figures in parentheses are the reciprocal mass density ratios. The measurements were repeated several times with small scattering between the results. The mass density for the soft tissue liquid was measured to be 1.08 g/ml, i.e. greater than the theoretical value of 1.03 g/ml, apparently due to an unnoticed evaporation loss of alcohol from the liquid during earlier measurements. It is concluded that water is also a good substitute for different tissue equivalent liquids for measurement of internal β-doses, and that mass density is the decisive parameter for the absorbed β-dose. The measurements on the lung granule show that factors other than the mass density may influence the absorbed β-dose. Further measurements on other granule materials with a more realistic elemental composition are required to determine these factors.

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Monitoring For Internal Contamination in the WAK-Reprocessing Plant

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By choosing suitable dosemeters individual monitoring of external radiation causes no problems. The risk of exposure due to internal radiation, however, seems to us by far more difficult to control, especially in the case of unexpected events.
Besides the general guideline of keeping any exposure as low as possible, routine monitoring for internal contamination is required by a German BMI recommendation /1/ whenever 5 % of the annual limits of intake (ALI) are supposed to be exceeded.

In the WAK ca. 500 workers (staff and contractors) with working permits for contaminated areas are to be observed using the monitoring program.

A. General Aspects

In a reprocessing plant with direct maintenance, detailed knowledge of the radiological working conditions is essential for specifying the adequate protective outfit. Any lack of information thereupon has to be taken into account by prophylactic means of protection.
During the over 15 years of experience since reprocessing started at the WAK, there has evolved a certain classification of rooms according to which a standard protective clothing and equipment has to be worn.

Under real working conditions room-ventilation has to be properly controlled. Advantage can be taken of the fact, that in the WAK air is generally conducted from rooms with a lower contamination potential to rooms with a higher contamination potential by pressure gradients up to 700 Pascal.

Surface contamination measurements are taken by direct monitoring, in the case of too high a radiation background by wipe-tests. After work has finished surface-contamination monitoring is compulsory. Warning alarms are defined at $3{,}7 \times 10^{-2}$ Bq/cm^2 for α-particles and $3{,}7 \times 10^{-1}$

Bq/cm^2 for β-particles. It has proven advantageous to retain one wipe test for nuclide specifc investigations in the lab. In the case of any accidental exposure a quick estimate of the total intake of radionuclides is obtained by taking only a body-counter measurement on a "leading" nuclide like Cs-137 and with that equating the proportions of the other nuclides according to your wipe test. Analysis of excreta taken after the accident corresponded well with the first estimates. Relevant nuclides to be considered are shown in table 1.

Actinides	Fission products
uranium (U-238,U-234,U-232)	caesium (Cs-137,Cs-134)
plutonium (Pu-238,Pu-239,Pu-240)	strontium (Sr-90)
americium (Am-241)	ruthenium (Ru-106)
curium (Cm-242, Cm-244)	tritium (H-3)

Table 1: Relevant nuclides for internal contamination monitoring

- In many areas of the plant this looks like a spectrum calculated by the ORIGEN-Code. An exception may be given by Ru-106, which often shows different behaviour.

- Detailed analysis of all process-streams in the WAK shows that there is a relatively strong correlation between Cs-137 and Sr-90. As Sr-90 has a great radiological importance and a radiochemical analysis of Sr-90 is not immediately available a conservative estimate of a Sr-90 intake is possible by taking intake Sr-90 $\leq$ 2x intake Cs-137.

- Because of the low annual limits of intake for Pu (160 Bq for Pu-239 for example), the Pu-isotopes have to be considered in any contaminating event as possible contaminants, even if concentrations of Pu are very low.

B) Air Monitoring Systems

Air monitoring is an important tool for routine incorporation control. In the WAK there are about 50 rooms with continuous air sampling installed. The filters are analysed daily for α- and β-aerosols. Long term air surveillance of rooms which could be entered without protective mask showed Pu-activity concentrations of less than 10^{-4} Bq/m^3.

In operating rooms where there is a potential hazard of being contaminated and the need of longer working times for personnel, there is an additional continuous air-monitoring by means of pseudo-coincidence techniques, semiconductor and plastic-scintillator detectors for α- resp. β-aerosols.

Pseudocoincidence monitors are the most sensitive monitoring devices installed. With implementation of high volume samplers (~ 40 m^3/h) and large-size proportional counters a sensitivity of $2x10^{-2}$ $Bqxh/m^3$ for α and $5x10^{-2}$ Bqxh/m3 for β-particles and better is achieved.

Yet the conditions of a very low background radiation and very well defined Radon/Thoron activity concentrations cannot be guaranteed by routine operating practice in all plant areas. In these areas α-monitors with semiconductor detectors give a better performance. Selective air monitoring for Kr-85 is done in the fuel element reception hall only.

H-3 is continously monitored in the chemical make-up working area. Recovered HNO_3 in use in this area generally shows HTO contamination.

Iodine-air surveillance is only necessary when work is carried out on the iodine-filter systems.

C) Individual routine monitoring for internal contamination

Fission Products:
The WAK-routine monitoring program involves two whole-body counting measurements per year. This guarantees a safe monitoring of the investigation level of 5 % intake for Cs-137 (ALI = $1,3x10^6$ Bq) as a "leading" nuclide.

H-3:
Individual monitoring of excreta is foreseen only for those workers who are working in the chemical make up area.

Actinides:

Routine urine excretion analysis is carried out for uranium and plutonium once a year. The sensitivity is about 1,5 mBq for Pu and 1 μg/l for uranium. If the results show any inconsistencies, there is a special follow up with further samples of excreta (urine+faeces) and additional analysis for americium and curium.

Despite the fact that urine excreta-analysis for transportable Pu has only limited sensitivity/2/, for example covering a 7-10 days surveillance period assuming a constant uptake on a 5 % ALI level and a detection limit of 1,5 mBq, you gain information on a much greater surveillance period by taking the samples equally spread over the year and thereby picking out representative candidates of each working group.

D) Special individual monitoring for accidental exposures

Criteria for this monitoring scheme are given by:

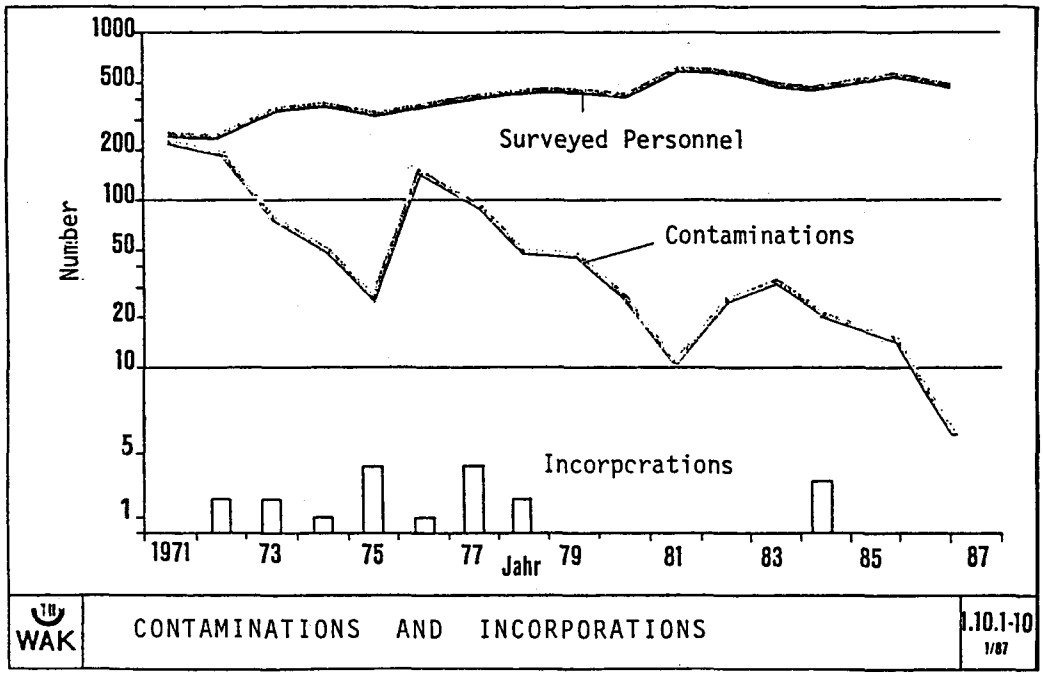
- skin contamination, especially face.
- damaged outfit.
- inconsistencies in routine monitoring especially values above detection limit
- any wound incurred in a contaminated area etc.

In the case of any of these criteria a series of special procedures is instigated.

- analysis of nose blows and throat swabs
- further analysis of excreta (urine+faeces)
- lung and whole-body monitoring
- wound detection
- analysis of blood samples.

In all these procedures the medical department is directly responsible.

The following foil shows statistics of incorporation events (> 5 % ALI) for the WAK.



E) Interpretation of results

From the data the total body-intake of radioactive material has to be defined.

Dose calculations are to be carried out according to the German BMI recommendations whenever 50 % ALI are exceeded. In general in the WAK a dose is recorded at intakes ≥ 10 % ALI. In nearly all dose calculations one can in fact take a class W solubility according to ICRP. Pu-oxide (class Y) can be ruled out for the majority of working areas in the plant.

Investigations of aerosol particle sizes in ventilated WAK rooms are in the 1 μm AMAD range, so the ICRP recommendations readily apply.

F) Conclusions

- WAK experience in monitoring for internal contamination shows that air monitoring is an important tool for routine incorporation control, especially in view of the difficulties in Cs-137 individual monitoring after the Tschernobyl incident.

- In any case of a positive monitoring for fission products Pu-isotopes had to be controlled too by excreta-analysis.

- Sampling of faeces is regarded as a non-routine measure, not only because of a psychologically motivated lack of acceptance, but also because there are indications that positive findings in faeces on a mBq detection level cannot be totally excluded in reference groups not engaged in the nuclear industry.

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CONSTRUCTION OF A HETEROGENEOUS PHANTOM FOR INTERNAL DOSIMETRY MEASUREMENTS

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1. INTRODUCTION

Intake of radionuclides could result both from routine work with radioactive materials and from accidents. For medical purposes, intakes of radionuclides are administered in both diagnostic and therapeutic treatments. Therefore, there is a need to evaluate the internal radiation doses received by the body organs from intakes. As measurements of internal doses are almost impossible to make in practice, internal dosimetry today is, to a large extent, based on calculations.

The current method of internal dose calculations is described in pamphlets from the MIRD (Medical Internal Radiation Dose) Committee [1]. The MIRD-method is based on calculations of the ratio of the energy absorbed by a target organ to that emitted by a source organ. The phantom used in these calculations - the Snyder-Fisher Phantom - is a heterogeneous phantom approximating the adult human body.

Some years ago the ICRP (International Commission on Radiological Protection) [2] published limits for the intake of radionuclides based on a method very similar to the MIRD-method.

Although there are limitations and uncertainties associated with internal dose calculations, little effort has been made so far to verify the calculations experimentally. Therefore, the present study has been undertaken with the goal to build a phantom more close to a "standard man" than the MIRD-phantom and use this phantom for measuring the Specific Effective Energy SEE(T←S) absorbed in selected target organs from radionuclides distributed homogeneously in specific source organs.

2. BASIC THEORY

One of the basic parameters in the field of radiological protection is the committed effective dose equivalent, H_{50}. To calculate H_{50}, it is necessary to determine the committed dose equivalents in a number of target organs T from the activity in a given source organ S, $H_{50}(T \leftarrow S)_i$, for radiation type i:

$$H_{50}(T \leftarrow S)_i = U_S \cdot SEE(T \leftarrow S)_i$$

where

U_S is the number of transformations of the radionuclide in the source organ S over a period of 50 years after the intake

$SEE(T \leftarrow S)_i$ is the specific effective energy in $MeV \cdot g^{-1}$ per

transformation for radiation type i (modified by the quality factor) absorbed in the target organ T from each transformation in the source organ S

$SEE(T \leftarrow S)_i$ is given by:

$$SEE(T \leftarrow S)_i = \frac{Y_i \cdot E_i \cdot AF(T \leftarrow S)_i \cdot Q_i}{M_T}$$

where:

Y_i is the yield of radiations of type i per transformation of the radionuclide

E_i is the energy of radiation i

$AF(T \leftarrow S)_i$ is the fraction of energy absorbed in the target organ T per emission of radiation in the source organ S

Q_i is the quality factor for radiation type i

M_T is the mass of the target organ

For an intake of a given radionuclide, a target organ can be irradiated by several source organs giving the committed dose equivalent to the target organ as:

$$H_{50,T} = \sum_S U_S \sum_i SEE(T \leftarrow S)_i$$

The value of U_S is the time integral over 50 years of activity of the radionuclide deposited in that organ:

$$U_S = \int_0^{50} q(t)dt$$

depending on both the biological retention and the physical half-life of the radionuclide.

3. CONSTRUCTION OF A HETEROGENOUS PHANTOM

Human organs from the body of a 180 cm, 75 kg male were used for the moulding process. The person did not die of a disease in any of the used organs, leaving these as reasonably representative of an adult male. The following organs were used:

- lungs
- kidneys
- liver
- thyroid
- pancreas
- spleen
- stomach
- bladder

All organs were saturated with a mixture of formalin and alcohol thereby keeping their original shape. The stomach and bladder were both filled with gypsum.

A negative form of each half of an organ was moulded by Sadocover 488, a 2-component polyethane surface coating material. A positive model of a half organ was moulded by filling the negative form with Sadocast 521, another 2-component polyethane casting material. After hardening, this second material is easily machined and polished to prepare a completely smooth surface. A heated sheet of 2-mm Acryl-plast material was placed at the top of the positive model, and with a vacuum technique a plastic shell identical in shape to the original half organ was produced. Because of the size of the trunk, head, and limbs it was necessary also to make negative vacuum forms of the casting material from the positive models.

The two plastic shell halves were then glued together leaving a hollow replica of the original organ. Several thin plastic tubes were inserted through each organ for the placement of TL dose meters. A hole in the side of the organ, closed with an O-ring tightened plug, makes it possible to fill the organ with liquid.

The positions of the organs were determined from anatomical atlases. The organs were fixed within the body in an easily replaceable way. Figure 1 shows different steps in the moulding process and Figure 2 the final phantom.

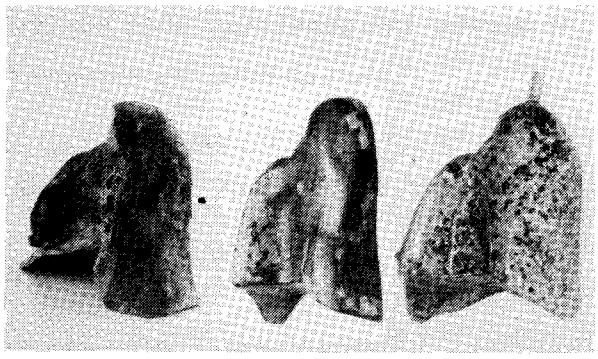

Figure 1. Moulding process for the production of a hollow replica of the right lung. From left to right: real lung, positive model of a half lung, and hollow replica of lung filled with granule.

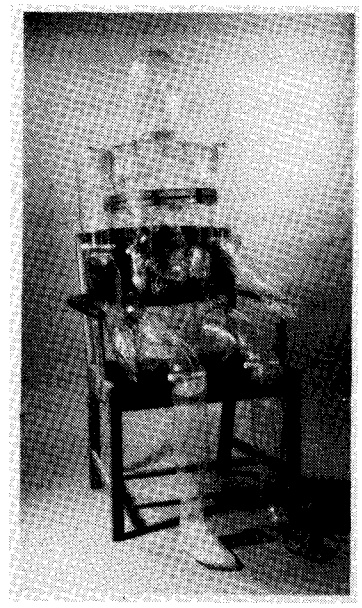

Figure 2. Phantom for internal dosimetry measurements.

4. EXPERIMENTAL PROGRAMME

An extensive experimental programme will be undertaken to determine the Specific Effective Energy SEE(T←S) absorbed in the target and source organs of the constructed phantom that will represent man more closely than the MIRD-phantom. Radionuclides of importance in health physics at nuclear installations and laboratories

as well as nuclides used in nuclear medicine will be used. The organs and the body will be filled with liquid except for the lung material which will be simulated by a granule [3]. The activity of the nuclide will be dissolved in the organ liquid and successively placed in all the source organs. For the lungs the granule will be soaked with liquid containing the radionuclide and then dried.

Two dosimetric methods can be used to measure the organ dose: small TL dose meters that measure the doses at several selected points within the organ or a volumetric dose meter with TL powder mixed in organic materials molding a solid dose meter with the same shape as that of the organ.

Both methods have their advantages and limitations. The measurement of dose at selected points within the organ is easily performed and gives the dose distribution in the organ directly. The total organ dose, however, has to be calculated from the point measurements. The volumetric dose meter gives the total organ dose directly, but not the dose distribution, although possible by measuring the response from sub-volumes of the organ. The difficulty here is to obtain a uniform mixture of TL powder and organic material and to recover all the TL powder from the mixture for the dose reading process.

In this study TL dose meters of the size 1 mm x 1 mm x 0.5 mm will be placed inside the thin plastic tubes passing through the organs to measure the γ-dose. The β-dose in the source organs will be determined in a separate arrangement where TL dose meters are submerged in the organ tissue liquid or granulate having linear dimensions large compared to the range of the β-particles [3].

The measured values of SEE(T←S) will be compared to the corresponding values in ICRP publication no. 30 [2].

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INFERENCES FROM DIRECT THORAX COUNTING ON OCCUPATIONAL WORKERS OF FUEL FABRICATION PLANTS

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Introduction: Indian fuel fabrication plants produce fuel assemblies for the country's nuclear power reactors(PHWRs&BWR). Uranium Oxide(UOP), Ceramic Fuel Fabrication(CFFP), Enriched Uranium Oxide(EUOP) and Enriched Fuel Fabrication(EFFP) are the four main production plants. Nat. and enriched U of enrichments 1.6-2.66% are handled in these plants.

In most of the areas of these fuel fabrication plants, despite rigorous controls, potential internal exposures through inhalations of (U_3O_8, UO_2) class Y aerosols, remain distinctly possible. Since periodic urine assay as the sole method of controlling internal intakes may not be adequate, direct thorax counting of fuel fabricators becomes obligatory. Fulfilling this obligation, we have monitored thorax burdens in a majority of occupational workers from these four fuel fabrication plants. This paper presents methodology, results, their analysis and atttempts to draw some notable inferences therefrom.

2. **Methodology**: It is based on direct detections of 63 & 93keV γ-rays from ^{234}Th assuming it in radioactive equilibrium with ^{238}U - a valid assumption for fuel fabricators and 185 keV γ-rays from ^{235}U. Two types of detectors-a 20cm dia x 3mm thick phoswich in centrally over supine chest geometry and 12.7cm diax 1.27cm thick NaI(Tl) in one over each lung configuration, housed inside a graded lined steel room shield[1], facilitate the measurements of low-energy photons(LEPs). Phoswich(10-150keV) and NaI(Tl)(20-300keV) measurements together provide indications of surface contamination and uranium enrichment as well.

The calibrations of the detection systems were performed with the aid of a REMCAL phantom. Effective soft tissue thickness concept is used to derive calibration factors for each subject separately. The spectral regions of 40-105keV for phoswich and 40-120keV and 165-215keV for NaI(Tl) are employed. Averaged value is reported as the subject's thorax burden.

3. **Results**: U-Thorax burdens in 128 occupational workers from four plants have been measured-about 1/3rd showing<5 mg, the rest in the 5-25mg range with only three exceeding it. The plots of assessed thorax burdens Vs. cumulative % probability on log-probability graphs are given in Figs.1 & 2 displaying data from Nat.& enriched Uranium plants respectively. Burdens exceeding 15mg are shown plotted in fig.3 against the length of occupational service being an index of inhalation period of the respetive worker.

4. **Discussion**

4.1 **Log-Normal Analysis**: Log-probability plots(fig.1&2) reveal that data upto 80-90%cumulative probability appear to fall on straight lines. Hence the thorax burden data from each plant are suitably describable by log-normal distribution. The internal exposure status of an occupa-

tional group from a particular plant would,thus,be signified by the geometric mean of thorax burden distribution.Calculated geometric means (GMs) & standard geometric deviations(SGDs)corresponding to the displayed distributions along with the total number of cases measured and the maximum burdens encountered are also given in the figures.

It is evident that UOP besides showing highest GM among the four plants shows SGD much greater than 2,implying perhaps an uncommon intake by some plant workers.For other plants,SGDs are fairly close to 2.

4.2**Comparisons with Lung Model Predictions:** The observed burdens from thorax countings,in fact,represent lungs+lymph nodes contents and therefore,reflect the cumulative effects of intakes over the period of occupational service. We,now examine their magnitudes vis-a-vis lung model predictions.Since the thorax burdens show log-normal distributions,their GMs are the proper parameters to be compared with.In Fig.3,lung model predicted build-ups of lungs & lungs+lymph nodes burdens as a function of exposure time are displayed.The results are for continuous inhalations at the rate of 1ALI/365/day of 1μm AMAD class Y aerosols of Nat. & 2%enriched uranium.Apparently,for chronic exposures,burden in lungs tends to saturate with time but that in lymph nodes continues to rise. To facilitate ready comparison,the values of GMs of the thorax burdens for four plants plotted against the mean service period of the monitored group are also shown in Fig.3.It is seen that all GMs fall well below the limits delineated by the lung model predictions.It implies that the working conditions in all plants are such that a majority of occupational workers would accumulate thorax burdens much less than those expected from an intake of 1ALI for each year of service.It should be noted that Fig.3 considers AMAD of 1μm whereas the average size of aerosols in fuel fabrication plants is 6μm. Nonetheless,this does not invalidate the conclusion reached as the lung model predicts almost identical values of the saturated burdens for chronic intakes of 1ALI/365/day irrespective of the particle size if same clearance half-time is assumed.

From the continuous intake model(Fig.3) & plant-wise estimated GMs of assessed U-thorax burdens,we deduced the projected average air concentrations as 8.8 & 7.6μg/m^3 Nat.U & 4.1 and 4.7μg/m^{3}enriched U in UOP,CFFP,EUOP & EFFP plants respectively. These estimates appear to be generally less than the five yearly averages of experimental measurements.Inevitable inference,thus,is that by and large generally satisfactory working conditions have prevailed in these fuel fabrication plants.

It is clear from Figs.1&2 that burden values lying above estimated GMs would need individual scrutiny too.Incidentally, more significant of these are data points lying beyond 80-84 cumulative % to which the assessed thorax burdens $\geq$15mg correspond.These are shown plotted in Fig.3.It is observed that except 3 data points from UOP,5 from EUOP/EFFP and 2 from CFFP appear to marginally exceed the delineated limits. However,due to in-built conservatisms in ALI and lung model,uncertainties inherent in the direct detection methods(max.$\pm$20%) and likelihood

of lower effective U-enrichment invivo,the points marginally exceeding the limits cannot be adjudged as over-exposures.Usually such subjects are referred for repeat monitoring after a certain interval .By any reckoning,3 data points from UOP,indeed indicate over-exposure.But it is plain from the foregoing discussion,that over-exposures are not expected from the working conditions and are necessarily attributable to poor working habits and possibly improper use of respirators.

4.3**Follow-up Measurements:** 3 follow-up measurements over a year have been made on one over-exposure case.The shape of pulse-height spectrum recorded from a NaI(Tl)detector positioned over right lung of this subject A is compared with the observed spectral shapes from a Nat.U source embedded inside Mix.D absorbers,identifying the internal contaminant as Nat.U beyond doubt(Fig.4).Worthnoting are the changes in LEP spectrum shapes occuring with different thicknesses of tissues over a Nat.U source.Invariably,such supplementary measurements aid spectral interpretations from contaminated subjects.

Extrapolating from the limited measurements on subject A,an usually long clearance half-life from thorax of about 970days was obtained. At least a few similar cases,have been reported from Canadian fuel fabrication plants[2].In addition,the phoswich response pattern over the frontal chest gave evidence of translocation of some Nat.U from lungs to lymph nodes-indicating a relatively old intake[1].Localised measurements with NaI(Tl)over skull and kidney sites of the subject with chest covered with 3mm Pb sheet,indicated no detectable Nat.U in these organs.These observations are subtly supportive of ICRP metabolic model of inhaled Y-Class uranium.

4.4**Detection Equipment Capabilities:**Accumulated thorax burdens of about 7.5 & 5.5mg Nat.U are expected from chronic (Fig.3) and acute intakes of 1ALI respectively at the end of one year. These are much above the currently achieveable MDAs (2-3mg marked in Fig.3) of the detection equipment employed.Therefore,the direct thorax counting techniques are capable of detecting intakes of atleast 1/2 ALI/yr in a yearly monitoring frequency.

5.**Conclusions:**Direct thorax counting conducted on a majority of occupational workers of fuel fabrication plants lead to these inferences: 1.About 2/3rd of the subjects showed measurable thorax burdens in 5-25mg range. 2. The assessed thorax burdens in each of the the four plants are log-normally distributed and their estimated GMs correspond to air activity concentrations of about 1/3 DACs-testifying satisfactory working conditions. 3. Comparisons with lung model predictions have identified 3 probable over-exposures. 4. Follow-up measurements on one case revealed unusually long clearance half-life,detectable translocations to lymph nodes but not to skull and kidneys. 5. With the available detection equipments, intakes of U at 1/2ALI level are measurable.

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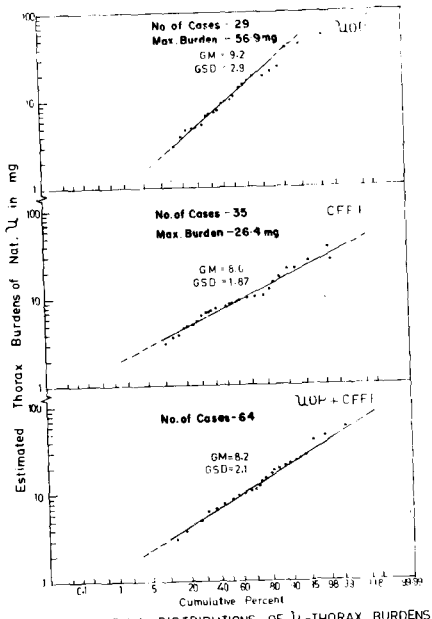

FIG. 1,- LOG-NORMAL DISTRIBUTIONS OF U-THORAX BURDENS FOR OCCUPATIONAL WORKERS FROM UOP, CFFP & BOTH PLANTS TAKEN TOGETHER.

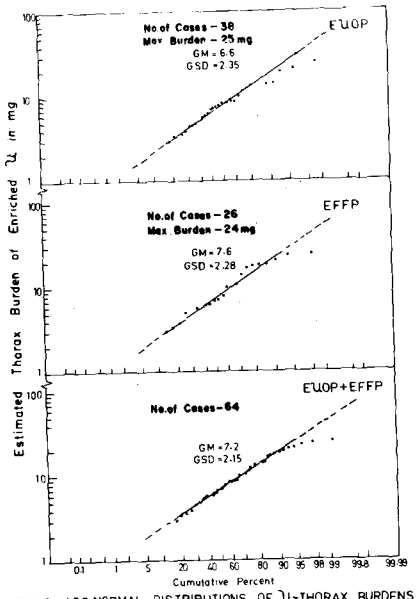

FIG. 2,- LOG-NORMAL DISTRIBUTIONS OF U-THORAX BURDENS FOR OCCUPATIONAL WORKERS FROM EUOP, EFFP AND BOTH PLANTS TAKEN TOGETHER.

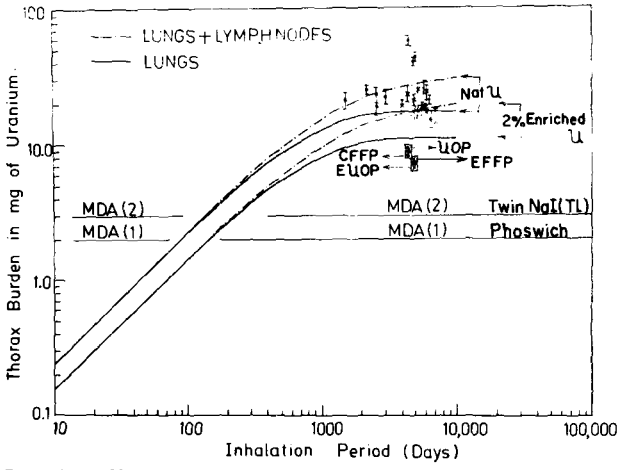

Fig. 3, Lung Model predicted buildups of lung and (lung + lymph node) burdens as functions of exposure time compared with observed thorax burdens from occupational workers. Model results are for continuous inhalation at the rate of 1 ALI / 365 / day of 1 μm AMAD class Y aerosols of Nat. and 2% Enriched uranium. Observed thorax burdens ≥ 15 mg for individual subjects are only displayed as two populations x — Enriched U; • — Nat. U together with the observed GMs from log normal analysis ALI 1500 Bq.

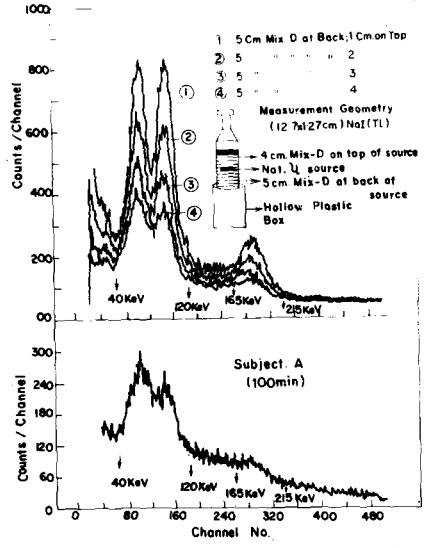

Fig. 4,- The shape of pulse-height spectrum recorded from a 12.7 cm. x 1.27 cm. NaI (TL) positioned over Right Lung of subject. A compared with the observed spectral shapes from a Nat. U source embedded inside Mix D absorbers.

ESTIMATION OF PLUTONIUM AT LOW LEVELS BY SOLID STATE NUCLEAR TRACK DETECTION (SSNTD) TECHNIQUE

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ABSTRACT

Integrating property of SSNTD, CR-39 in this case, was used in the measurement of low levels of plutonium in urine and liver tissue. The desired level of measurement was 1 mBq which corresponds to a daily stabilised urinary excretion of Pu-239 (for Y-class compound) following an intake of 1 ALI by inhalation.

Track development procedure consisted of chemical pre-etching (CPE) followed by electrochemical etching (ECE). Developed ECE spots were counted using a microfische card reader. Calibration experiments and measurements on spiked samples indicate that 0.5 mBq and 1 mBq of plutonium activity is easily measurable in urine and tissue samples respectively for exposure periods of 30 days. Possibilities about detection of still lower levels are discussed.

INTRODUCTION

With the discovery of SSNTD technique in 1958-59 (Young, 1958; Silk and Barnes, 1959) and proliferation of its applications from 1961 onwards, measurement of low levels of activity in a variety of mattrices has received a new impetus. Several such uses have already been reported (Fleischer, Price and Walker, 1975; Bhagwat & co-workers, 1976; Ellis, 1986; Jeanmaire and co-workers, 1986; Cecchi & co-workers, 1986) where alpha/fission track recording properties of SSNTDs have been made use of. In this paper results are reported which utilise high sensitivity of CR-39 as well as the advantages offered by ECE technique.

EXPERIMENTAL

Background Measurement and Calibration

Influence of background was minimised by i) choosing films with low background track densities and ii) reducing the film area under investigation. The film area in this study was restricted to 10 mm diameter because the smallest diameter of the electrodeposited (ED) sources which could be conveniently prepared was 6 mm.

Track development procedure for alpha tracks consisted of 6 hours' chemical pre-etching (CPE) at 60°C in 6N KOH followed by 8 hours' ECE at room temperature using the same etchant. The a.c. voltage applied across the film during ECE was 10 Vp/ μm of the film thickness at a frequency of 500 Hz. The CPE step also incorporates one advantage: it helps to reduce number of background ECE spots by converting sharp shallow surface defects (or pits) into blunt pits unsuitable for ECE.

Track detector pieces were exposed to ED sources in a way that eliminated contact of film with ambient air. This exposure arrangement is shown in fig.1. The activity used in the trial experiments was 0.5 mBq (= 0.03 dpm) of Pu-239 which, being lower than the desired estimation level by a factor of two, gave a corresponding safety margin. Table 1 shows the data from control and exposed CR-39 films obtained under different conditions. It demonstrates that the level of 0.5 mBq of activity is not difficult to assess for exposure periods of the order of 30 days.

In a separate calibration experiment, employing a stronger Pu-239 source, the efficiency of track registration for the given exposure and etching conditions was evaluated and found to be 21.4%.

Application of the Method

The technique was then applied to actual samples of urine and liver tissue which were spiked with plutonium activity. A radiochemical procedure was followed for the separation of plutonium from the given mattrices (Rudran, 1985). The ED sources prepared from spiked urine and tissue samples, along with their respective blanks, were exposed to CR-39 films for a period of 33 days. The results of this exposure are shown in Table 2. They clearly indicate that the levels of added activities of plutonium are easily measurable under the experimental conditions outlined. For tissue samples the plutonium activity added was 1 mBq instead of 0.5 mBq. This was due to the fact that initial blank runs with tissue controls had generally shown the background to be higher by about a factor of 1.5. A photograph of ECE spots in CR-39 film is shown in fig.2 which clearly brings out the difference in track densities of the exposed area and the surrounding background area.

CONCLUSIONS

i) These studies indicate that levels of 0.5 mBq in urine and 1.0 mBq in liver tissue can be conveniently estimated by the procedure outlined above.

ii) It should be possible to detect still lower levels of activity (a factor of 2 or more lower) by just increasing the film exposure times correspondingly.

iii) Further lowering of detection limits may be possible by controlling observed background in CR-39 films. This could be achieved by a detailed investigation of interplay of etching parameters used during CPE and ECE.

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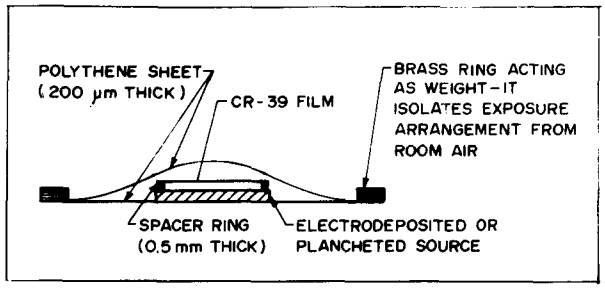

FIG.1, FILM EXPOSURE ARRANGEMENT

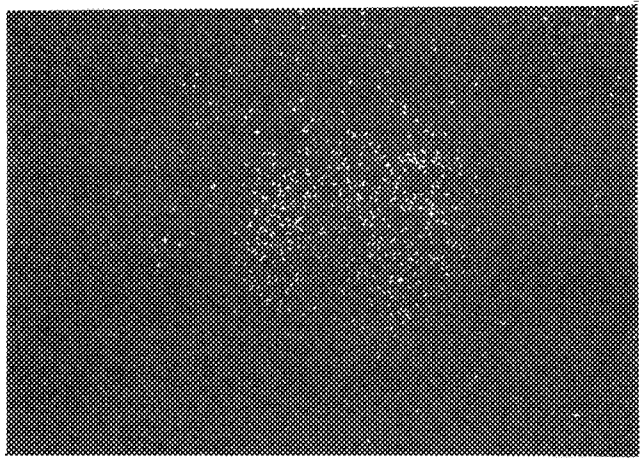

FIG.2 A PHOTOGRAPH OF ECE SPOTS (MAG = 6)

Table 1: Data from Control and Exposed CR-39 films obtained during initial trials (diameter of scanned area = 10.3 mm)

S.No.	Nature of Exposure	No. of ECE spots	Average No. of spots
1	Control film	58	
2	"	69	
3	"	84	
4	"	46	62.7 ± 11.8
5	"	65	
6	"	59	
8	Exposed to plancheted sources	302	
9	of 0.5 mBq activity, for 31.6 days	311	324.3 ± 31.2
10		360	
11	Exposed to electro-deposited sources	284	
12	prepared from solutions containing	280	
13	0.5 mBq of Pu-activity, for 31.6 days	334	294.5 ± 20.6
14		283	
15		300	
16		286	
17	Exposed to planchets, subjected to blank	92	90.5 ± 9.5
18	runs of electrodeposition, for 31.6 days	89	

Table 2: Data from spiked and liver tissue samples (Exposure duration = 33 days; diameter of scanned area = 10.3 mm)

S.No.	Sample Details (Matrix and activity)	Track Data		
		Recorded No.	Average No.	Net Average
1	500 ml urine control	87		
2	"	132	98.7±29.3	
3	"	77		
4	500 ml urine+Pu (0.5 mBq)	239	239.5±15.5	141 ± 33
5	"	240		
6	15 g liver tissue control	144	168.5±34.6	
7	"	193		
8	15 g liver tissue + Pu(1 mBq)	515	516.5±22.7	348 ± 41
9	"	518		

A COMPARISON BETWEEN A PHYSICAL AND A MATHEMATICAL MODEL USED FOR THE DETERMINATION OF THE COUNTING EFFICIENCY IN THE EVALUATION OF 239-PU AND 241-AM BURDEN IN LUNGS AND LIVER.

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INTRODUCTION

The determination of calibration factors for "in vivo" measurements of 239-Pu and 241-Am in order to assume a failure of the radionuclide in the considered organ, may produce some differences in the final evaluation.

A mathematical model, which permits to simulate the different distributions and to obtain the calibration factor for each one of them, can produce important estimations in these errors without the necessity of reproducing the different cases in physical models.

This report shows the comparison of the calibration factors obtained with the most advanced physical model, the Livermore phantom and a mathematical model based on the application of Monte Carlo method to the photon transport in a modified MIRD-V phantom.

The compared calibration factors include geometries with one or two phoswich detectors in lungs and liver as source organs.

MATHEMATICAL MODEL

The model used is based on the application of the Monte Carlo method to the photon transport in a MIRD-V heterogeneous anthropomorphous male phantom.

This phantom is a mathematical representation of the reference man, defined in the ICRP publication Nr. 23,and it includes three types of tissue which simulate bone,muscle and lung.

In this phantom it was introduced a chest modification which permits to change the soft tissue thickness around the lungs in order to obtain calibration factors for different chest wall thickness (CWT) and abdominal wall thickness (AWT).

This model is completed with a code which permits to modify the source system reproducing different kinds of distributions located in lungs and liver, and another one which considers the detector efficiency for the energies of the incident photon and generates a pulse height spectrum similar to the obtained by a multichannel analyzer.

PHYSICAL MODEL

It was used a Livermore phantom with homogeneous contamination of 241-Am in liver and, 239-Pu, 241-Am in lungs granted by IAEA within an intercalibration program under the agreement Nr.3698/R1/RB.

The measurement geometry of the first organ consists in one detector placed on the organ, while for the second two geometries were used: one detector on the sternum and a detector on each side of the trunk and tangent to the clavicle and sternum (standard geometry).

This work was carried out with chest plates which simulate 100% tissue muscle with CWT in the range of 2 to 4 cm for lungs and 1.8 to 4 for liver.

The measuring system is formed by two phoswich detectors of 127 mm diameter (2.3 mm of INa(Tl) and 50 mm ICs(Tl)) with an aluminium window of 0.076 mm thickness and it was completed by a pulse shape discrimination system.

RESULTS AND CONCLUSIONS

In figures 1, 2 , 3 and 4 are shown the calibration factors resulting from the different distributions of the radiactive material in different organs considered.

Due to these results, if we consider as real the calibration factors derived from the physical model, we observe that the diference between both models is always lower than 13%.

The short difference obtained between both models permit us to estimate the validity of the mathematical model to study other distributions, geometries, organs and nuclides avoiding to obtain them through experimental technical and physical models.

Besides, the flexibility of the code and the possibility of using different kinds of sources in each of their organs as well as changing the position and the type of detector used permit to use it in the study of any kind of located internal contamination to be detected by in vivo meassurement.

CWT [cm]	Efficiency [counts/photon]			
	Livermore Phantom		MIRD-V Phantom	
	17 Kev	60 Kev	17 Kev	60 Kev
2.18	2.37x10-3	1.95x10-2	2.23x10-3	1.94x10-2
2.78	2.06x10-3	1.75x10-2	2.04x10-3	1.82x10-2
3.24	1.82x10-3	1.57x10-2	1.76x10-3	1.65x10-2
3.98	1.52x10-3	1.32x10-2	1.57x10-3	1.48x10-2

Fig 1.Efficiencies for 241-Am in lungs.One phoswich on the sternum .

CWT [cm]	Efficiency [counts/photon]			
	Livermore Phantom		MIRD-V Phantom	
	17 Kev	60 Kev	17 Kev	60 Kev
2.18	5.41x10-3	4.18x10-2	4.76x10-3	4.70x10-2
2.78	4.48x10-3	3.69x10-2	3.94x10-3	4.03x10-2
3.24	3.78x10-3	3.15x10-2	3.35x10-3	3.50x10-2
3.98	3.02x10-3	2.63x10-2	2.69x10-3	2.94x10-2

Fig 2.Efficiencies for 241-Am in lungs.Two phoswich in standart geometry.

CWT [cm]	Efficiency [counts/photon]	
	Livermore Phantom	MIRD-V Phantom
	17 Kev	17 Kev
2.18	7.70x10-4	6.94x10-4
2.78	4.34x10-4	3.83x10-4
3.24	2.76x10-4	2.40x10-4
3.98	1.44x10-4	1.27x10-4

Fig 3.Efficincies for 239-Pu in lungs.Two phoswich in standart geometry.

CWT [cm]	Efficiency [counts/photon]			
	Livermore Phantom		MIRD-V Phantom	
	17 Kev	60 Kev	17 Kev	60 Kev
2.18	4.28x10-3	3.25x10-2	4.14x10-3	3.67x10-2
2.78	3.69x10-3	2.93x10-2	3.60x10-3	3.23x10-2
3.24	3.18x10-3	2.58x10-2	3.09x10-3	2.78x10-2
3.98	2.59x10-3	2.07x10-2	2.58x10-3	2.34x10-2

Fig 4.Efficiencies for 241-Am in liver.One phoswich on the liver.

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EVALUATION OF BODY CONTENT ^{90}Sr IN WORKERS USING EXCRETION ANALYSIS

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ABSTRACT

The review of the radiotoxicological control of the seven workers which were contaminated by radiostrontium in an accident is presented. The contamination occured in the laboratory condition during the work with the $^{90}SrCl_2$. These condition may be accepted as the single intake. It was estimated that the main way of the ^{90}Sr intake was the gastro-intestinial canal.

Evaluation of the internal contamination levels was found out by determination of the ^{90}Sr content in the 24^h urine samples, using the quantitative radiochemical methods in the programmed time intervals. From the excretion data of ^{90}Sr by urin, the initial body activity of radiostrontium Ao(Bq) where calculated and further the theoreticaly treated by extrapolation function for excretion and retention.

The satisfying correlation between the experimental and theoretically expected data (according to ICRP recommendation) was found.

THE BACKGROUND OF IN-VIVO MEASUREMENT AND THE SELF-COMPENSATION METHOD

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INTRODUCTION

The objective of this study is to solve the background problem in whole body counting(WBC), especially for these nuclides emitting low energy gamma rays. We have examined that where does various background in in-vivo measurement come from and how to eliminate them. The history of reduction and subtract methods for WBC background is developing gradually. At first screening the subject and the detector in a lead chamber, and subtracting the background of equipment, then subtracting the background of phanton, recently using control person and appropriate detector. Even if one carefully choose control person whose height, weight and potassium-40 content are similar to a subject, the results of subtraction is rarely satisfactory. The radiation irradiated from the naturally occurring radionuclides or contaminated source in the subject or control person is scattered by his own body and degraded its energy,but only a part of scattered radiation with same energy as the radiation being detected is taken as background. The reason is that human body is very complicat, same source strength does not mean it would have same amount of low energy scattering radiation, and any detector can not distinguish the rays with same energy but different sources. We have developed a new subtraction method taking a part of the in-vivo spectuum as a compensation term.

ANALYSIS

Suppose a subject with internal contamination of A_s Bq. radionuclide S emitting gamma ray with energy E_s. In a given WBC equipment the measured energy spectrum can represent as

$$A_S \cdot \eta \cdot F(E_S, E_X, B_i$$

where η is a transfer factor, the counts per minute due to 1 Bq. of radionuclide S, with similar distribution in body, at ideal condition, i.e. there is no any scattering by body tissue or orgin. F is characteristic function of energy spectrum, it is dependent on ray's initial energy E_S, emitting energy E_X and human body factors B_i, such as height, weight, muscle thickness, fat proportion, relative position of different tissues density and the distrabution of nuclide S in the body. The function always express as follows:

$$F(E_S, E_X, B_i) = \begin{cases} 0 & E_X > E_S \\ 1-\Delta & \text{when } E_X = E_S \\ \Delta_X(E_X) & E_X < E_S \end{cases} \qquad (1)$$

where Δ is the decrease fraction of the primary ray, while it emit out the body. Δ_X is the low energy gamma ray fraction, which is a succesive scattering spectrum and as a function of E_X.

If D is the measured nuclide, its gamma ray energy E_d, activity A_d, and S is a nature radionuclide, $E_S > E_d$. The energy E_d is detected in energy range E_1 to E_2. Then the counting rate M in this range is

$$M = \int_{E_1}^{E_2} (A_S \eta F_S + A_d \eta F_d) dE_X = \int_{E_1}^{E_2} A_S \eta \Delta_X dE_X + \int_{E_1}^{E_2} A_d \eta (1-\Delta_d) dE_X \qquad (2)$$

The first term in Eq.(2) is the background due to S, only the second term is just the demanded quantity.

The criterion for choosing control person is that whose S content approximates to the subject. That means the both terms:

$$\int_{E_S-\delta}^{E_S+\delta} A_S \eta F_S dE_X = \int_{Es-\delta}^{Es+\delta} A_S \eta (1-\Delta) dE_X \qquad (3)$$

are equal. Even if Δ greatly vary, the variation of $(1-\Delta)$ may be not obviously, while Δ is small, it is in general much less than 1. The variation of Δ_X is greater than that of Δ, and Δ_X is a factor in background term of Eq.(2). This is the reason why same A_S can not obtain similar background in the E_1 to E_2 range.

SELF-COMPENSATION METHOD

Beside the characteristic energy range E_1 to E_2 of nuclide D, we take another energy range E_3 to E_4, E_3 is larger than E_2 but much less than E_s. The counts in E_3 to E_4 range is:

$$\int_{E_3}^{E_4}(A_s \eta F_s + A_d \eta F_d)dE_x = \int_{E_3}^{E_4} A_s \eta \Delta_x dE_x \qquad (4)$$

This is quite like the background term of Eq.(2). If E_3, E_4 is very near E_1, E_2, the difference between Δ_x in these two range might be very small. It is reasonable to use Eq.(4) to compensate Eq.(2), then we have

$$M_c = \int_{E_1}^{E_2} A_s \eta \Delta_x dE_x + \int_{E_1}^{E_2} A_d \eta (1-\Delta_d) dE_x - K \int_{E_3}^{E_4} A_s \eta \Delta_x dE_x \qquad (5)$$

where K is the compensation constant, which can be determined in following method.

For non-contaminated person $A_d=0$, then Eq.(5) becomes

$$M_c^o = \int_{E_1}^{E_2} A_s \eta \Delta_x dE_x - K \int_{E_3}^{E_4} A_s \eta \Delta_x dE_x \equiv 0 \qquad (6)$$

These two integrals are measurable, so that the K is solved. In practice, K is statistically determined by a lot of ordinary persons.

APPLICATION

We have measured lead-210 in skull for five tin miners and nine ordinary non-contaminated men. The spectrum data are treated by four methods for comparison.

(1) Counts in the range of E_1 to E_2 of miner is subtract by counts in same range of control person.

(2) Same as (1) but only take the counts in the peak area.

(3) Constant slope method: the slope is determined by $N(E_3)$ and $N(E_4)$ in the spectrum, then only the upside counts in range of E_1 to E_2 are taken into account.

(4) The self-compensation method as mentioned above.

The results of ordinary men are given in Table 1. The perceptibility is more or less like the concept of minimum detectable activity but with a definite coefficient. The miners' data are given in Table 2, the values in Table 2 are in unit of perceptibility. The data in these two tables show that the proposed method gets best effect.

Table 1 Ordinary men's data (in counts per 30 minutes)

		Method (1)	(2)	(3)	(4)
Mean value	$\bar{N}$	217	-3.9	-920.6	0
Standard deviation	s	20.25	18.99	661.5	18.74
Perceptibility: Definition		$2\sqrt{2}\,\sigma$	$2\sqrt{2}\,\sigma$	2σ	2σ
Perceptibility: Value		57.3	53.7	1223.	37.5
Perceptibility: Equivalent value*	(Bq)	6.1	5.7	129.6	4.0

*This value is equivalent to that obtained at following condition, two detectors, measuring time of 120 minutes with a calibration factor 9 Bq./cpm.

Table 2 Miners' data (in perceptibility)

Number	Total measuring time (min.)	Analysis method (1)	(2)	(3)	(4)
M_1	60	1.5	1.3	-1.3	2.8
	345	2.9	3.6	-2.1	6.5
M_2	155	0.3	-0.4	-1.5	0.1
M_3	60	-0.3	0.2	-0.3	-0.2
	150	0.9	0.7	-0.7	0.4
M_4	150	-0.5	0.8	-1.9	0.4
M_5	60	0.1	0.7	-0.7	1.1
	150	1.2	0.9	-1.4	1.9

RADIOLOGICAL ASSESSMENT OF THORIUM WORKERS

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ABSTRACT

Brazilian soil is a major depository of Natural Thorium. It is extracted from monazite sand and is concentrated in a mill located in São Paulo, the largest industrial city in Brazil. Workers in this mill have been employed for times up to 33 years. We have monitored these workers through urine, feces and whole body counting bioassay techniques. We have developed a technique for "in vitro" monitoring in which it is possible to analyse both uranium and thorium, since monazite sand has a percentage of uranium. "In vivo" whole body monitoring is accomplished by analysing both Tl-208 and Ac-228. "In vivo" and "in vitro" measurements are compared.

The thorium concentrate produced in this mill is sold to several industries, including a gas lamps factory. We have monitored works from this industry using the same techniques described above.

Results from monitoring performed in the mill and in the industry will be shown. A comparison giving emphasis to the employing time is made.

SALIVA AS A PRACTICAL PARAMETER FOR ESTIMATING RADIATION DOSE FROM INTERNAL CONTAMINATIONS

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The main methods for estimating radiation dose, received by the body and its component parts from radionuclides taken into the body, are the external detection method by means of a total-body counter and the internal detection method by measurements of radionuclides in excreta (urine, faeces, breath, nasal discharge, sputum, saliva, sweat and blood).

In analysing biological materials for the assessment of internal contamination, the ICRP (International Commission on Radiological Protection) recommends, mainly for practical reasons, the use of urine as the most relevant parameter, because of the relationship between the urine content en extracellular fluid content.

With respect to this relationship, there may be a reasonably good correlation between the concentration of the radionuclide in urine, blood and saliva. In addition, saliva is relatively easily collected and therefore of practical importance.

In the light of the above mentioned, a preliminary study was conducted to investigate the suitability of saliva as a parameter for estimating radiation dose.

Rats were i.v. injected with 150 μCi sodium (^{3}H)-acetate. Samples of blood, urine and saliva were collected and measured for radioactivity, twice daily, during three weeks. No significant differences in fractional excretion curves were evident when comparing the date of blood, urine and saliva.

In conclusion, the preliminary results, obtained with sodium (^{3}H)-acetate, suggest that saliva may be an excellent parameter, because of its practical availability at any time, and the close correlation with the recommended standard method of the ICRP using urine. Further investigations on this subject are in progress.

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RADIATION PROTECTION PRACTICE

VOLUME III

Organised by the Australian Radiation Protection Society

FURTHER APPROACHES TO BIOLOGICAL INDICATORS OF RADIATION INJURY

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Despite of the decades-long investigations, the search for proper biological indicator of radiation injuries did not result in techniques fulfilling all the requirements as yet (6). So far, the most reliable assay is the dicentric chromosome aberration analysis (9,10). New developments have been made recently on a cytogenetic technique, i.e. the micronucleus assay, and for local injuries on the application of thermography. Our own results as presented below suggest that these methods are promising.

MICRONUCLEI IN X-IRRADIATED HUMAN LYMPHOCYTES

The detection of micronuclei in lymphocytes has come into the limelight a few years ago. The fairly consequent dose-effect relationships obtained by several laboratories made the technique rather promising (ref. 1), especially after the modification recommended to recognize interphase cells after their first mitotic divisions (2,3,11).

On Fig. 1. our data are shown on dose-response relationships of appearances of micronuclei in human lymphocytes from x-irradiated whole blood as detected in traditional lymphoblasts and cytokinesis-blocked ones.

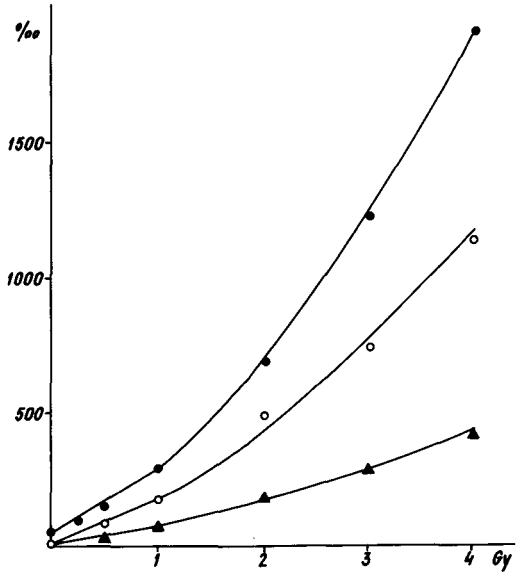

Fig.1. Dose-effect curves of appearances of micronuclei after x-irradiation of human blood in mononuclear lymphoblasts (▲), binuclear cytokinesis-blocked cells (o) both after 72 hrs and the frequency of acentric fragments (●) after 48 hours incubation.
The points represent data obtained from 3 donors. Cytokinesis-blocking was performed by 3 μg per ml cytochalasin B through the last 28 hrs of incubation.

In addition, for comparison, the yield of radiation-induced acentric fragments is shown as well. The data from 3 donors per test and per dose could be best fitted to linear-quadratic relationship. The yield-equations for micronuclei in mononuclear lymphoblasts (MN) and for citokinesis-blocked (CB) cells are as follows:

$$Y_{MN} = (6.9\pm2.8) + (38.9\pm5.8)D + (16.3\pm2.1)D^2$$

$$Y_{CB} = (11.3\pm4.3) + (139.0\pm44.0)D + (36.8\pm18.6)D^2$$

When comparing the three curves on Fig. 1. it is obvious that in CB cells more micronuclei can be detected by appr. 40-50 per cent than in mononuclear lymphoblasts but the frequency of acentric fragments is still higher.

Based on the yield equations presented, if the "c" value is properly low and the frequency of micronuclei in CB-cells is at least double of the background level, an estimate of a dose of appr. 0.1 Gy can be made. The "c" value, however, is somewhat dependent on age (1,3,4) which fact has to be considered in dose assessment through micronucleus assay. Therefore, further collection and comparison of background data by various laboratories will help the wide acceptance of this technique as a reliable biological indicator or even dosimeter of radiation injury.

It has to be noted, however, that further studies are needed also to reveal the origin and mechanism of formation of micronuclei appeared in mononuclear and CB lymphoblasts that the size distribution in the latter is wider, i.e. a push toward the larger diameters were found (5).

THERMOGRAPHY FOR LOCAL INJURIES

In cases of partial body irradiation including extremities the usual tests applied most commonly have only limited values in assessing the severity of local tissue damages. For such situations a few laboratories including our one have introduced various techniques of thermography. We have reported on a clinical case when an industrial radiographer has suffered an accident with Ir-192 gamma radiation source (7). Especially three fingers of his left hand were seriously irradiated. Although clinical symptoms were restricted to the distal parts of the fingers, by contact thermography (Flexi Therm^R) it was indicated that larger areas of the thenar and hypothenar were also involved. Telethermogrammetry has indicated even larger areas involved (8). The case raised several problems on detectability of dose- and time-dependent tissue damages. Investigations on mini-pigs revealed thermographically detectable alterations even in phases without clinical symptoms. The Figs. 2. and 3. present data obtained after x- and gamma-irradiations of thighs of mini-pigs in the dose range between 12.5 and 75 Gy.

We could conclude from our clinical and experimental experience collected so far that thermography is a sensitive and useful technique to detect local radiation injuries also in the phases when clinical symptoms are not evident, i.e. in the early and latent periods.

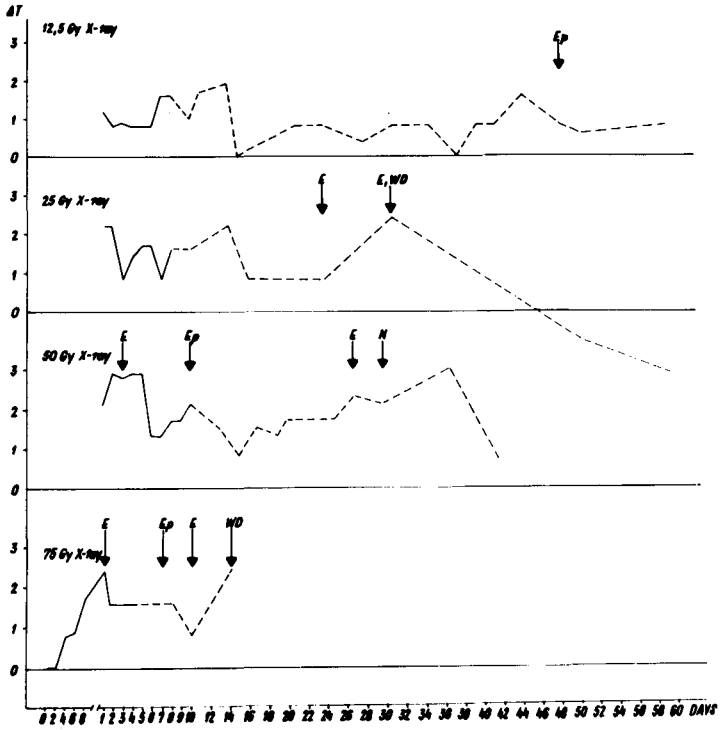

Fig.2.
Thermal alterations in time (hours then days) on surfaces of mini-pig thighs after x-irradiation with 12.5, 25,50 and 75 Gy as detected by NovaTherm^{R} contact thermography. The ▲T expresses differences compared to the temperature of unirradiated surrounding areas. Letters and arrows indicate time-points when clinical signs appeared, i.e. Ep: epilation, E:erythema, WD: wet desquamation, N: necrosis.

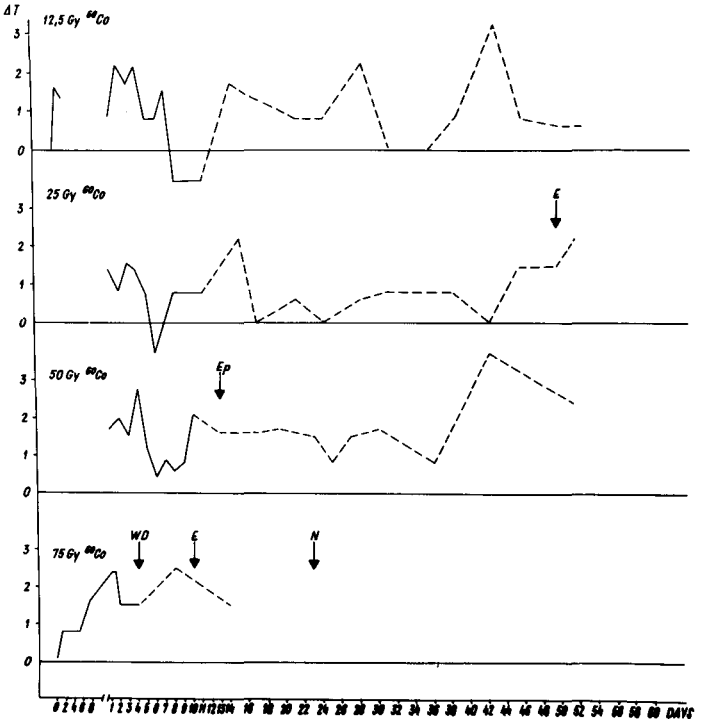

Fig.3.
Thermal alterations in time (hours then days) on surfaces of mini-pig thighs after ^{60}Co-gamma-irradiation with 12.5, 25, 50 and 75 Gy as detected by NovaTherm^{R} contact thermography. Abbreviations as on Fig.2.

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BREAST CANCER RISK IN SCOLIOTIC WOMEN EXPOSED TO MULTIPLE DIAGNOSTIC X-RAYS

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ABSTRACT

Although exposure to ionizing radiation is a well-known risk factor for development of breast cancer, the potential risks from low-dose, fractionated diagnostic exposures during the period of breast development have not been thoroughly evaluated. Women with scoliosis may be at increased risk of breast cancer as they are exposed to multiple diagnostic x-rays during childhood and adolescence, when the breast may be highly sensitive to the effects of radiation. To evaluate these risks, the authors conducted a historical cohort study of 1,036 women treated for scoliosis at four Minneapolis-area hospitals between 1925 through 1965. Medical records were abstracted for information on treatment and number and types of x-ray films taken. Breast doses were reconstructed on the basis of machine settings, tube type, and other relevant variables. Ninety percent of the group was located, and over 88% responded to a mail questionnaire or telephone interview. Eleven cases of breast cancer were observed compared with less than 6 expected (O/E=1.84, 95%CI=0.9, 3.3). The O/E ratios stratified by time since first exposure (<15 years, 15-29 years, 30+ years) were 0, 1.6, and 2.4, respectively. The rates of breast cancer per 10^3 exposed women by number of x-ray films taken (a preliminary surrogate for radiation dose) were 0, 5.8, 12.9, and 20.7 (for 0, 1-29, 30-59 and 60+ films, respectively). The potential effects of concomitant breast cancer risk factors are being evaluated. The data suggest that exposure to low level diagnostic radiation during childhood or adolescence may increase the risk of breast cancer.

EVOLUTION OF AN ACUTE RADIATION SYNDROME IN A CRITICALITY ACCIDENT

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INTRODUCTION

On September 23 rd., 1983, during the assembling of the Ra-2 critical facility, located at the Constituyentes Atomic Center, Buenos Aires, Argentina, an uncontrolled increase of power occurred, causing the operator's death.
The RA-2 is a critical facility with a nominal power of 0.1 W. It has been under operation since 1966 for experimental purposes and for trainning courses. It works with high enriched uranium plate fuel and with light water as a moderator and as a reflector. The reactor pool is cylindrical with a diameter of 2 m and a height of 1.5 m. To be assembled, the reactor core can be reached through an open upper surface. The reactor shielding consists of rectangular blocks placed in such a way that the radiation field to the surrounding laboratories is limited.

The accident occurred when the operator was trying to change the core configuration, resting on the edge of the reactor pool (1). According to the operation procedure, it was necessary to fully evacuate the moderator, but this was, in fact, only partially done. During the core configuration change, a transient reactivity occurred. Under these circumstances, the operator, who was the only person present in the facility, resulted highly overexposed. Some other people located in the control room and in contiguous laboratories were exposed at a much lower level.

PRIMARY MEDICAL INTERVENTION

All the people were exposed to a field of gamma and neutrons beams liberated in a very short time.
Initially, measurements to detect personal external and internal radioactive contamination, as well as measurements of body induced activity of Na-24 and hair induced activity of P-32 and activation of personal elements were done. In the meantime, the dosimetric system, consisting of personal dosemeters and dosemeters for criticity accidents located in the area, was processed.

The medical evaluation was clinical, haematological and cytogenetic, by chromosome aberration analysis.
Thirty minutes after the overexposition, the operator had nausea, vomiting and diarrhea. When examined, prostration and facial and right hand erythema were observed. Two hours later, lymphocytes count had dropped to 46 % of normal values. All these events made it possible to diagnose Acute Radiation Syndrome in severe evolution. The initial clinical and dosimetric results on the other persons involved made it possible to estimate whole body doses below 1 Gy, values that were confirmed later (2). None of

them showed located injuries.

PLANIFICATION OF THE MEDICAL ASSISTANCE

According to the expected risks, the following decisions were taken to treat the overexposed persons:
1- Hospitalization of the operator in Prodromal Stage of Acute Radiation Syndrome, with skin injuries and
2- Ambulatory control of the remaining personnel without clinical, haematological and ,local manifestations.

CLINICAL EVOLUTION OF THE HOSPITALIZED PATIENT

The chronometry of the events that were observed during 48 hours and 25 minutes to the lethal evolution is as follow:
During the next two hours following the overexposure, it was observed after 15 minutes, gastrointestinal and neuromuscular symptoms of the Prodromal Stage, different from the manifestations of the psychologic reactive phase of the Adaptation General Syndrome, that in its primary phase mimethasies it (3). It started at 15 minutes with painful vomiting that did not respond to medication. During this period of time skin-mucous located non-stochastic effects like conjunctional hyperaemia and erythema on the right hand were detected. The pharynx's pain could be related with the oedema observed some hours later.
Between hours 2 and 26, the evolution was apparently calm without clinical manifestations. However, during this period the haematopoyetic harm increased; lymphocytes were less than 100 per microliter and neutrophils were more than 12000 per microliter, the latter indicating reactive alarm against the accidental stress. The important decrease in lymphocytes, as well as the possible harm to the cell membranes of the surviving ones, prevented the estimation of dose by chromosome analysis. During this period the skin-mucous injuries on the pharynx, forearm,and right hand increased. Besides, erythema on the right side of the chest appeared.
After 26 hours, the patient was in oliguria and arterial hypotension, without changes in ECG; 25000 neutrophils per microliter were counted, LDH, SGOT and SGPT were elevated.
After 29 hours, the patient had a sanguinolent vomiting. The platelet count was not too low (100000/microliter).
Between 32 and 38 hours, the patient became anxious and excited. He had 44 breaths/minute and his respiratory difficulty increased. Laboratory tests indicated metabolic acidosis. The chest film showed diffuse haziness on the right lung. Physical dosimetry estimated a dose by neutrons on the right lung of 20 Gy and on the left one of 6 Gy (4).
After 48 hours, the patient did not respond to painful stimulus and did not display pupillary light reactions. Convulsions started, this was the onset of coma, quick evidence of the Crisis Stage of the Neurological Form of the Acute Radiation Syndrome.
The ECG presented a broadening of the QRS complex, specially at the expense of the repolarization component, possibly associated with the destruction of the active transport mechanisms in the

myocardium fibers.
Three cardiopulmonary arrests followed, one after the other, during the last one he died. The necropsy at brain level showed vascular injuries, intravascular coagulation, demyelination and nervous cell destruction.
The distribution of the damage caused upon the tissues and its clinical manifestations is consistent with the estimated doses in the reconstruction done by physical dosimetry, characterized by a craniocaudal decreasing of dose with prevalence upon the right side of the body.
Death occurred after 48 hours and 25 minutes of evolution of the Acute Radiation Syndrome, with superposition of clinical manifestations of the Cardiovascular and Neurological Forms including radiopneumonitis in the right lung.
The reconstruction of physical dosimetry estimated a whole body dose of 43 Gy, 21 Gy by gamma and 22 Gy by neutrons (4).

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THE EXPERIMENTAL STUDY OF COMBINED EFFECTS OF IONIZING RADIATION, ALCOHOL AND TOBACCO SMOKE ON WARM BLOODED ANIMALS

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ABSTRACT

Results are presented of experimental study of socially significant (life span, reproductive function, working capacity) and medico-biological (immunological, biochemical, genetical, etc.) indices under the influence of ionizing radiation, alcohol and tobacco smoke.

The experiments were performed on 7000 white non-linear mice of both sexes. The effects of strontium-90 and polonium-210 differing in their radiation quality and metabolism in a warm blooded organizm have been studied. The radionuclides were given with drinking water daily over the animal lifetime both separately and in combination with alcohol and tobacco smoke. The duirnal amount of alcohol simulated man's daily consumption of 2.5 g pure ethanol per 1 kg body weight, and the action of tobacco smoke corresponded to daily smoking of 20 cigarettes.

The mean lifetime shortening and the increased tumours yields observed in the experiments were found to be caused principally by the chronic action of ionizing radiation. But at the doses above 10 Gy the three factors combined caused the intensive metastatic spreading.

The gonadal changes and disturbances of the reproductive function in males were observed under the action of all factors in question, but in females the most manifestative deviations of their reproductive capacity from the control level were noted under the combined action of ionizing radiation, alcohol and tobacco smoke. The combination of these detrimental factors revealed summation and synergism of effects for humoral nonspecific protection and some immunological reactions.

EFFECT OF DTPA $CaNa_3$ ON THE DISTRIBUTION OF Ce-144 IN THE SOLUBLE CYTOPLASMATIC FRACTION OF ISOLATED HEPATOCYTES

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INTRODUCTION

Numerous animal experiments and therapeutics applications have demostrated the usefulness of DTPA $CaNa_3$ to increase excretion of actinides and lanthanides in cases of internal contamination.

Because of the high affinaty of these radionuclides for the liver, hepatocytes in primary culture was chosen as a simple model system.

The purpose of this work was to study the distribution of Ce-144 between compounds of high and low molecular weight of the soluble cytoplasmatic fraction obtained from isolated hepatocytes after they were incubated with and without DTPA $CaNa_3$, since the effectiveness of this chelating agent to remove contaminants is larger while they remain in this fraction.

MATERIALS AND METHODS

The methodology consisted in the intraperitoneal contamination of Wistar rats with Cl_3 Ce-144. Prior to sacrifice, 48 h later, the liver was perfused "in situ" with modified Hank's buffer supplemented with collagenase 0.05 % and then removed and digested in a volume of the same complete buffer.

The suspension was centrifuged and the cells resuspended in Waymouth's MB 752/1 medium supplemented with 10 % Fetal bovine serum and antibiotics.

Aliquots of the final suspension were incubated in a humidified 5 % CO_2 air incubator at 37°C during 6 h. The chelator DTPA $CaNa_3$ was added at a final concentration of 5×10^{-3}M, prior incubation. A corresponding control was processed.

The suspension was removed, centrifuged and the cells were resuspended in TRIS 5 mM - Sucrosa 0.25 M and ruptured in a Dounce homogenizator. The soluble cytoplasmatic fraction was obtained by centrifugation at 105,000 x g for 80 min. The supernatant was examined by gel permeation chromatography through ULTROGEL ACA-22 columns (1.6 x 80 cm) using 0.1M TRIS-ClH, pH 8.0, containing 0.1M ClNa and 0.02 % sodium azide as eluting buffer. The column was calibrated using horse spleen ferritin and human transferrin.

Fractions of 4 cm^3 were collected at a flow rate of 0.15 cm^3/min. The eluate was monitored spectrophotometrically at 254 nm and each fraction was counted for radioactivity by a multichannel, well-crystal, scintillation counter.

RESULTS

Two peaks with Ce-144 activity were observed with the hepatocytes with DTPA $CaNa_3$ (fig 1) and with the controls (fig 2). One of the peaks was coincident with the elution volume of ferritine (MW 700,000 - 800,000) and the other was located in the lower limit of exlusion of the gel (fractionation range 100,000 - 1,200,000). The chromatographic profile for a preparation of Ce^{144}-DTPA complex is shown in figure 3. The ratio between Ce-144 activity in high and low molecular weight fractions is 0.08 for the sample incubated with DTPA $CaNa_3$ and 3.60 for controls.

DISCUSSION

In the treatment of actinides and lanthanides contamination with DTPA, the amount of radionuclides which can be removed from the liver decrease with the length of time wich elapses between contamination and the start of treatment.

This decreasing efficiency of DTPA probably reflects the transfer of nuclide from the soluble fraction to the formed elements of the cell becoming less accesible for reaction with the chelating agent.

Our previous experiments showed that Ce-144 content in the rat liver reached a maximun at 2 days P.I. and that the fraction of Ce-144 in the cytosol was larger in hepatocytes incubated with DTPA.

The purpose of the present work was to investigate the distribution of Ce-144 amongst proteins of the liver cytosol with and without DTPA using gel chromatography on Ultrogel ACA-22.

At 48 hours after injection the major cerium-bearing peak was in the region of the ferritin elution volume. In the presence of DTPA the pattern of isotope destribution had changed. The proportion of cerium in the low molecular weight region had increased and in the ferritin region had declined.

According the hypothesis of Bhattacharyya et al, DTPA may be taken up by the parenchymal cells of the liver by pinocytosis of extracellular fluid. Fusion of pinocytotic vesicles with lysosome containing Pu-239 could result in direct chemical contact between the retained DTPA and hepatically deposited Pu. Excretion of Pu-DTPA complex into bile was demonstrated.

The observed differences in the distribution of cerium by chelate action might be explained by this hypothesis but we can not assert yet that the increase is due to the Ce-DTPA complex.

Clearly more experiments need to be done to elucidate the finer details of the mechanism for DTPA-induced removal of lanthanides from the liver rat.

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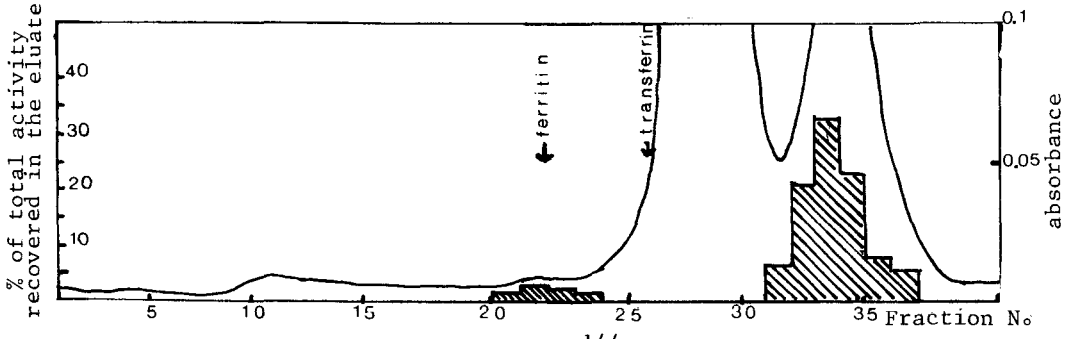

FIG. 1: UV absorbance at 254 nm and Ce^{144} γ - radioactivity profiles obtained by chromatography on ULTROGEL AcA 22 of the soluble fraction of hepatocytes incubated with DTPA Ca Na_3

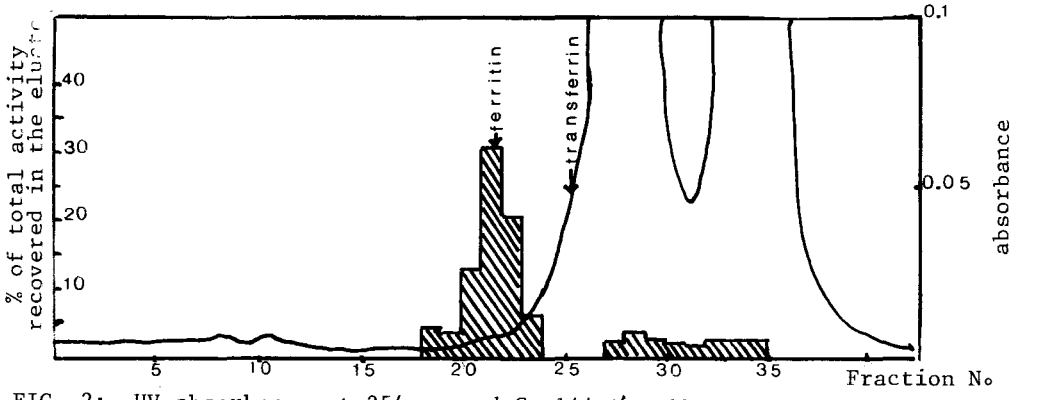

FIG. 2: UV absorbance at 254 nm and Ce-144 γ radioactivity profiles obtained by chromatography on ULTROGEL AcA 22 of the soluble fraction of hepatocytes

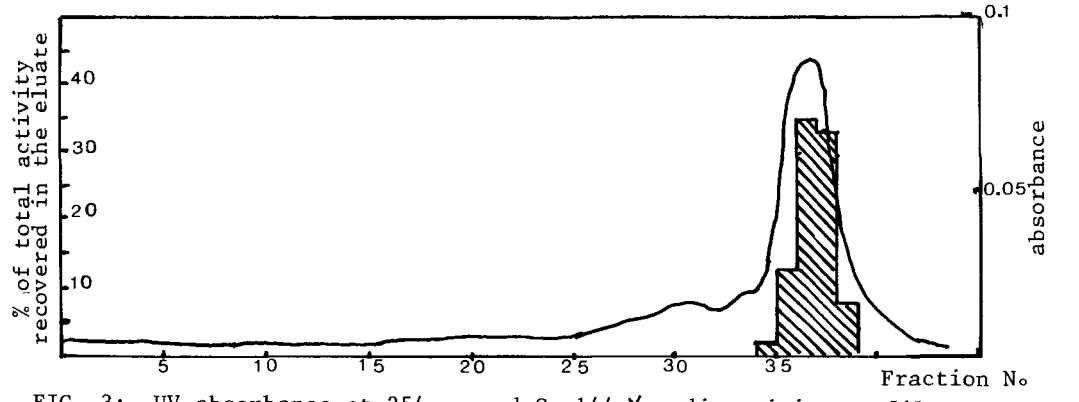

FIG. 3: UV absorbance at 254 nm and Ce-144 γ radioactivity profiles obtained by chromatography on ULTROGEL AcA 22 of the complex Ce^{144} - DTPA

BIOLOGICAL DOSIMETRY IN AUSTRALIA

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The accuracy of biological dosimetry (B.D.), by enumerating dicentric chromosome aberrations in peripheral blood lymphocytes, has been improved over the period 1969-1986 by increasing the number of cells examined from 300 to 500. The use of the bromodeoxyuridine technique to score cells only in first division has not, however, contributed to improved accuracy as in our laboratory metaphases in second division do not exceed 10% at 48 hours. Routine practice is to score cells immediately while awaiting the ageing and staining of the bromodeoxyuridine slides.

Blood samples from 74 subjects involved in 45 real or suspected radiation accidents were examined for B.D. (Table 1). No dicentrics (D) were observed in 70 subjects suggesting that if an exposure did take place it was less than the lower limits of detection of 0.1 Sv for acute uniform whole body low LET radiation ("zero" dose estimate). This included four subjects where a physical reconstruction of the accident indicated exposures exceeding 1 Sv and three others where film badge readings of similar magnitude were found on processing but for which no explanation could be given. Doses in four subjects where a biological dose > 0.1 Sv was estimated are shown in Table 2.

Table 1. Cases investigated by occupation

Occupation	Subjects	Accidents	"Zero" dose estimates
Industrial radiography	26	16	23
Diagnostic X-ray	14	13	14
Radiotherapy	2	2	2
Dental	1	1	1
Physicist	2	2	2
University worker	2	2	2
Radiation worker	8	8	7
Public	19	2	19
Total	74	46	70

Table 2. Biological dose estimates > 0.1 Sv

Occupation	Dicentrics/ cells scored	Biological dose (Sv)	Film badge dose (Sv)
Industrial radio-grapher	7D/400	0.27	0.15-0.20
Industrial radio-grapher	6D/300	0.29	0.22
Industrial radio-grapher	6D/300	0.29	?
Factory worker	2D/500	~0.10	0.20-0.60

Industrial radiography accounted for 35% of accidents investigated. In nearly all cases the ^{192}Ir source became detached from the windout cable leaving the source unshielded. Radiation exposure occurred as a result of the fault not being detected until the termination of operations and/or in endeavouring to retrieve the source. Accidents involving hospital staff were investigated in 37% of cases, particularly staff in diagnostic radiology departments. Virtually all exposures were low, <0.1 Sv, but were investigated for B.D. as the circumstances suggested that the film badge reading might not accurately record the true dose. All external exposures involved X or γ rays except for three accidents involving mixed n and γ radiation from moisture density gauges.

Sensational and misleading information was published in the media concerning two accidents involving the "public". In the first, eight government workers were allegedly exposed to a n and γ source during clean-up operations following a truck accident. Two subjects showed symptoms of acute radiation exposure although no breach of the ^{137}Cs and $^{241}Am/Be$ source had occurred. In the second accident involving 11 subjects working for a private firm, one of the subjects was reported as "minutes away from death". His γ exposure was estimated as <0.06 Sv and a "zero" biological dose estimate was also calculated.

Details of some of the more interesting types of cases investigated are shown in Table 3.

Table 3. Highlights of radiation exposures and biological dosimetry

1. Excess film badge readings

   10 subjects with badge readings 0.15-2.50 Sv. Explanation

   - nil in 5 subjects
   - confirmed by B.D. in 2
   - probably malicious exposure in 2
   - chemical vapors in 1

2. Chronic exposures

   - investigation of leukemia
   - 12 months exposure to 740 MBq ^{90}Sr through wall
   - internal U contamination
   - no details

3. Non uniform exposures

   - no B.D. estimates reported
   - 4 subjects, fingers or feet
   - all subjects showing 1-2 incomplete cells with chromosome aberrations

4. Occasional dicentrics

   - 8/74 subjects with 1D/300-500
   - 5/8 subjects industrial radiographers
   - no B.D. estimates reported

There were 8 subjects, including five industrial radiographers, where one dicentric was found in 300-500 cells. These cases pose a dilemma in biological dosimetry as to whether the dicentric can be rightfully ascribed to the radiation accident under investigation, to chronic occupational exposure or to the spontaneous background rate for chromosome aberrations (1D/1500-2000 cells).

The presentation of cytogenetic data in court cases, where subjects are seeking compensation for alleged radiation accidents involving acute or chronic occupational exposures, is a new aspect of B.D. In the latter type of exposure B.D. estimates can only be very approximate as the aberration yield is influenced by a combination of factors including age, time of sampling, differing patterns of dose accumulation and variation in lymphocyte kinetics between individuals. In all court cases so far the prosecution has used the absence of chromosome aberrations as evidence that no significant radiation exposure has taken place, despite the lack of scientific validity in making such a statement.

In one case a physicist employed in the radiotherapy department of a large hospital was diagnosed as having chronic myeloid leukemia at 37 years of age. His film badge records showed a total accumulated dose of 0.02 Sv for his 15 years of employment with little exposure over the last 7 years. No film badge doses exceeded monthly permissible limits. There was no history of other radiation exposure or of clastogenic drug ingestion. No dicentrics were detected in 1000 cells. The subject sued the hospital for negligence, claiming that the film badge records did not give an accurate record of his actual exposures, particularly during an early 10 year period when he was loading and transporting radium sources for intracavitary exposures. For this case a probability of 3.9% was calculated for the induction of chronic myeloid leukemia due to radiation exposure based on the radioepidemiological tables of the National Institute of Health (1985).

COMPARATIVE STUDY ON SCORING AND IDENTIFICATION OF CHROMOSOME ABERRATIONS AMONG 4 LABORATORIES IN CHINA

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and

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ABSTRACT

The aim of this paper was to examine the differences of metaphase selection, identification and analysis of chromosome aberration among 4 laboratories in China. Based on this comparison a standard dose-response curve (mean value of 4 Lab.) and 4 separate curves were established. The radiation source was ^{60}Co r-rays at dose rate 100rad/min. FPG method was used to analysis first metaphases. For the purpose of examining mentioned above differences, researchers from 4 laboratories have scored the chromosome number and aberration on the same predefined areas of 4 slides.

The range of cell number selected for analysis was 56.8-65.5% with the mean of 61.73%. The frequencies of aberrant cell, Dic+R and acentrics varied within the ranges of 38.7-46.8%,30.8-42.6% and 19.4-20.6% respectively. There was no significant statistical difference in metaphase selection and aberration identification among the scorers.

The four dose-response relationships obtained by each laboratory were best fitted by using regression analysis. Three best fitting models have been as $Y=A\ Exp(B/D+C)$ for aberrant cells, $Y=AD+BD^2$ for D+R and $Y=C+AD+BD^2$ for acentrics, where Y is the yield of aberrations, D is the radiation dose in Gy. There was more variation of A and B in the latter models than those of metaphase selection and identification of this study, but all were not significant from the mean curve. The difference may be due to scoring criteria of chromosome aberration and distribution deviation of aberrations in different slides as well.

THE EFFECT OF LOW LEVELS OF IONIZING RADIATION ON PERIPHERAL BLOOD CHROMOSOMES

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ABSTRACT

The shapes of various dose-effect curves for the frequency of chromosome aberrations induced by low doses (up to 0.05 Gy) of ionizing radiation turned out to be not linear, as usually expected by theory, but rather revealed a plateau in a certain dose range. A review will be given of several in-vitro and in-vivo investigations hitherto published. Regarding the rising environmental radioactive burden of man information on low dose effects and especially of alpha irradiation, e.g. due to radon inhalation, is becoming more and more important. Since the possibilities of in-vivo studies on people exposed to elevated levels of alpha emitters are limited, in-vitro experiments have to be carried out. The induction of chromosome aberrations with alpha particles causes, however, many difficulties regarding the chemical toxicity of the radionuclides and the uniform dose distribution within the blood sample. To avoid these problems we developed out a novel method to expose human blood lymphocytes in-vitro to Rn 222 decay products. Preliminary results confirmed the nonlinearity of the dose-effect relationship at very low doses. In concordance with our former in-vivo investigations on a population living in an elevated radioactive environment the aberration rates increase sharply with dose at very low doses and allow us to obtain a more accurate dose effect curve in this important dose range.

THE CYTOGENETIC EFFECTS OF TRITIATED WATER (HTO) IN HUMAN LYMPHOCYTES: RBEs AND COMBINED EFFECTS WITH CHEMICALS

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ABSTRACT

Tritiated water (HTO) is expected to be released from nuclear-fusion research facilities and power stations in the future. Biological effect-monitoring systems should be developed to assess and control genetic effects of HTO in both working and living environments. The chromosome aberration, a directly-visible change in genetic material, is one of the most sensitive indicator of radiation exposure. Human lymphoctyes are also cell samples that can be most easily obtained.

We have performed experiments on human lymphocytes in vitro to get information on dose-response relationships between the frequency of chromosome aberrations (dicentric and ring chromosomes; D+R) and radiation doses by tritium beta-rays and ^{137}Cs-gamma-ray. These relations can be readily applicable for estimating exposure to the human body.

Human lymphocytes at G_0 were exposed in vitro to HTO (1) to obtain the dose-response relationships between exposure and the dicentric and ring (D+R) frequency, (2) to see the dose-rate effect on the D+R frequency, and (3) to investigate the enhancing effect of the combined treatment with metabolic inhibitors on the HTO-induced chromosome aberrations. The dose curve obtained was shown to be fitted by a power model: $Y = 0.205\ X^{1.29}$, or a linear quadratic model: $Y = 1.5 \times 10^{-3}\ X + 1.9 \times 10^{-6}\ X^2$, where X is HTO exposure in rad and Y is the D+R frequency. When lymphocytes were exposed to HTO at 3 different dose rates, it was found that exposure at 0.042 Gy/h produced a 30% smaller number of D+R chromosomes than that at 0.5 Gy/h. In cultures exposed to 2-Gy HTO along with various metabolic inhibitors such as cytosine arabinoside (ara-C), 3-aminobenzamide (3-AB), hydroxyurea (HU), fluorodeoxyuridine (FUdR), Caffeine, and cycloheximide (CHX), a remarkably enhanced ratio of D+R frequencies was observed when cells were combinedly exposed to ara-C, varying from 2 to 10 among lymphocytes from different blood donors. Caffeine, FUdR and 3-AB also enhanced HTO-induced chromosome aberrations by 20-50% whereas HU and CHX appeared not to affect the D+R frequency.

Relative biological effectiveness (RBE) was also assessed using the chromosome aberrations as a biological endpoint with different dose rates. The data indicate that RBEs of HTO beta-particles were 1.5-1.8 with dose-rates ranging from 0.3 to 10 rads/min, but that the RBE appeared to be several times larger with a dose rate of 0.1 rad/min.

FLOW CYTOMETRIC QUANTIFICATION OF PERIPHERAL BLOOD RETICULOCYTES AFTER RHODAMINE 123 STAINING

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ABSTRACT

The counting of the rare peripheral blood reticulocytes, which are immature red blood cells, can be used as an indicator of radiation exposure. However, the current microscopic method cannot be envisaged on large scale, and more rapid and precise techniques are needed.

Flow cytometry could be one of them. Indeed, some results have been still obtained in reticulocyte quantification by fluorescent measurement, after staining with acridine orange, pyronin Y, thioflavine and cyanin dye DIOC1(3). Nevertheless, the fluorochromes used have several drawbacks such as: difficulty in handling, apparatus specificity, low quantum yield, narrow Stroke's shifts, and no commercial availability.

We have set up a flow cytometric method (ATC 3000, ODAM-Bruker), using rhodamine 123, a common fluorescent probe for mitochondrion. The results obtained in reticulocyte detection in human and rat blood, showed close agreement between the manual method using brilliant cresyl blue, and the flow cytometric one. Indeed, the presence of mitochrondrions have been demonstrated by electronic microscopy, and allow to suggest a new operational definition of the reticulocyte.

Although further work is needed, this new method, incomparably more rapid and precise, seems promising for the determination of reticulocyte frequencies after irradiation exposure, especiall in case of mass examination.

DETECTION OF HUMAN CHROMOSOME KINETOCHORES BY FLOW CYTOMETRY: A NEW TOOL FOR BIOLOGICAL DOSIMETRY?

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ABSTRACT

In case of individual exposure to irradiations, the microscopic counting of dicentric chromosomes is used to estimate the dose level. Unfortunately, this manual method, slow and tedious, is unusable for major environmental accidents, and for systematical studies of large populations.

Flow cytometry is able to fast analyse chromosomes according to their DNA content, but only sophisticated prototype slit-scan apparatus can detect dicentric chromosomes.

In last years, it has been shown that the blood serum of patients with the CREST syndrome of scleroderma, contains antibodies against kinetochore, which is the proteinaceous structure of the centromeric region, that attaches to the spindle microtubules.

The goal of our study is to use such antibodies to label chromosome centromeres, in order to detect and quantify the relative number of dicentric and acentric chromosomes with a commercial flow cytometer.

With this aim, different points are to be resolved: (1) the kinetochore labelling of chromosomes in suspension, using antikinetochore sera and FITC-labelled secondary antihuman antibodies; (2) the sensitivity of the flow cytometer used (ATC 3000 - ODAM Bruker), as kinetochore being a very small part of the chromosome gives a weak fluorescence; (3) the design of a double fluorescent labelling for recognition of the chromosomes; (4) the enormous difference of fluorescence between the chomosomal DNA and the kinetochore.

Several of these points have been resolved, allowing to demonstrate clearly chromosomal kinetochore detection by immuno-flow cytometry. Although there is a lot of work before to use this technique as a routine one, our results show the feasibility of this chromosome labelling.

The use of monoclonal antibodies against kinetochores should improve the detection of the centromeres and, consequently of dicentric chromosomes in irradiated cell population.

QUANTIFICATION OF RADIATION TRANSFORMATION FREQUENCIES

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ABSTRACT

The relationship between the administered dose of irradiation and the frequency of transformation *in vitro* remains to be clarified. Several authors have detected 'bell shaped' or plateau type dose response curves, where the transformation frequency either reaches a constant level or falls off at high doses, even after an allowance is made for cell kill.

We have recently been studying the development of transformation in primary thyroid cultures and found a characteristic bell shaped curve when our results were corrected for initial cell kill using a clonogenic assay. However, we also noticed while trying to isolate immortal clones in serial subcultures that the plating efficiency of cells contained in irradiated survivor colonies in a variety of cell types including C3H 10T 1/2 and throid cells was considerably below normal. The effect is dose dependent and is significant out to the third subculture. This means that using the first survival curve will lead to a considerable overestimate of the number of surviving cells and a consequent underestimate of transformation frequency. Because of the problem of senescence of primary thyroid cells it is not easy to quantify the degree to which the effect alters the observed transformation frequency but when the effect is taken into consideration using C3H 10T 1/2 cells it raises the observed transformation frequency based on the initial surviving fraction considerably and converts the characteristic plateau type C3H 10T 1/2 transformation dose response curve to a curvilinear or linear response. The results have disturbing implications for the use of *in vitro* transformation data in the assessment of carcinogenic risk.

RADIATION EFFECT ON HUMAN DIPLOID CELLS

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ABSTRACT

We studied in vitro the influence of $^{239}PuO_2$ with MMD 1.3um on short-term effect and long-term effect of human diploid cell population called 2BS cell line that is human embryo lung fibroblasts. The short-term effect includes proliferation, surviving fraction and ultrastructural changes. The long-term effect involves life-span shortening and transformation of the cells.

The short-term effect observed is that the changes of proliferation and surviving fraction of the cell line was significant following exposure at concentration of 0.0006uCi/ml observed for a week. On the 7th day after exposure the proliferation of control and treated cells was 10.5 and 4.88 PDN (population doubling number) and the surviving rate of them was 96% and 56.3% respectively.

For the long-term effect studies, the follow-up observation for 12 groups of culture exposed to $^{239}PuO_2$ was made. In three groups exposed at 0.003uCi/ml, the shape of the cells became shorter and shorter and their life-span shortened with higher mortality appearing early death. In 7 cultures exposed at 0.0015uCi/ml, the similar changes of cell morphology were the same as above and their life-span was shortened by 58.7% averagely. In other cultures at same concentration, not only the life-span was not shortened but it was longer than that on the controls. At last they appeared epithelium - like shape of transformed cells.

By electron microscopy the treated cells revealed sinking of the nuclear membranes into nucleus and hypertrophy of the nuclealus with netted stracture. The nuclear volume increased largely and the ratio between nucleus and cytoplasm increased as well.

In this paper we provided the data of human cells in vitro for radiation protection, which are considered important in the absence of human experiment conditions.

THE INFLUENCE OF Cs-137 ON THE TRANSMEMBRANE RESTING POTENTIAL AND ON THE PROLIFERATION OF HUMAN LUNG CELLS

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ABSTRACT

Since Cs-137 was a main fallout product detected after the Chernobyl accident, it is of primary interest to investigate the effects of this radionuclide on human cells. Human embryonic lung fibroblasts in vitro allow experiments under standardized and reproducible conditions. Specific Cs-137-activities in the range of 0.56 Bq/g up to 1850 Bq/g were produced in the cell growth medium. The transmembrane resting potential (TMRP), measured with the aid of glass microelectrodes, was used as an electrophysical parameter. The TMRP is an indicator for changes of the cellular membrane and physiology such as due to transformation processes or radiation damage. The changes of the TMRP of the cells after exposure to different doses of beta-rays are evaluated by statistical analysis and related to membrane and physiology changes.
In addition, microscopical observation and recording of cell growth make it possible to study the influences of Cs-137 on cell proliferation.

THE NECESSITY OF REVISING OF A-BOMB SURVIVORS DATA IN RESPECT OF INDIVIDUAL SENSITIVITY TO RADIATION

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ABSTRACT

Nowadays reconsideration of 'dose-response' relation in A-bomb survivors data is mainly concerned with the re-evaluation of doses. However, new findings on the individual sensitivity of organism to ionising radiation imposes the real necessity of reconsidering the data on A-bomb survivors.

These findings are described briefly in the paper. They are as follows:

- the endogenous content radioprotective enzyme - Superoxide Dismutase (SOD)- that naturally exists in the body, in healthy, non-exposed, human white population varies in the interval of three ranges of magnitude,

- the endogenous content fluctuates during the human life but generally remains of the same level,

- the experiments with animals (with similar to human distribution of SOD enzyme in organism) reveal that the survival of acute dose of low-LET radiation is the simple, proportional function of endogenous SOD content in the body, moreover,

- the freqency of chromosomal aberrations strongly depend on the SOD endogenous level in lymphocytes. The subgroups with different levels of SOD reveal 3-fold difference in chromosomal aberration at the same level of exposure.

Therefore, these observations impose the absolute necessity of reconsidering the A-bomb survivors data in respect of possible individual and ethnic differences in natural endogenous content of SOD.

Paper to present the above mentioned findings and briefly discuss, giving proposals of possible research projects regarding A-bomb survivors and other naturally or occupationally exposed populations. It seems to be reasonable that these ideas would be considered by interesed researchers and relevant scientific organisations (ICRP, IAEA, WHO) for possible initiation of scientific study or modifying present studies.

CELLULAR METABOLISM OF URANIUM AND THORIUM. A STUDY BY ANALYTICAL MICROSCOPY

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ABSTRACT

Two actinides, Uranium and Thorium have been injected as soluble form in rats, and tissues (Liver, bone marrow and kidney) have been studied with two methods of analytical microscopy : Electron probe X rays Microanalysis and Nuclear Microscopy. It has been shown that these elements are concentrated in a non soluble form in different varieties of cells. In the liver Uranium and Thorium are observed in hepatocytes, in the bone marrow the elements are in the macrophages and in the kidney they are in the proximal tubule cells. In each of these cells, uranium and thorium are specifically concentrated by the lysosomes where high concentrations have been measured. In these lysosomes, the actinides are precipitated as insoluble phosphates.

The mechanism of intralysosomial concentration is explained by the hight phosphatase activity of these organelles.

In the kidneys the intralysosomial concentration and precipitation are followed by a renal elimination of these elements expelled in the tubular lumen as microscopic crystals. In hepatocytes and bone marrow macrophages, the actinides may stay for a long time inside the cells and chronic intoxication may produce high local concentrations of these alpha emitters.

SURVIVAL OF V79 CHINESE HAMSTER CELLS IRRADIATED WITH HIGH-LET PROTONS: IMPLICATION FOR THE RBE-LET RELATIONSHIP.

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INTRODUCTION

Information about the biological efficiency of high linear energy transfer (LET) radiations is considered relevant to radiation protection because of:

(i) the increasing use of densely ionizing particles in several human activities;

(ii) the need for a better understanding of the basic mechanism of radiation action.

A particular interest should be deserved to the biological effects of low-energy protons. Neutrons exert their action mainly through proton recoils in the MeV range and a great part of the dose delivered by high-energy protons used for therapeutical applications is concentrated in the low-energy, high-LET region near the path end (Bragg peak). However, experimental work in this field is scarce and limited to relatively high energies.

We have studied the survival of V79 Chinese Hamster cells irradiated with proton beams in the energy range of 0.73-3.36 MeV, corresponding to an average LET inside the cells of 10.6-34.5 keV/μm.

MATERIALS AND METHODS

Irradiations with monoenergetic proton beams have been performed using the facility for radiobiological studies, set up at the 7 MV Van de Graaf accelerator of the INFN-Laboratori Nazionali di Legnaro, Padova(1). X-rays produced by an apparatus operating at 200 kV with a 0.2 mmCu filter have been used as reference.

Survival curves have been obtained in the dose range 0.5-6 Gy, at the dose rate of 1 Gy/min. 18 hrs before irradiation, $1x10^5$ V79 cells were seeded in each stainless steel Petridish especially designed to fit the beam geometry. The cells were grown at 37 °C as a monolayer attached to a mylar foil, having an area of 133 mm^2, placed at the bottom of the dishes. The cell monolayer was exposed to the beam through the mylar foil.

Immediately after irradiation the cells were detached by trypsin-EDTA treatment, counted, diluted and plated at the appropriate concentration. After 8 days of incubation at 37 °C, the survival was calculated by scoring the number of visible clones.

RESULTS AND DISCUSSION.

Fig.1 reports the survival curves for V79 cells irradiated in air with protons having energies of 3.36, 1.62, 1.16 MeV (Fig.1 A), 0.84 and 0.73 MeV (Fig.1 B), in the dose range 0.5-6.0 Gy. The energy values are referred to the incident surface of the cells. Both LETs and doses are expressed as the average values calculated at 3 μm depth, assuming a thickness of 6 μm for the cell monolayer. Each data point represents the mean of 4-10 individual experiments with one standard error of the mean.

In fig.1A experimental data were fitted using the expression:

$$S = \exp(-\alpha D - \beta D^2)$$

Increasing the LET, the initial shoulder of the survival curves tends to disappear. Linearity is reached at 23.9 keV/μm where the β coefficient results statistically irrelevant.

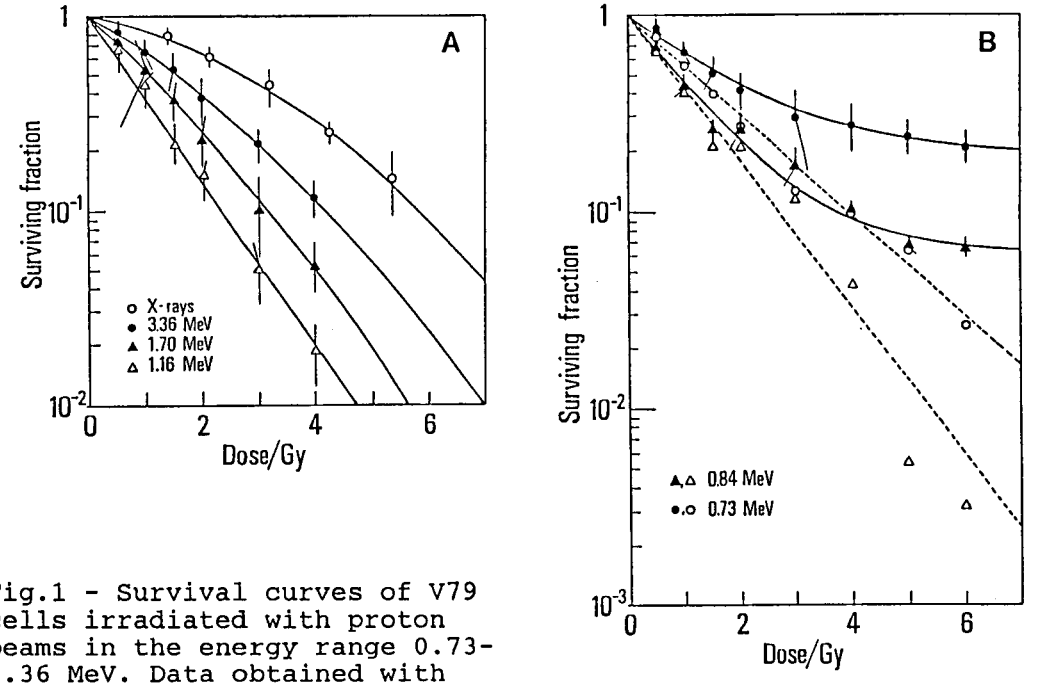

Fig.1 - Survival curves of V79 cells irradiated with proton beams in the energy range 0.73-3.36 MeV. Data obtained with X-rays are reported for comparison. Dotted lines in panel B represent the exponential parts of the fitting equation. The open symbols are the experimental data corrected for the fraction f.

For higher LET, such as those considered in Fig. 1B, the survival data show a tail at high doses, leading to apparent negative beta values when they are interpolated with the linear-quadratic expression. This effect cannot be explained in terms of LET variations and/or proton "loss" inside the cell monolayer. There are reasons to believe that it is presumably due to a portion of not flat, poorly attached or shielded cells receiving an attenuated dose of radiation. Assuming as a first approximation that a fraction f of cells has received no dose at all, we used the expression:

$$S = (1-f)\ \exp(-\alpha D)+f$$

to fit these data. The dotted lines show the fitting of the corrected data using the exponential part of the latter expression.

Table I lists the survival parameters obtained from the best fit procedures. The ratio α/α_x reported in the last column can be considered as the RBE at low doses. It can be seen that it increases with the LET, showing a maximum at about 24 keV/µm and decreases at higher LET values. Our data are consistent with those recently reported for 5.8 and 12.1 keV/µm protons (5).

Table I

Radiation	α/Gy^{-1}	β/Gy^{-2}	f	α/α_x
X-rays	0.128+/-0.023	0.046+/-0.005		1.0
Protons				
10.6 keV/µm	0.378+/-0.031	0.040+/-0.009		2.9
17.8 keV/µm	0.586+/-0.052	0.037+/-0.016		4.6
23.9 keV/µm	0.938+/-0.019			7.3
30.4 keV/µm	0.803+/-0.059		0.061+/-0.007	6.3
34.5 keV/µm	0.536+/-0.040		0.189+/-0.015	4.2

Interesting suggestions can be drawn comparing the RBE-LET relationships for protons and other particles (Fig.2). The curve appears shifted to lower LET values when compared to the results obtained with alpha particles and heavier ions (2, 3, 4). Therefore, in the rising part of the curve the RBE for protons is higher than that of alpha particles of comparable LET.

Preliminary data on the induction of $HGPRT^-$ mutants are consistent with this finding.

These results suggest that: (i) the LET could not be a suitable parameter of radiation quality and (ii) the quality factor evaluation based on the "classical" RBE-LET relationship (4) may not be of general validity. It appears that differences in the track structure, at the sub-micrometer level, among different types of radiation are of biological significance.

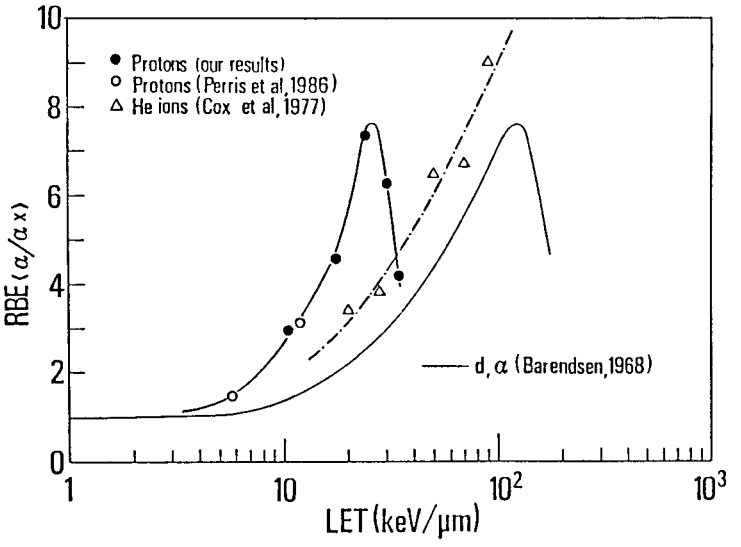

Fig.2 - RBE-LET relationships for protons and other particles.

This points out the importance of an approach based on the microscopic distribution of the energy deposited by radiation in relevant targets.

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INTERNAL CONTAMINATION WITH SEVERAL RADIONUCLIDES AND METHODS OD THERAPY

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INTRODUCTION

In cases of accidental exposure in the environment, internal contamination is likely to include several radionuclides. Special concern should be given to radioactive strontium, caesium, iodine and to highly radiotoxic transuranium elements. Efficient therapeutic agents for reducing retention of single radionuclides are already in use for radiological protection purposes i.e. alginate for radiostrontium (Skoryna et al., 1964), ferrihexacyanoferrate(II) for radiocaesium (Madshus et al., 1966), potassium iodide for radioiodine (Ramsden et al., 1967) and diethylenetriaminepentaacetic acid (DTPA) for transuranium elements (Catsch and Harmuth-Hoene, 1979). It would be desirable to devise a therapeutic regimen which would reduce the body burden not only of one but of all the radionuclides that present a hazard and which at the same time is simple to apply. Such treatment involving necessarily simultaneous administration of several different therapeutic agents could modify the efficacy of each agent or cause other adverse effects due to interactions between them. Studies presented here are related to the problem of internal contamination with several radionuclides and therapeutic treatment administered.

ANIMAL STUDIES

MATERIALS AND METHODS

The experiments were performed on 6-8-week-old female albino rats (about 150 g body weight). The animals were divided into three groups according to the therapeutic treatment administered. The first group, CONTROL, was fed standard rat diet; the second, MIXTURE + Ca-DTPA, was given food supplemented with the mixture of 15 g calcium alginate *, 2.5 g ferrihexacyanoferrate(II)§ and 0.015 g KI# per 100 g of diet and received also chelating agent Ca-DTPA intraperitoneally (380 µmol/kg body weight); the third group, MIXTURE + Zn-DTPA, received the aforementioned mixture of therapeutic agents in food together with 3.3 mmol Zn-DTPA per 100 g of diet. On the second day of the experiment, the animals received 74 kBq Sr-85, 37 kBq Cs-137, 740 kBq I-131, 1850 kBq Ce-141 orally or 37 kBq Ce-141 intraperitoneally.

The complexing agents Ca-DTPA or Zn-DTPA were prepared by dissolving diethylenetriaminepentaacetic acid⊕ in distilled water

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§Radiogardase-Cs, Heyl and Co., West Berlin, FRG.
#Kemika, Zagreb, Yugoslavia
⊕Fluka, A.G., Buch, S.G., Switzerland.

in the presence of an equimolar amount of calcium or zinc chloride and neutralizing it with 20% NaOH to pH 6.4. Ca-DTPA was administered intraperitoneally while Zn-DTPA was added to standard rat diet which was ground and already mixed with calcium alginate, ferrihexacyanoferrate(II) and KI. The therapeutic agents were administered in food during the first three days of the experiment (one day before and two days after radionuclide administration) and afterwards the animals were fed standard rat diet.

Radionuclides of Sr-85, Cs-137 and Ce-141 were supplied from New England Nuclear, Dreieich°, as chlorides of high specific activity. Radioiodine was supplied from the Institute "Boris Kidrič"- Vinča€, as natrium iodide of high specific activity. All the animals except those which received Ce-141 orally were killed six days after radionuclide administration. Those which received Ce-141 orally were killed after 24 hr because of its low intestinal absorption. The radioactivity in the whole body, carcass and gut was determined in a single-channel, twin-crystal scintillation counter (3X3" cylindrical NaI(Tl) crystals) and of the organs in an automatic well-type scintillation counter∞. All values are expressed as percentage of the administered dose and presented as arithmetic means and standard error of the means.

RESULTS

Oral administration of alginate, ferrihexacyanoferrate(II) and KI together with intraperitoneal application of Ca-DTPA reduced the whole body retention of orally administered Sr-85 about 5 times, Cs-137 retention was reduced about 56 times and I-131 retention about 8 times while retention of intraperitoneally administered Ce-141 in liver and bone was reduced about 10 times. In case of simultaneous oral administration of all therapeutic agents (calcium alginate, ferrihexacyanoferrate(II), potassium iodide and Zn-DTPA), the retention of orally administered Sr-85, Cs-137 and I-131 was reduced 9, 40 and 11 times respectively while retention of intraperitoneally administered Ce-141 was reduced 1.4 times. The very low retention of orally administered Ce-141 was significantly increased by administration of the mixture and Zn-DTPA. Compared to controls the whole body and bone retention were increased 2.5 and 10 times respectively.

DISCUSSION

Simultaneous oral administration of the mixture of calcium alginate, ferrihexacyanoferrate(II) and KI with intraperitoneal (Ca-DTPA) or oral (Zn-DTPA) administration of chelating agent did not diminish the efficacy of the single antidotes. The retention of intraperitoneally administered Ce-141 was less reduced by

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orally administered Zn-DTPA than by intraperitoneally administered Ca-DTPA. The difference in effectivenes in our experiment is probably due to the lower efficacy of Zn-DTPA in comparison to Ca-DTPA when it is administered immediately after the radionuclide (Seidel, 1976). Increased values of Ce-141 retention after its oral administration together with chelating agent are due to the fact that oral chelation therapy enhances intestinal absorption of ingested transuranium elements for which chelation proves to be an efficient therapy. However, it also enhances the elimination of the chelated metal. The ultimate body burden may therefore be only slightly higher than if the chelating agent had not been employed. In our experiment measurements were done 24 hr after Ce-141 oral administration i.e. at a time when most of the chelated Ce-141 was not yet eliminated. We therefore assume that cerium retention values in these animals would be much lower if allowing enough time for cerium elimination.

The results obtained indicate that Ca-DTPA or Zn-DTPA did not cause a loss of efficacy of antidotes from the mixture nor did the antidotes from the mixture alter the effect of chelating agents. All four therapeutic agents can be used together without undesirable interactions. Very high efficacy of parenterally administered Ca-DTPA can not be reached by much higher oral doses of Zn-DTPA, especially in cases of their immediate administration after radionuclide entry in the body (Taylor and Volf, 1980). However, Zn-DTPA has substantially lower toxicity than Ca-DTPA and its oral administration would greatly simplify the therapy especially in cases of prolonged treatment

HUMAN STUDIES

In order to obtain relevant information in humans we used a similar therapeutic treatment and examined its effect on I-131 and Sr-85 uptake in four adult volunteers (one man and 3 women). Three of them received two oral doses of radioactive iodine I-131 (185 kBq each). The second dose of I-131 was administered 13 days after the first one. The subjects fasted 10 hr before and 4 hr after radionuclide administration. Uptake values after the antidote treatment were measured after the first radioiodine administration and normal uptake values for each subject were taken after the second I-131 dose. Measurements were made 2, 4 24 and 48 hr after each radioiodine administration. The antidote treatment consisted of 10 g calcium alginate, 3 g ferrihexacyanoferrate(II), 130 mg potassium iodide and 5 g Zn-DTPA in a volume of about 150 ml water. The antidote treatment was given 30 minutes before the first I-131 administration. Normal I-131 thyroid uptake values 24 hr after radionuclide administration were 26, 42, 30% in three subjects. After antidote treatment the respective I-131 values were 0.5, 1.3, 0.9 % indicating the almost complete block of thyroid uptake. This indicates that simultaneous administration of other antidotal agents do not cancel or decrease the effect of potassium iodide on I-131 thyroid uptake.

In the separate study one of the volunteers received the same therapeutic treatment as in aforementioned study but the

radionuclide administered was Sr-85. Two oral doses of Sr-85 were administered within the interval of 8 days. The first dose was given together with the therapeutic treatment while after the second dose no treatment was given. The plasma radioactivity level was used as indicator of radiostrontium absorption.The results were expressed as percentages of the oral dosage per liter of plasma. Plasma radioactivity was 0.1%/L after the first and 1.8%/L after the second Sr-85 administration. The therapeutic treatment reduced radiostrontium absorption by a factor of 18. This is in agreement with the results of Hodgkinson and collaborators (1967) obtained in subjects who were without a treatment (control values) or received 4 g of sodium alginate. Presence of other therapeutic agents had no effect on the efficacy of alginate to reduce radiostrontium absorption.

The results obtained deserve attention since simultaneous administration of several therapeutic agents (alginate, ferrihexacyanoferrate(II), KI and chelating agent) might present a convenient method for early and delayed therapy of internal contamination with biologically dangerous radionuclides.This may be importand in cases of increased exposure in the environment in situations where radionuclide identification is difficult or impossible to make before administration of the therapy.

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RADIATION PROTECTION BY BOTANICAL EXTRACT FROM ARALIACEAE FAMILY

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A single post-irradiation injection of a ginseng extract increased the 30-day survival ratios of X-ray-irradiated mice, rats and guinea pigs. The injection accelerated the recovery of hemogram (erythrocyte, leukocyte and thrombocyte couts), particularly strongly that of thrombocyte count, commonly in the three species of the experimental animals. The extract increased the 30-day survival ratio, but stimulated recovery of only thrombocyte count in splenectomized mice. We consider that the recovery of thrombocyte count is one of the most important factors for the decrease in bone marrow death after acute ionizing irradiation.

Extracts from several plants of Araliaceae family also showed similar radioprotective activity on X-irradiated mice. Those plants are Acanthopanax senticosus (Rupr. et Maxim.) Harms., Aralia cordata Thum., Aralia elata (Miq.) Seem. and Tetrapanax papyrifera (Hook.) Koch.

Ginseng (root of Panax ginseng C. A. Meyer) is a plant of Araliaceae family, and has been considered as the most prized medicinal tonic herb in China, Korea and Japan. A partially purified aqueous-buffer extract or saponins of the plant was reported to posess hematopoiesis-stimulating activity (1). We applied the ginseng extract to the irradiated laboratory animals and confirmed that the extract has radioprotective activity even administered after irradiation.

The extract was prepared by the following procedures; extraction of powdered ginseng with 0.05 M Tris-HCl buffer (pH 7.6) by stirring for 1-2 days at 4 ^{o}C, centrifugation of the filtrate, concentration (about 1/4 volume) of the supernatant, ammonium sulfate (0.7 saturation) precipitation, dialysis, and lyophilization. This preparation corresponds to fraction 3 of the purification procedure of ginseng saponin described by Oura et al. (2). In some experiments the thermostable supernatant fraction was used which was obtained after heating ginseng extract dissolved in physiological saline (adjusted to pH 7) in a boiling-water bath for 15 min.

When mice were irradiated with 7.2 Gy of X-rays they died from 10 to 20 days after the irradiation. Injection of the extract within 5 min after the exposure increased the 30-day survival ratio with increasing dose of the extract as shown in Fig. 1. Even a dose of 1.8 mg resulted in a significant ($p<0.001$) radioprotection. Injection of the extract 2.5 hr after irradiation was samely efficacious as just after irradiation (3).

Radioprotective activity of the extract on rats and guinea pigs are shown in Table 1 and 2, respectively (4).

An extract similarly prepared as noted above from Acanthopanax senticosus (Rupr. et Maxim.) Harms, another plant of

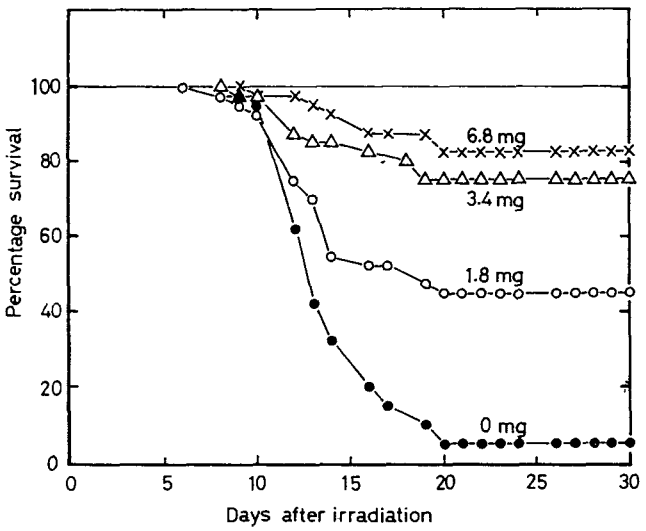

Fig. 1. Effect of dose of the ginseng extract on the survival of irradiated mice. Forty animals were used for each group. The extract was i. p. injected after irradiation of 720 R.

Table 1. Radioprotective effect of the thermostable fraction of ginseng extract in rats irradiated with 825 R of X-rays.

Injection	Dose (mg)	Dose ratio*1)	30-Day survival ratio (%)	Statistics
Thermostable fraction	6.0	1.0	80.0 (40)*2)	P<0.001
Physiological saline	0.0	–	30.0 (40)	–

*1) Ratio of dose to body weight compared with that for mice.
*2) Numbers in parentheses show the number of animals.

Table 2. Radioprotective effect of the thermostable fraction of ginseng extract in guinea pigs irradiated with 325 R of X-rays.

Sex	Average body weight (g)	Dose (mg)*1)	Dose ratio*2)	30-Day survival ratio (%)		Statistics
				Experimental	Control	
Male	357	26	1.2	45.0 (20)*3)	40.0 (20)	P>0.05
Male	288	72	4.1	47.2 (36)	10.0 (40)	P<0.001
Female	294	76	4.3	65.0 (20)	10.0 (20)	P<0.01

*1) Injected in 4.0 ml of physiological saline per 300 g of body weight.
*2) Dose to body weight compared to that for mice.
*3) Numbers in parentheses show the number of animals.

Araliaceae family, also decreased the incidence of bone marrow death in mice (Table 3). Recovery of hemograms in X-irradiated mice was accelerated by the extract (5). The effect was most pronounced on the recovery of thrombocyte counts, as in the case of ginseng extract.

Table 3. Radioprotective effect of Acanthopanax senticosus extract

Group	Dose (mg/animal)	30-day survival ratio (%)	Significance
Extract	8.2	70.0 (21/30)	$p < 0.001$
Control	0	2.5 (1/40)	

Methanol-solble fraction of both ginseng and A. senticosus extracts failed to protect the irradiated mice (5, 6). The results show that the radioprotective principles in the extracts were not low molecular weight substances such as saponins or eleuterosides, originally thought to be the active principles in the plants.

We then improved the preparation procedure of the extracts to an easier and simpler one. Chopped dry herb was decocted with 10 vol/wt of water. The filtrate was centrifuged, and the supernatant was concentrated to a syrup with a rotary evaporator under reduced pressure. The syrup was dropped into 10 volumes of ethanol. The precipitate was collected by centrifugation, and dissolved in a small amount of water. Dried extract was obtained after lyophilization.

Extracts prepared from plants of Araliaceae family by the preceding procedure were subjected to examine their radioprotective activity. Injection dose per mouse (about 30 g of body weight) was beforehand determined by measuring body weight, splenic weight and observing coat of fur on the next day after injection. Table 4 shows that Aralia cordata Thum., Aralia elata (Miq.) Seem. and Tetrapanax papyrifera (Hook.) Koch are radioprotective, and Hedera rhombea Bean, Gilibertia trifida Makino, Fetsia japonica Decne. et Planch and Acanthopanax Sieboldianus Makino are not.

Table 4. Radioprotective activity distribution in Araliaceae plants

Name of plant	Dose (mg)	Survival	Significance
Control	0	3/40	
Aralia cordata Thum.	9.0	22/40	$p < 0.001$
Control	0	1/40	
Aralia elata (Miq.) Seem.	8.2	12/40	$p < 0.01$
Hedera rhombea Bean	22.0	5/40	$p > 0.05$
Gilibertia trifida Makino	18.0	3/40	$p > 0.05$
Fetsia japinica Decne. et Planch	27.0	2/40	$p > 0.05$
Control	0	5/40	
Acanthopanax Sieboldianus Makino	9.0	6/40	$p > 0.05$
Tetrapanax papyrifera (Hook.) Koch	24.0	14/40	$p < 0.05$

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A NEW RADIOPROTECTIVE MEANS BY STIMULATING BODY DEFENSE MECHANISM AND METALLOTHIONEIN INDUCTION

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ABSTRACT

Radioprotective means against damage due to lethal dose of radiation was found by the administration of heavy metals or immunostimulants, or subjecting the organism to some kinds of stresses. Mice were subjected to various treatments prior to the whole body single irradiation of lethal dose X rays, 5 - 9 Gy. Treatments performed were (a) Subcutaneous or intraperitoneal injection of metal ion, e.g. $CdCl_2$, $MnCl_2$ or Zn acetate, (b) Intraperitoneal injection of immunostimulant e.g. OK-432, PS-K or Il-1, or (c) Excision of 2 x 2 cm^2 portion of dorsal skin.

LD_{50} during 30 days of post-irradiation in the pretreated mice were 1.5 - 2.0 Gy higher than that of the irradiated control mice without treatment. Pretreated mice which had showed strong tolerance against radiation had always higher contents of metallothionein in their liver during 0 - 3 days of post-irradiation period, i.e. ten to twenty tomes of metallothionein level in comparison to the control animal without treatment.

This method provides a new and powerful means of protection against radiation within the body with almost no adverse effect, as the parallel to the usual exogeneous controls of radiation exposure.

TRITIUM METABOLISM IN ANIMALS AND ESTIMATION OF THE ACCUMULATED DOSE

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We aimed to establish a beneficial model for the prediction of dose commitment in man after receiving tritium . For this purpose mother mice and their offsprings were used as a model system of tritium transfer and the accumulated dose was estimated for various organs.

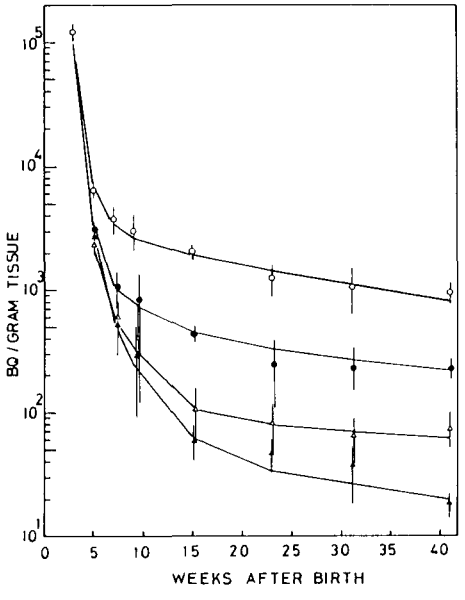
When the newborn mice were nursed by the radioactive mothers, the total radioactivity per gram tissue decreased initially afer birth and at 2 weeks after birth it reached 80 - 90 % of the level at birth. Between the second and the forth week after birth it increased and after deletion of the tritium source rapidly decreased. In another experiment where the newborn mice were nursed by the non-radioactive mothers, tritium was eliminated from the animal bodies exponentially. These experiments revealed that the major portion of tritium activity is contained in the acid soluble component as long as T-TdR was supplied prenatally or neonatally. The analysis of the acid soluble fraction showed that it contained more than 95 % of free water except for the heart at birth. Considering the fact that, at 2 - 3 weeks after birth, weaning of the newborn animals takes place and they begin to ingest tritium orally, it is suggested that more than about 95 % of the OBT orally ingested and absorbed from the gastrointestinal tracts was catabolized to HTO before incorporated into DNA .

The DNA-bound tritium exhibited a time dependence being quite different from the total tritium . The tritium activities for the both cases decreased almost in parallel with each other until 2 weeks after birth. After weaning which occurs at 2- 3 week after birth the difference of the DNA-bound radioactivity became obvious between the two types of experiment especially in rapidly proliferating organs such as spleen, liver, stomach, intestine and thymus. This observation suggested that thymidine as a DNA precursor is transfered to embryos placentally but not to the milk components.

The fate of incorporated tritium in the offsprings nursed by the mothers ingested with HTO had been studied . The biological half-lives and intitial concentrations of major components can be obtained by the curve fitting (Fig. 1). The half-life for the first component took the value between 2.2 and 2.6 days. The second component had the half-lives between about 10 and 16 days and the third component exhibited halflives between about 110 and 160 days. The acid soluble component was the rate limiting factor of tritium retention at the initial stage of animal growth. The acid-soluble component contained tritium activity which decreased in percent contribution with further animal growth. At

3 weeks after birth, the patterns of tritium distribution to various molecular components in various organs were are similar to each other while they tended to be organ-specific after a long-term experiment. At 10 -15 weeks after birth, in the liver,intestine and kidney the DNA component became the major determining factor of tritium activity while in the lung and heart, the protein component.

Tritium was most strongly retained in the brain. This was due to the high tritium retention in the lipid component of the brain. The slow component of the total tritium retention curve can be interpreted by low turnover rates of lipid-,DNA- and protein-bound tritium. At 41 weeks after birth, the largest contribution to the total tritium activity in the spleen , the liver and the lung was found for the lipid (27 %), the DNA (52 %) and the protein component (54%), respectively. Thus for each organ there were molecular components critical for dose accumulation (Fig. 2).



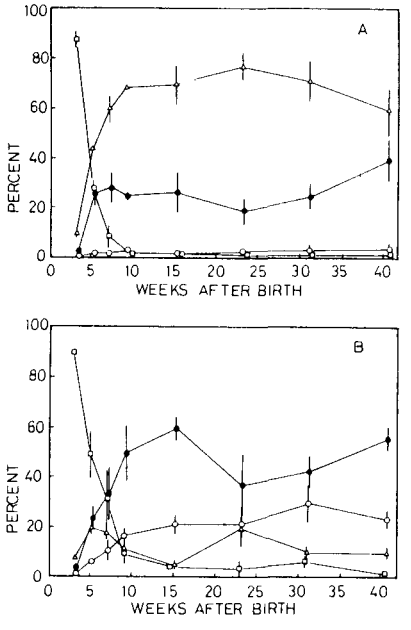


Fig.1(left) Total tritium activity in various organs of mice after supplied with tritium as milk components from their mothers. (○),the brain;(●),the lung;(△),the intestine;(▲),the liver.
Fig.2(right) Percentage contributions of various molecular components to the total tritium activity in mice given tritium from their mothers. A,the brain;B,the lung. (□),acid soluble component; (△),lipid;(○),DNA;(●),protein.

The percent contribution to the total dose of the acid insoluble components was organ-specific and was between about 17 and 42 %.(Table 1). The contribution of organically-bound tritium to the committed dose equivalent would become much greater if the period of dose accumulation is extended to the life span of the animals. This finding indicates that the contribution of organically-bound tritium to dose commitment is comparable to the contribution of the free water component.

Table 1 Percent contributions of various molecular components to to the total accumulated dose in newborn mice which received tritium through suckling stage from mothers which are given HTO as drinking water.

Organ	Acid soluble	Lipid	RNA	DNA	Protein	Total acid insoluble
Spleen	83	7	1	3	6	17
Liver	84	11	1	<1	4	17
Intestine	83	12	1	1	3	18
Lung	79	8	2	3	9	21
Kidney	83	9	<1	2	5	17
Heart	82	8	1	1	8	18
Brain	58	30	<1	<1	11	42

Tritium localization to specific subcellular components or structures must be emphasized for OBT since OBT is retained in the cell to more extent than LT and inhomogeneity of tritium distribution becomes more and more evident by extending the period of observation after deletion of tritium source or during chronic ingestion. We attempted to determine the dose modifying factors (DMF) for DNA and protein-bound tritium. Various cytological factors relevant to DMFs were determined experimentally. Computer simulation works were performed to help these works.

The absorbed dose due to nucleus-bound tritium decreased with a decreasing cell diameter since the extent of the edge effect becomes more and more evident with decreasing the nucleus diameter. The energy deposition in the cell nucleus was reduced by 30 and 20 % for the spleen and the intestine respectively, compared with the dose to a nucleus with an infinite diameter. The accumulated dose in the cell nucleus of the newborn mouse was calculated using the values of specic activitiy of DNA. The values of the DMF for DNA-bound tritium under the present experimental condition are also given in this table.

Organ-specific dose accumulation was obvious from the experiment using newborn mice. Such organ-specificity of the dose from DNA-bound tritium was interpreted by the difference mainly of the two factors, i.e. average diameter of the cell nucleus and the nuclear DNA content. In fact we observed that the specific activity of DNA in heart was about 4 times that in spleen while the calculated dose for the heart nucleus was only about 80 % of

the dose for the spleen nucleus.

In the mice which received tritium from their mothers milk, the contribution of the protein component to the accumulated dose is organ specific. Among the spleen, liver , intestine, lung, kidney, heart and brain , the largest contribution of protein is seen in the brain (11 %) and the smallest in the intestine (3 %). If the DMF value of the protein component deviates considerably from unity, the contribution of OBT to the total dose will be comparable to the contribution from the free water component. However, under the situation that the proteins are uniformly labelled in the whole organ, the DMF value for the protein-bound tritium is about between 0.8 and 1.3. Thus for a practical use of the metabolic data in dose evaluation, within an ambiguity of at most 30 %, the dose commitment from incorporated tritium can be evaluated without considering the localization of tritium to the protein molecules.

The accumulated dose due to the protein-bound tritium in the cell nuclei of animal organs may depend on the difference of protein concentration between cell nucleus and the extranuclear region of tissue. To test this possibility experimentally and theoretically and to estimate the localization factor for the accumulated dose for the cell nucleu, a simplified model of protein distribution in animal tissues was made and compared with experimental results.

From the result of computer simulation , it was concluded that the contribution of the nucleus tritium of neighbouring cell nuclei can be neglected compared to the contribution from the tritium distributed in the cell nucleus under consideration. In our model calculation, it was assumed that the cell nucleus is spheric in shape. Under the assumption that protein is uniformly distributed either in the cell nucleus or in the extranuclear region individually, the dose to cell nucleus can be calculated as that averaged for the whole nucleus. At the subnuclear level there might be an inhomogenous dose distribution which occurs as a result of inhomogeneity of protein distribution. However, the inhomogeneity of tritium distribution seems to be of minor importance to determine the DMF value since the percent contribution from the extranuclear region to the dose to nucleus is about 10 - 20 %. This situation gives the DMF values about between 0.8 and 1.3. These values are much less than the DMF values for the DNA-bound tritium and have less deviations as well.

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METABOLISM AND GENETIC ACTION OF BIOGENIC TRITIUM IN MAMMALS

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ABSTRACT

In the process of biosynthesis the general metabolic behaviour of ^{3}H-labeled precursors of biopolymers consists in stable binding of 5-80% of tritium activity by mammalian tissues and simultaneous chipping of 90-10% of the label and its incorporating as THO into the body liquids. When injected into the body of mice and rats, ^{3}H-glucose and ^{3}H-amino acids will cause the tissue dose exceeding that from the equal quantity of THO by 1.1 - 10 times. The nonreplaceable amino acid L-lisin-^{3}H will cause the greatest dose and within the space of 3-5 months increase the frequency of dominant lethal mutations (DLM) in male mice by 2-4 times as compared with THO.

^{3}H-desoxinucleosides are incorporated into chromosomes during S-phase of the cell cycle for about an hour. The retention function of ^{3}H-TdR and ^{3}H-CdR depends on their dosage owing to reparation of DNA damages and labeled cells destruction. The injection of activity in several fractions in 10-15 hours conduces to prolonged DNA-bound tritium retention in tissues and increased radiobiological effect than does a single injection of the same dosage. Dependence of DLM and reciprocal translocations frequency in spermatogonia on the dosage of ^{3}H-TdR is non-linear. The effect per unit activity administered is the greatest at 37 kBq/g (1 μCi/g) and reduces by 2-4 times or by one order of magnitude with increasing dosage up to 0.37 MBq/g and 1.11 MBq/g respectively. By the genetic tests in male mice injection of 37 kBq/g of ^{3}H-TdR and of ^{3}H-CdR is equivalent to 0.2 - 0.4 and 0.05 - 0.01 Gy of gamma-radiation respectively.

The materials of the investigation have been used for developing radiation standards.

A NEW PREPARATION FOR PREVENTION AND TREATMENT OF X-RAY DERMATITIS

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ABSTRACT

Results are presented of development of a new clinical preparation (PP-116), being a synthetic antioxidant, intended for prevention and treatment of skin lesions induced by ionizing and non-ionizing radiations.

The ointment was tested in animal experiments and in clinical practice. Skin was exposed to 60 Gy of β- and soft X-irradiation simulating radiation therapy conditions, as well as to ultra-violet radiation in doses causing skin burns of the second degree.

The experiments showed the decreasing of intensity and duration of skin responses to radiation by more than 50%. It was shown that PP-116 did not protect tumour cells. Its application for treatment of radiation dermatitis allowed one to reduce inflammatory processes, to shorten substantially the period of recovery and to accelerate the process of skin regeneration.

Intravital microscopic and cytomorphologic analyses showed no epidermal and dermal pathologic changes in the recovered skin areas. The protected intracellular membranes, particularly mitochondrial ones, were prevented from destruction.

The preparation entering into inner media is insignificant. Clinical testing of PP-116 on patients exposed to telecurie therapy for various oncologic diseases showed its marked protective effect at skin doses up to 40 Gy and therapeutic effect at the doses up to 60-70 Gy. PP-116 application creates conditions for increasing exposure doses to tumours during radiation therapy and for accelerating the course of treatment. One of the significant merits of the preparation is its analgetic effect, that is psycologically readily accepted by patients.

No side effects or any contraindications for application of the preparation have been detected, and its toxicity is very small.

CHEMICAL RADIOPROTECTION TO BONE MARROW IN WHOLE BODY GAMMA IRRADIATED MICE

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ABSTRACT

Groups of mice received pretreatment before whole body gamma irradiation with (1) saline (2) a combination of suboptimum radioprotecting dose of hydroxytryptophan (HT) and aminoethyl-isothiuronium bromide hydrobromide (AET), (3) combination of the dose of HT and optimum radioprotecting dose of β-mercaptopropionylglycine (MPG) and (4) an optimum radioprotecting dose of AET.

Bone marrow histology has been studied at different time intervals after a lethal dose of irradiation (10 Gy). From 5th day onwards very impressive improvement in bone marrow cellularity has been noted in all drug treated groups. Best regeneration being noted with the HT and AET combination treatment while other treated groups closely followed. Protection to mice bone marrow stem cells by the treatments has also been noted when studied after two days of whole body gamma irradiation (3 Gy, 5 Gy and 10 Gy exposures) using spleen colony forming method.

ESSAIS DE DECONTAMINATION APRES INHALATION PAR DES BABOUINS DU COMPLEXE PLUTONIUM-TRIBUTYLPHOSPHATE. COMPARAISON DTPA-LICAM(C)

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INTRODUCTION

Il a été montré qu'une contamination par inhalation de plutonium sous forme de complexe avec le tributylphosphate (Pu-TBP) était, si elle était massive, plus difficile à traiter par le DTPA que les autres formes transportables de plutonium (Métivier 1983, Stradling 1983). Un nouvel agent chélatant, le LICAM(C), décrit par Durbin et al (1984) semblait prometteur, puisqu'il dissocie in vitro, plus facilement que le DTPA, le complexe Pu-Transferrine formé après injection de Pu-TBP (Métivier et al 1985). Toutefois des études récentes ont montré qu'après traitement par le LICAM(C) le plutonium se déposait plus facilement au niveau rénal, quels que soient sa solubilité et son mode de synthèse (Stradling 1986, Volf 1986, Durbin 1987 Communication personnelle). Depuis, Gérasimo et al (1986) ont proposé des traitements mixtes modifiant les pH urinaires pour diminuer la charge rénale. Dans ce papier nous comparons les traitements par le DTPA, le LICAM(C) et des traitements mixtes après inhalation de ^{239}Pu-TBP.

MATERIELS ET METHODES

Vingt cinq babouins, pesant 4 à 6 kg ont inhalé un aérosol de ^{239}Pu-TBP (solution à 30% dans le dodécane) pendant environ 20 minutes, sous anesthésie (10mg/Kg de chlorhydrate de ketamine). La charge pulmonaire était comprise entre 50 et 400 KBq. Trente minutes après l'inhalation, les animaux reçoivent la première phase du traitement; injection intraveineuse ou aérosol ou perfusion pour les traitements mixtes. Les traitements suivants, effectués pendant 30 jours, sont décrits dans la figure. Pour les traitements par le DTPA nous avons utilisé: 2 doses de CaDTPA (30 et 300 uM/Kg), et l'association Ca et ZnDTPA, successivement ou alternativement à la dose de 300 uM/Kg. Pour les traitements par le LICAM(C) nous avons utilisé les doses de 3 et 30 uM/Kg en variant les modes d'injection. Pour les traitements mixtes, le LICAM(C) n'est utilisé qu'en traitement d'attaque durant la première semaine puis fait place au CaDTPA.

Les animaux sont placés individuellement en cage à métabolisme pour collecte des excrétats. Au sacrifice les différents organes sont prélevés pour analyse, un rein est prélevé pour analyse histologique par microscopie optique et électronique.

Sur chaque singe est effectué avant expérimentation et 1 jour avant sacrifice un bilan permettant d'apprécier l'exploration fonctionnelle rénale (clearance de l'urée, de la créatinine et de l'acide urique), le métabolisme hydrominéral (Ca, P, Cl, K, Na, Fe et le coefficient de saturation de la sidérophiline), le métabolisme protidique, glucidique et lipidique.

RESULTATS

Décontamination: Les résultats sont exprimés soit en pourcentage de la dose inhalée soit en pourcentage par rapport aux valeurs des témoins. Les résultats du tableau 1 montrent une relative inefficacité du traitement par le CaDTPA à la dose de 30 uM/Kg. Cette efficacité peut être accrue en augmentant la dose (300 uM/Kg), mais apparait alors, 1 mois après le

traitement du sang dans les urines. Si on utilise à ces concentrations, une association Ca, ZnDTPA, cette toxicité disparait, mais l'efficacité du traitement diminue. Le tableau 2 fait apparaitre qu'une dose dix fois moindre de LICAM(C) (3 uM/Kg) est aussi efficace que le DTPA au niveau du dépôt osseux. A dose égale (30 uM/Kg) le LICAM(C) est plus efficace que le DTPA, mais on déplore alors, une rétention rénale 5 à 10 fois supérieure aux témoins, suffisante pour entrainer une pathologie. Les traitements mixtes abaissent ces dépots rénaux à des valeurs égales ou supérieures aux témoins mais perdent une partie de l'efficacité du LICAM(C) employé seul.

Biochimie clinique: Lors du traitement par 30 uM/Kg de LICAM(C) l'anomalie la plus caracteristique est representée par l'abaissement des clearances, principalement celle de l'urée, qui ne représente plus qu'un pourcentage variant entre 30 et 70% de la clearance des témoins (Gérasimo et al, à paraître). Le chiffre le plus bas est obtenu avec l'animal traité par la série d'injections IV. Un tel abaissement de clearance est en faveur d'un trouble fonctionnel rénal. Il existe d'autres signes d'insuffisance rénale; augmentation des chlorures, du potassium, diminution du calcium avec augmentation du phosphore, léger abaissement des proteines plasmatiques et du rapport albumine sur globulines, protéinurie moyenne, présence de traces de sang dans les urines. Les animaux traités par le DTPA à 300 uM/Kg ou par perfusion à 30 uM/Kg, ont leur clearance légerement abaissée (60% des témoins) et leurs paramètres du métabolisme hydrominéral et protidique modifiés dans le sens d'une insuffisance rénale comme avec le LICAM(C).

Lors des traitemenyts mixtes, par le système de perfusion il y a restauration partielle des clearances (70% des témoins), le métabolisme protidique est normal, en revanche le métabolisme hydrominéral est aussi modifié que dans le cas des aniamux traités par le LICAM(C) seul (Gérasimo et al. à paraitre)

Histopathologie: L'étude histologique a montré l'induction de lésions tubulaires après administration de LICAM(C). Ces lésions concentrées au niveau des tubes proximaux sont caractérisées par la présence de vacuoles au pole basal des cellules. Le nombre de ces vacuoles augmente en fonction de la quantité de LICAM(C) administré mais leur présence devient inconstante après le traitement mixte (Fritsch et al. à paraitre)

CONCLUSIONS

Cette étude montre qu'en cas de contamination importante par inhalation de complexe Pu-TBP, il sera difficile de décontaminer l'agent par le DTPA. Le LICAM(C) peut apparaitre comme un palliatif, a condition de l'employer uniquement comme traitement d'attaque et sous contrôle des fonctions rénales. Il ne peut en aucun cas, dans l'état actuel de nos connaissances, être employé en routine. Il semble donc raisonnable de poursuivre la recherche d'une autre molécule pour traiter de telles contaminations.

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Ce travail a été fait dans le cadre d'une coopération EULEP, et partiellement financé par un contrat CEC BIO-D-370-81 F.

Figure: Schémas thérapeutiques étudiés.

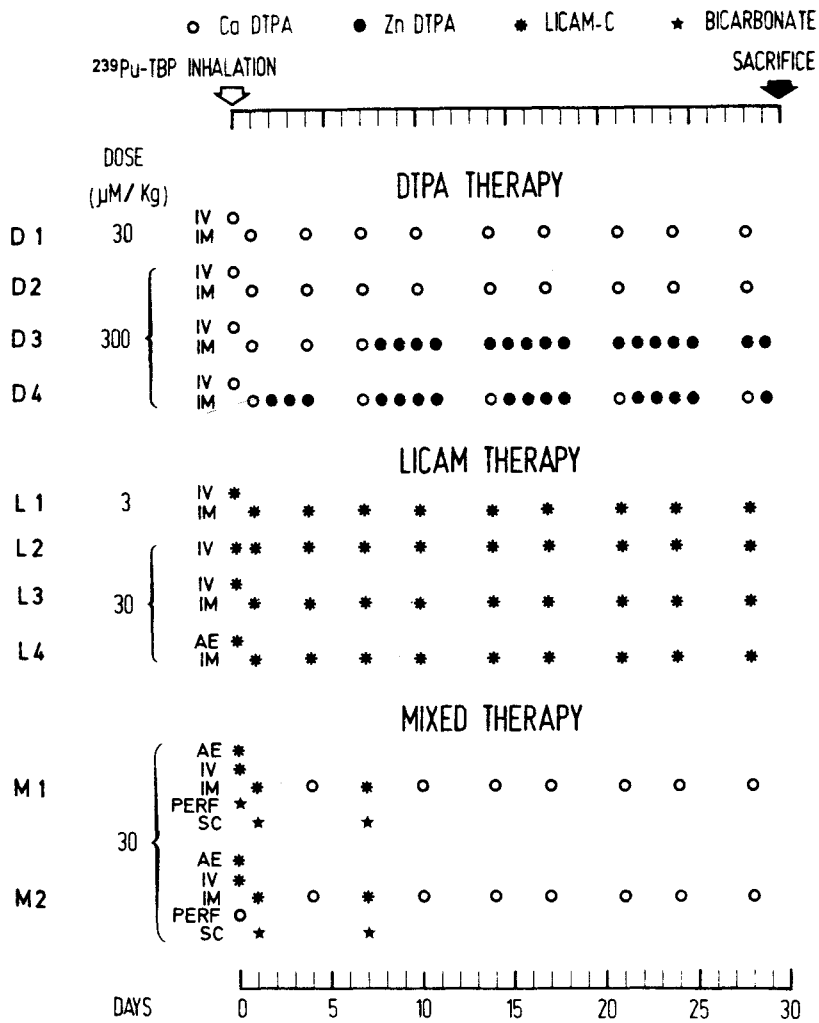

Tableau 1: Effet du DTPA sur la rétention du plutonium-239 chez le singe après inhalation du complexe Pu-tributylphosphate

	Témoins	Ca DTPA 300um/kg	Ca DTPA 30um/kg	Ca-Zn DTPA 300um/kg successif	Ca-Zn DTPA 300um/kg alternatif	DTPA IV + perfusion
			Pourcentage de la dose inhalée			
Lung	69.2	59.7	56.6	63.4	68.9	98.7
Trachea	0.045	0.042	0.17	0.68	0.11	0.0134
Thoracic LN	0.86	0.67	0.56	1.84	0.63	0.0057
Liver	1.1	0.083	0.87	0.103	0.092	0.6
Skeleton	3.9	0.31	1.69	1.11	0.75	3.4
Kidneys	0.047	0.0071	0.011	0.0096	0.0077	0.03
Feces	25	21.9	27.5	21	14.3	8.6
Urines	0.6	17.2	12.5	12.3	17.1	19.1
			Pourcentage par rapport aux témoins			
Lung	100	86	82	92	99	99
Liver	100	7.5	79	9.4	8.4	55
Skeleton	100	7.9	43.3	28.5	19.2	87
Kidneys	100	15.1	23.4	20.4	16.4	64

Tableau 2: Effet du LICAM-C sur la rétention du plutonium-239 chez le singe après inhalation du complexe Pu-tributylphosphate

	30um/kg IV+IM	30um/kg IV	3um/kg IV+IM	30um/kg aér.+IM	mixte: 30um/kg + bicar.	mixte: 30um/kg + DTPA
			Pourcentage de la dose inhalée			
LUNG	62	72	69	56	68	57
Trachea	0.082	0.13	0.022	0.06	0.0052	0.057
Thoraciq LN	0.97	1.19	0.49	0.97	0.47	1.1
Liver	0.70	0.380	1.36	0.93	0.41	0.37
Skeleton	0.34	0.38	1.94	0.75	0.59	0.80
Kidneys	0.22	0.53	0.45	0.23	0.06	0.12
Feces	22.5	15.1	21.5	35.3	24.5	23.3
Urines	12.9	9.6	5.4	5.3	6.8	18.3
			Pourcentage par rapport aux témoins			
Lung	89	91	99	80	98	83
Liver	63.6	73	123	84.5	37.2	33.1
Skeleton	8.7	9.7	49.7	19	15.1	20.6
Kidneys	468	1128	957	489	127.6	244

INDIAN LIFE-TABLE AND RADIO-CARCINOGENIC RISK ESTIMATES

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ABSTRACT

Using ICRP-27, BEIR III and GOFMAN coefficients for the radiation-induced cancer fatalities, age-specific radiocarcinogenic risk factors have been evaluated for the Indian urban male population based on the relative risk projection model. In an Indian urban male cohort population of occupational workers exposed continuously at a uniform rate of 0.01 Sv/y between ages 20 and 65, the life-long cancer risk estimates work out to be about 0.08%, 0.25% and 2.03% respectively. These are considerably less than that would be obtained for typical western populations of the world.

INDIAN LIFE-TABLE PARAMETERS

The Indian life-table (1) and the age-specific cancer mortality patten (2,3) are typical of a developing country with the average life-span and the cancer mortality rates being lower than those among the developed western nations. For the case of the Indian urban males, the surviving population N(A) at any given age A has been fitted in the expression :

$$N(A) = K \exp [\alpha \exp (\beta A)]$$

as suggested by David Maillie (4). Fig.1 gives the data points and the fitted curve with the values of the various constants applicable between ages 5 and 70. The age-specific cancer mortality rates have been taken as a mean for four Indian cities of Bombay, Madras, Nagpur and Poona as given by the National Cancer Registry (2) and the Indian Cancer Society (3). The age-specific cancer mortality rate C(A) at any given age A could be fitted in the expression :

$$C(A) = 0.79 \exp (0.0876\ A) \qquad 10 \leq A \leq 70$$

CANCER RISK COEFFICIENTS

Cancer risk coefficients from three different sources viz. ICRP-27(5), BEIR III (6) and GOFMAN (7) have been adopted in the present calculations as these seem to project respectively a conservative, moderate and generous estimate of the radio-carcinogenic risk in existing literature. For the sake of uniformity of approach and to see clearly the effect of the Indian life-table parameters, risk estimates are made assuming linear hypothesis and relative risk projection model. As ICRP-27 gives only age-specific absoluterisk coefficients, the life-long risk in a typical USA cohort population is first calculated and then expressed as fractional increase in spontaneous cancer mortality per 0.01 Sv for the surviving cohort from the age at exposure. BEIR III report also does not give directly any relative risk coefficients; however working back on the results given for relative risk projections, reasonable estimates of the coefficients could be made for the cases of 0, 20, 35 and 50 years of age at exposure on the same lines as suggested by David Maillie (4). The GOFMAN risk

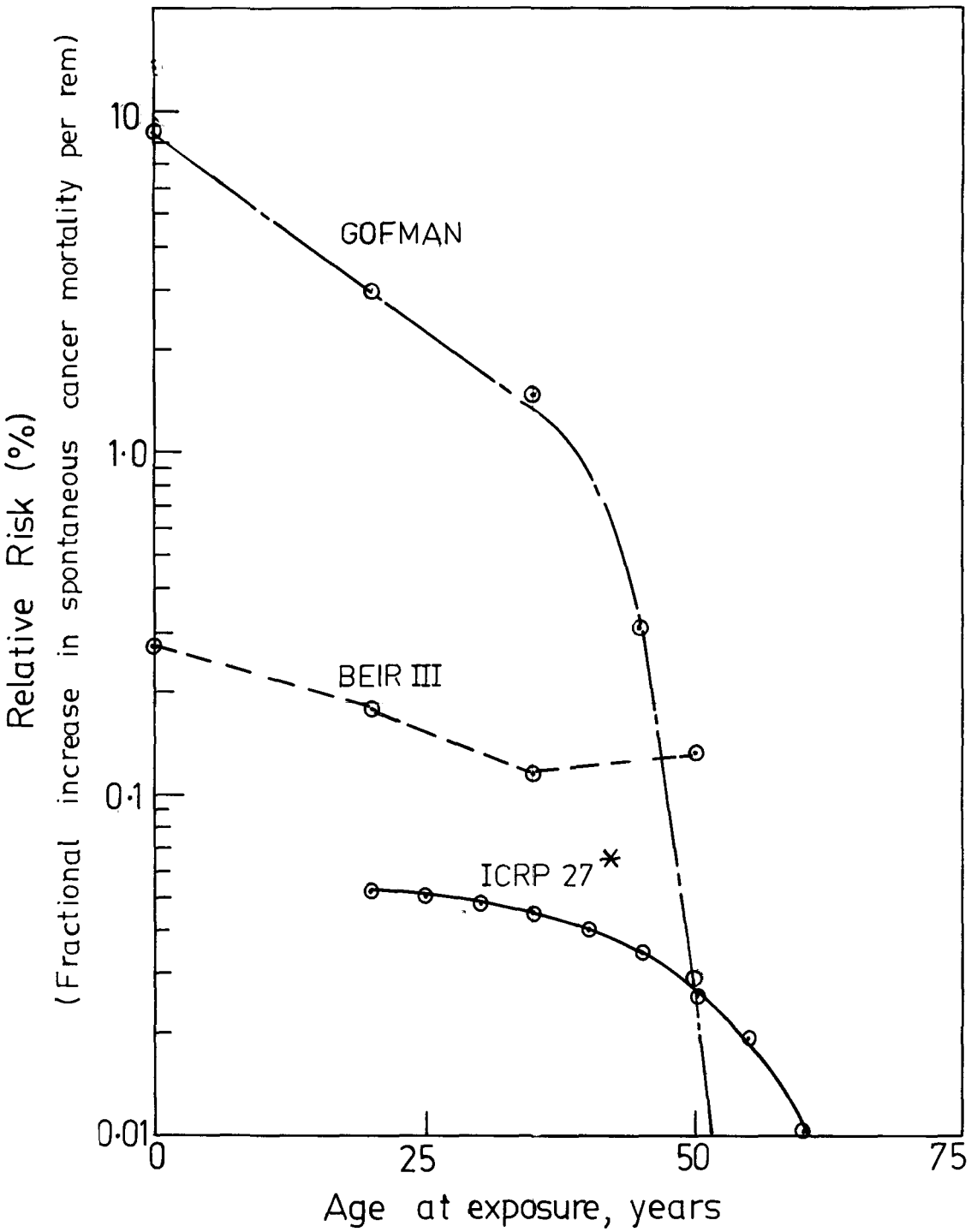

FIG. 2 - AGE SPECIFIC RELATIVE RISK COEFFICIENTS FOR MALES ESTIMATED FROM DIFFERENT LITERATURE SOURCES (* see text)

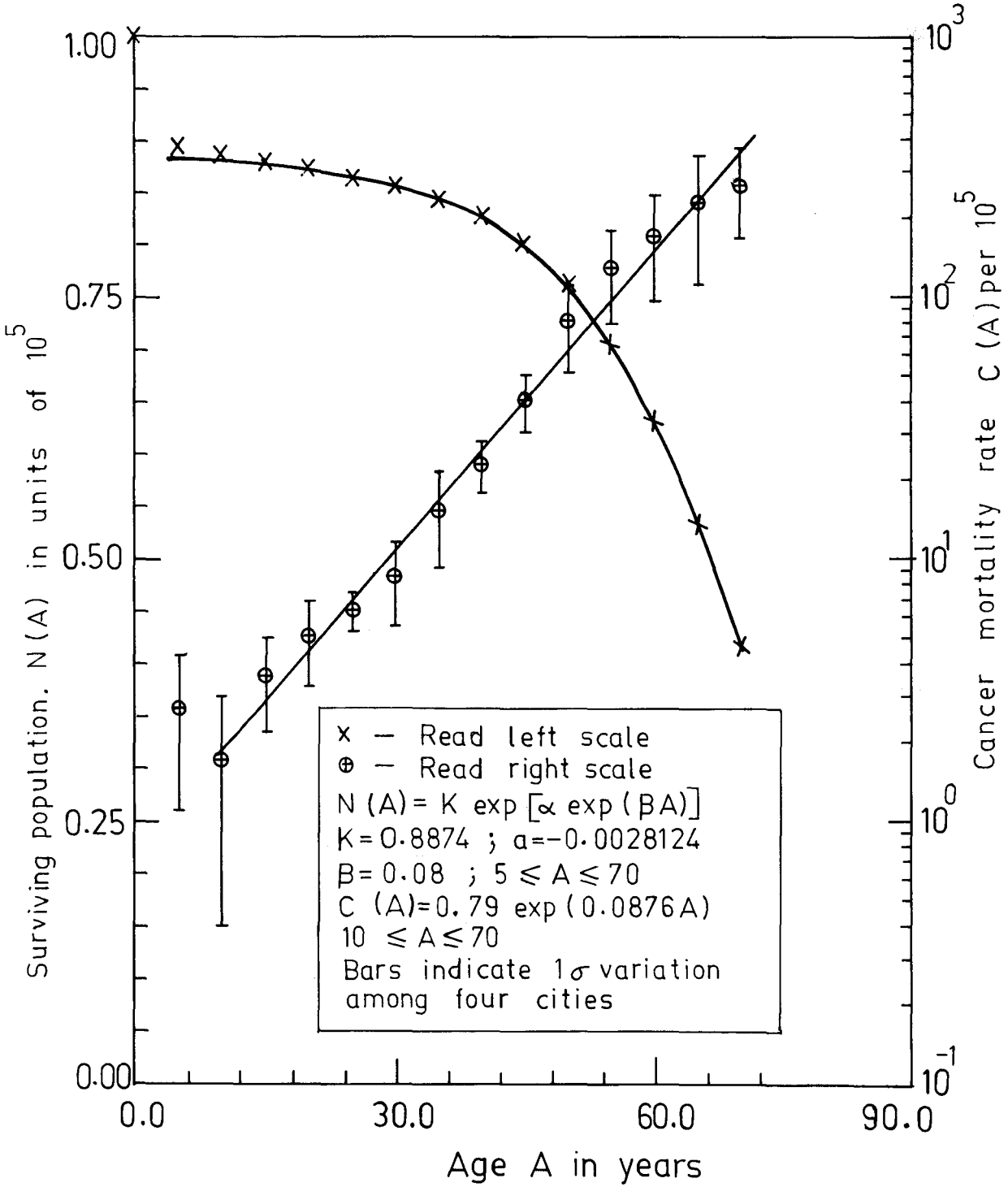

FIG.1,- POPULATION COHORT LIFE-TABLE AND AGE SPECIFIC CANCER MORTALITY RATES FOR INDIAN URBAN MALES

coefficients have been modified to the Indian life-table as per procedures given in (7). The age-specific relative risk coefficients thus obtained from the above three data sources are given in Fig.2.

RESULTS

Using the factors given in Fig.2, sample calculations have been made to obtain risk estimates for an assumed occupational exposure history for a population cohort starting from age 20. Although an Indian worker retires at about 60 years of age, for ready international comparison purposes, exposures have been considered upto age 65; the latter exposures however do not increase the cancer risk significantly. The results are presented in Table 1.

Table 1: Cancer risk estimates for an occupational exposure history of 0.01 Sv/y between ages 20 and 65

Population Cohort	Life long cancer risk estimates		
	ICRP 27	BEIR III	GOFMAN
U.S.A Males	0.58%	1.10%	5.65%
India Urban Males	0.08%	0.25%	2.22%

It is obvious that the relative risk projection model would predict far less cancer risk to the Indian occupational worker.

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ASSESSMENT OF DOSE-TIME-EFFECT SURFACES FOR SOMATIC LATE EFFECTS AFTER LOW DOSE IRRADIATION

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ABSTRACT

Awaiting new data from RERF, an attempt was made to estimate the somatic risks of low doses of radiation for members of the German public. The new estimate follows closely the approach used in deriving the values published in the NIH Radioepidemiological Tables. The lifetime risk factors thus calculated are significantly larger than the estimates presented in ICRP-report 26 and are consistent with estimates recently made in the ongoing reactor safety study of NUREG. The increase in lifetime risks is mainly due to the introduction of a new time projection model which appears to be more consistent with the epidemiological data for many late effect end points than the constant absolute risk model previously employed. Results from a study in which epidemiological data were stochastically simulated with a computer indicate that the shapes of actual dose effect curves might be significantly more influenced by pure chance than by the dose dependency of underlying biological mechanisms.

1. INTRODUCTION

Results of probabilistic reactor accident consequence studies and the public concern about biological effects of low radiation doses after the Chernobyl reactor accident demonstrate the need for realistic estimates of dose-time-response surfaces for late somatic radiation effects for members of the public (i.e. for both genders and all age groups). Therefore, the epidemiological data available in 1986 have been re-evaluated along the lines employed in the American Radioepidemiological Tables (1); the derived lifetime risk factors are compared to previous estimates. Major problems still exist in the assumptions about the time projections, i.e. the number of cases to be observed in the future in study cohorts, and the shape of the dose-response curve at low doses. That is why these two important aspects in radiation risk analysis will be discussed in detail.

2. SHAPE OF DOSE-EFFECT RELATIONSHIPS

Dose-time-response surfaces are the basic subjects of radiation risk assessment; they depend on the effect under consideration, time after exposure, dose rate, dose fractionation, radiation quality, gender, ethnic groups, age at exposure, living habits, occupation, etc.. In addition, the characteristics of early radiation responses need not reflect themselves in those observed at later times, i.e. a quadratic dependency of primary effects on dose could very well be converted to a threshold type or linear dependency of later effects by subsequent reactions to this primary insult. This fact makes it rather difficult to draw inferences from the shape of dose-response curves for radiation biological effects occurring at early times to the shapes of dose-effect relationships observed at later times, possibly even decades later.

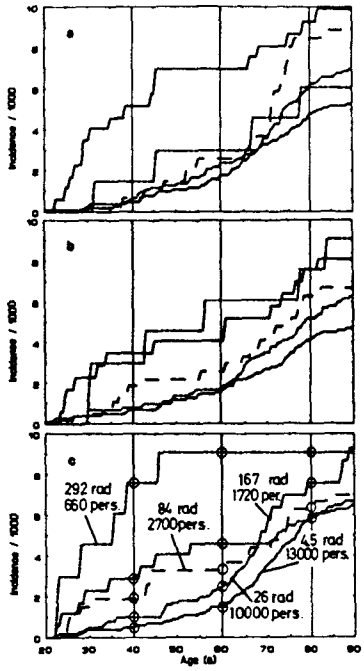

Fig. 1: Randomly chosen results of simulated "cohort studies"

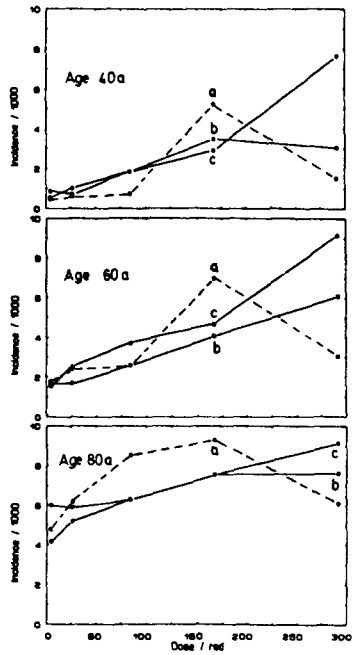

Fig.2: Dose-response relationships for randomly chosen results of simulated "cohort studies"

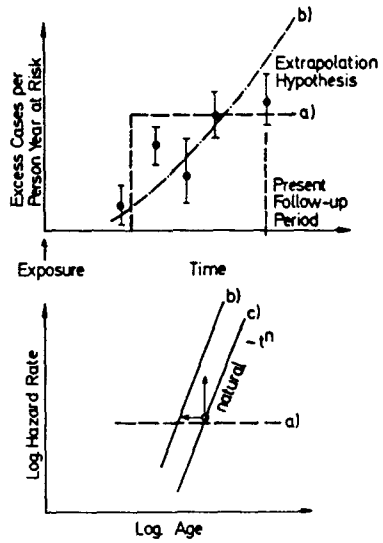

Fig. 3: Methods for the temporal extrapolation of observed rates

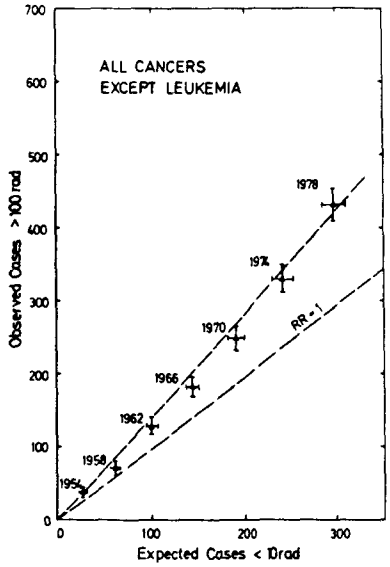

Fig. 4: Number of cases of all cancers except leukemia observed among the atomic bomb survivors in a highly exposed cohort versus the number of cases found in the non-exposed cohort

Even if this would not be the case, it is rather difficult to draw any significant conclusion as to the shape of dose-effect relationships from realistic epidemiological data. That is because of the usually small numbers of observed excess cases, the usually small number of exposed persons and the fact that the end points considered (e.g. cancer or leukemia) also occur rather frequently and with large temporal and regional variations as causes of mortality. This may be demonstrated with the help of a simple Monte Carlo simulation of epidemiological data on a computer. For simplicity only the fates of young German men with twenty years of age at time of "exposure" was followed in time with respect to the mortality from leukemia; the baseline risk was taken from German statistical tables and the mean doses and cohort sizes were adapted from the Hiroshima and Nagasaki life span study (indicated in fig. 1c). Figs. 1a, b and c give three arbitrary stochastic samples of the relative number of deaths from leukemia that might have been observed in the real data as a function of time if there were a strictly linear dose-effect relationship for this endpoint with the ICRP-Report 26 (2) lifetime risk factor, a 2 year latency period and a cinstant risk expressed over the period of 25 years. The differences in the three data sets are of purely stochastic nature. A cross section with respect to dose at age 40, i.e. 20 years after irradiation, would give the data shown in fig. 2a, cross sections at age 60 and at age 80 are shown in figs. 2b and c. Evidently a wide range of "likely" dose-effect curves could subjectively be drawn and confirmed by statistical tests (from a threshold type curve to a quadratic curve with "exponential correction for cell killing") although it was a strictly linear dose-effect model which was used in this simulation. After this experience the statistical insignificance of the differences between the shapes of the leukemia dose effect curves for the survivors in Nagasaki in the life span study group resp. the tumor registry will be realized even without stringent statistical tests. However, employing different assumptions, other analysts might well have interpreted the varying differently, although this would not be justified from objective epidemiological points of view.

3. TIME PROJECTION

In study cohorts exposed to elevated levels of radiation persons are often still alive at the time of reporting and assumptions about their future fates must be made when estimating lifetime risks. In the field of radiation risk analysis the assumption was often made in the past and found to be consistent with available epidemiological data, that after an exposure to an elevated level of ionising radiation, there is a constant increased risk of mortality due to cancer or leukemia after a certain latency period during which no excess cases could be observed (absolute or additive risk model), In other absolute risk models this constant risk decreased to zero after an extended risk expression period (curve a in fig. 3). The increased length of follow-up time, however, brought epidemiological data which appear to be more consistent with the assumption of a relative risk model (curve b in fig. 3), i.e. the irradiation possibly increases the pre-existing baseline risk (curve c in fig. 3), which is usually a high power of age t, by an amount depending on dose (lower panel of fig. 3). From principle, it is not possible to descriminate whether this is an increase in time (time acceleration model) or in the number of cases. Evidence for the plausability of the assumption of a relative risk time extrapolation model can be seen in the data of the atomic bomb survivors of Hiroshima and Nagasaki (3) for the sum of all cancers except leukemia (fig. 4) as well as in the lung cancer cases; a straight line in this observed-expected cases plot indicates the consistency with a relative risk model, i.e. the

larger number of cases in the highly exposed population occur in the same time pattern as the expected cases do. Most naturally occuring cancers steeply increase in frequency with increasing age. It is then evident, that the assumption of a relative risk model will significantly increase the number of cases to be expected in the future in an observed study cohort as compared to the assumption of a constant absolute risk model. However, it should be clearly pointed out that even in a relative risk model, the relative increase of the base-line risk might well go down with time after exposure for several reasons not to be discussed here.

4. LIFETIME RISK ASSESSMENT

From the fact that the base lines steeply increases with age, it must be expected that the change from an absolute to a relative time projection model will affect in particular the estimates made for the younger age groups at exposure. This is in fact the case as can be seen in the risks for radiation induced breast cancer as a function of age and time after irradiation in both models (table 1). This estimate is based on the same procedure of transporting the absolute excess risk observed among the A-bomb survivers from the Japanese women to German women and then re-calculating the national relative risk. Unfortunately, there is presently no very convincing argument available neither in favor of this way of inter-national risk extrapolation nor against it.

When the same underlying assumptions as in ref (1) are made for the German population, the new risk factors as indicated in table 2 were derived (A = absolute risk model, R = relative risk model). Also given, for comparison, are the data of ref. (4) for the US-population where the "central" estimate essentially assumes a dose rate reduction factor of 0.3 as compared to the upper (i.e. linear) estimate. The GSF-estimate for the assumption of a linear dose effect relationship is in good agreement with the American "upper" - (i.e. linear) estimate; our linear - quadratic model using a dose rate reduction factor (DRF) 0.4 as assumed in the NI-report (1) generally lies somewhat above the "central" - estimate of ref. (4) employing a DRF of only 0.3. From the existing data for carcinogenesis by radiation, it is presently difficult to decide on the likelihood of the existence of any DRF for this particular endpoint.

These data for organ specific radiation risk factors are considerably above previous estimates of the ICRP (2). The main reason is the application of a relative risk time projection model suggested by more recent epidemiological data. It remains to be seen whether this new way of extrapolation will be confirmed by future data and to what extent the new dosimetry presently incorporated at RERF will modify these risk estimates.

Table 1
Risks of breast cancer induced by radiation as a function of age and time after exposure

Age	absolute Model Time since irradiation 10a	30a	∞	relative Model 10a	30a	∞
0- 9	23	98	259	-	45	1770
10-19	45	196	445	1	120	945
20-29	29	125	239	6	95	294
30-39	29	121	192	15	106	198
40-49	8	30	39	6	28	38
50 +	4	11	12	4	11	12

Table 2
Somatic lifetime risks per 10^6 per rem, (low-LET radiation, doserate < 5 rem/d)

Organ	Model	NUREG central	NUREG upper	GSF (LQ)	GSF (L)
Bone marrow	A	14	48	21	52
Bone	A	1	2	1	1
Breast*	R	60	87	80	80
Lungs	R	20	138	36	90
GI-Tract+	R	57	189	90	224
Thyroid	A/R	7	7	17	17
Others	R	29	96	15	38
Total		188	567	260	502

in utero (1 %): additionally 3 cases of leukemia, 3 other cancers

* including men
\+ Stomach, Colon, Liver, Pancreas

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   Health Effects Model for Nuclear Power Plant Accident Consequence Analysis NUREG ICR-4214, Boston 1985

ABSTRACT

KINETICS OF TRANSPLACENTAL TRANSFER OF SELENIUM

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This study was designed to investigate the influence of the quantity of selenium intake by pregnant rats on the rate and amount of selenium absorbed by the fetus.

Sodium selenate, tagged with Se-75, was added to the drinking water of pregnant rats at 3 different concentrations, and the rats were serially sacrificed during the gestation period and the selenium in the fetuses was measured. Gross selenium uptake was measured by sequential whole body counting of the rats, and elimination was determined by radiometric analysis of the excreta. From these data, we constructed a three compartment model for the kinetics of transplacental transfer of selenium.

The distribution pattern of selenium was found to be consistant with other reports, with most of the selenium being in the liver, kidneys, and blood. The kinetic parameters of the excreta and the transfer from the fetus to the mother's blood was found to be dependent on the selenium intake level. Other kinetic parameters were found to be independent of the selenium intake level.

RESPONSE OF MATERNAL IMMUNE CELLS TO IRRADIATION OF MOUSE EMBRYOS

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INTRODUCTION

This work began as an attempt to explain the paradox of pregnancy - the survival and growth of the semi-allogeneic embryo in an immunologically hostile environment. We[3,4] rediscovered the work of Brewer[2] who in 1937 had observed the phagocytosis of maternal blood cells including lymphocytes by cells of the trophoblast. In 1982 and 1983 we reported the tracing of quinacrine labelled maternal leukocytes (WBC) in maternal, placental and embryonic mouse tissues by fluorescence microscopy. We found that cells in the placenta phagocytose labelled WBC, so that after 1-2 hours the labelled nuclear DNA is found as brightly fluorescing particles in the cytoplasm of the phagocytes with no evidence of it in the nuclei. Identical cells were observed in slide preparations of embryos which had been carefully separated from their placentas.

We also found a small population of intact labelled lymphocytes, clearly maternal in origin, in the embryos. This seems to be another paradox- placental phagocytes are observed to be phagocytosing maternal WBC in the placenta and embryo, but there are also free maternal cells in the placenta and embryo. A theoretical explanation is that maternal lymphocytes alloreactive against the embryo will attempt to react with placental cells and in the process be phagocytosed, while other maternal cells will be able to enter the embryo where they could have a surveillance function, removing dead or mutant embryonic cells.

To test this theory we carried out the series of experiments reported here.

MATERIALS AND METHODS

These fall under 3 headings: (1) the quinacrine labelling method referred to above; (2) the use of the histochemical α-naphthyl butyrate esterase (α-NBE) procedure (Markovic, 1987[6]) to identify maternal lymphocytes and monocytes in 11 day-post conception mouse embryos (BALB/c); and (3) the creation of artificial embryonic damage with varying doses of γ-radiation.

(1) Quinacrine labelling. Maternal whole blood (approx. 0.2ml) was labelled with 0.1ml quinacrine solution (9.4 mg/10ml of normal saline). After standing for a few minutes, the blood sample was washed 3X with normal saline to remove quinacrine unbound to the WBC nuclei. The blood was then reinjected into the original donor. Two hours or more later the mouse was sacrificed and various tissues, placentas and embryos

were examined. Embryonic tissues could be dissociated on a microscope slide using the end of another slide, which was then used to make a smear as in making a blood slide.

(2) The α-NBE procedure (modified from Sigma) was used on embryonic tissues: an air dried slide preparation of embryo was fixed for 5 minutes in glutaraldehyde below -10°C, followed by incubation for 1 hour at 37°C with α-NBE reagent. The rinsing step was then repeated, followed by 4 minutes of staining with methylene blue. After a third rinse, the slides could be mounted in aquamount.

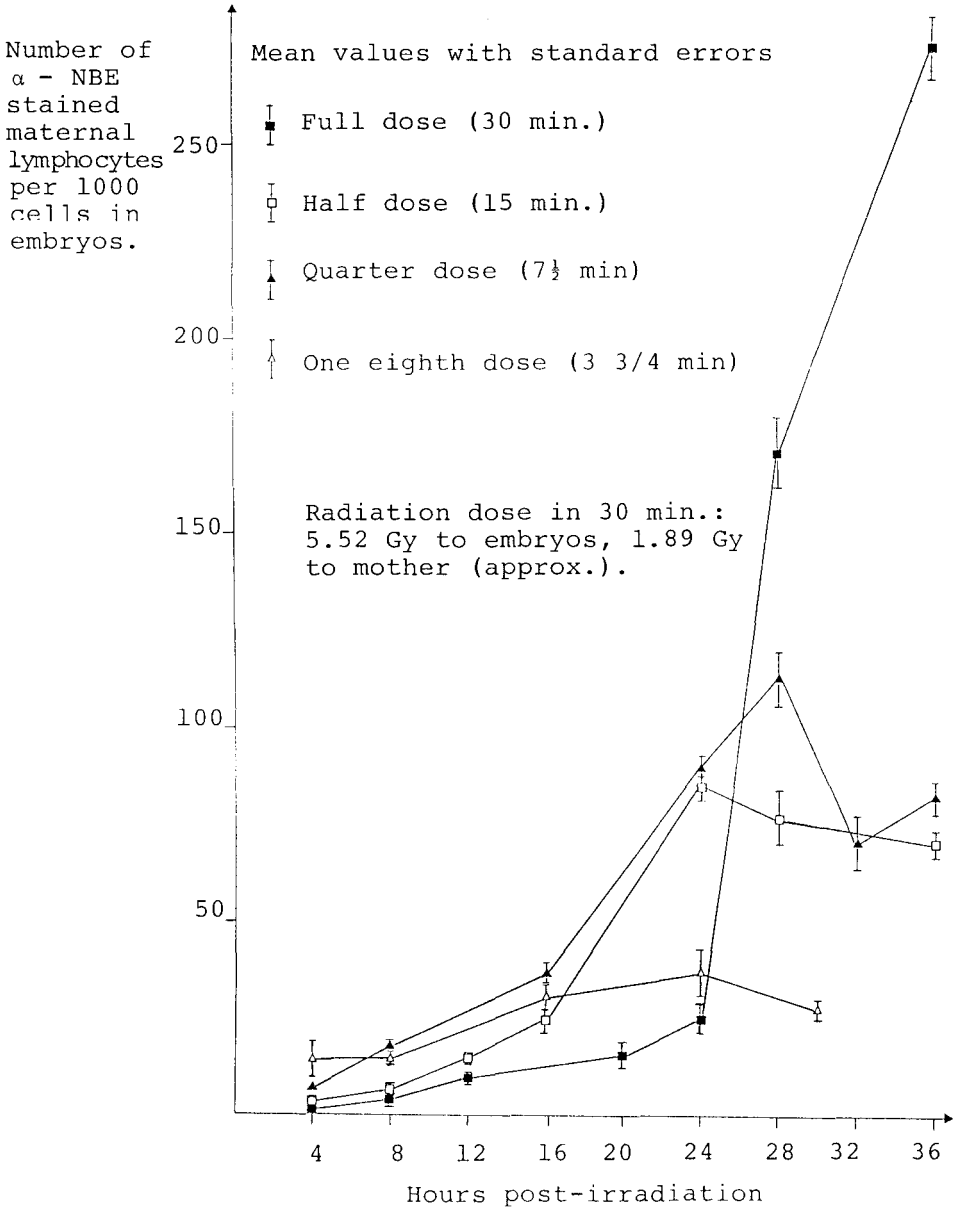
(3) γ-irradiation. A method was devised to lethally irradiate the embryos of a pregnant mouse while delivering a lesser dose to the mother. A Cobalt-60 source of γ-rays was used. The mother was shielded by 30 mm of lead, this being the maximum shielding possible because of the construction of the housing of the source. Two lead radiation chambers were prepared from molten lead cast to fit comfortably into the hydraulically lowered cylinders of the source. A chamber with a diameter just large enough to house a pregnant mouse was moulded lengthwise through each cylinder, from one end almost to the other end. A hole 3 mm in diameter was drilled at the closed end of the cylinder to provide air for the animal while in the chamber. A 10 mm diameter hole was drilled at right angles to the midline of the cylinder at the position where the abdomen of the pregnant mouse, and hence the embryos, would be located and in a position to be aligned with the radiation beam. The mouse was held in position after entering the chamber by inserting a plug of foam plastic. Irradiation initially was for 30 minutes, a time chosen because it was shown to give a lethal dose to all embryos. The radiation dose was determined by Thermoluminescence Dosimetry (TLD) and found to be maximally 5.52 Gy/30 min to the embryos. Most maternal tissues received in that time only 1.89 Gy. Dosages of radiation were subsequently varied by serial 2-fold reductions of radiation time down to a 1/8th dose (3 3/4 minutes, 0.69 Gy to embryos, 0.24 Gy to mother).

RESULTS

(1) Quinacrine labelling. This method is indirect in determining the population of maternal cells in the embryo. About 10% of the maternal blood was labelled in each experiment. An estimate could be made of the number of WBC this entailed. However, the precise division between circulating and tissue populations of WBC could not be ascertained, and more importantly perhaps, the number of maternal WBC phagocytosed while entering the embryo was not known. Thus this was a qualitative rather than quantitative study. It was seen that the numbers of labelled lymphocytes in the embryo increased after irradiation of the pregnant mice (Markovic, 1984[5]).

(2) α-NBE staining. The figure shows the results of examining 205 embryos. The lymphocyte counts represent direct and total counts of maternal cells (the embryo develops its own mature lymphocytes (α-NBE +ve) after day 15 of pregnancy (Bannerman, 1983[1])

Figure: Maternal lymphocyte numbers in embryos after various doses of radiation, and at various times post-irradiation.



DISCUSSION

The preliminary experiments using quinacrine labelling confirmed previous work and expanded our understanding of the processes under observation. (1) The placental phagocytes are very efficient at removing maternal cells. If these maternal cells are those with allogeneic reactivity to the embryo it is easy to understand why embryos are not rejected. (2) A uniform population of labelled maternal lymphocytes was found in every embryo examined.

(3) Placental phagocytes containing fluorescent particles in the cytoplasm were found in every embryo.

The second series of experiments with α-NBE were more informative. Direct counts of numbers of maternal lymphocytes in the embryos could be made. What we had postulated as the mechanism of reabsorption in naturally reabsorbing embryos was clearly demonstrated with the higher levels of radiation. The embryos contained a massive invasion of maternal lymphocytes at 36 hours (> 25% of total cells in the embryo) and the relationship of this value to the normal population of lymphocytes in the embryos could be seen. Lowering the dose of radiation to a level where most embryos would survive showed two things: (1) the maternal influx of lymphocytes occurred earlier with the lower doses, thus negating the objection that lymphocytes were entering the embryos only because of placental damage; (2) the peak values appeared earlier, 24-32 hours after irradiation, than occurred with the lethal dose of irradiation.

CONCLUSION

We have in these experiments a valuable method for the study of irradiation mechanisms as they affect pregnancy. Much more work can be done. The dose-response relationships are clear. Recent experiments have demonstrated comparable results using toxic chemicals, and hence the method may have value as a screening method in toxicology.

The theoretical implications are also significant. The concept can be developed as follows: artificial damage to, or natural death of, embryos leads to a stimulation of some of the non-alloreactive lymphocytes in the embryos. These lymphocytes send messages via lymphokines to the placenta where more maternal leukocytes will enter the embryo. With this influx the embryo will either be purged of damaged cells or will be destroyed and reabsorbed. Thus there is an efficient tissue surveillance mechanism operating in the otherwise immunologically inert embryos.

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LATE EFFECTS OF EXTERNAL GAMMA IRRADIATION AT LOW DOSE-RATE IN THE FOETUS AND YOUNG RAT

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ABSTRACT

LATE EFFECTS OF EXTERNAL GAMMA IRRADIATION AT LOW DOSE-RATE IN THE FOETUS AND YOUNG RAT.

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In order to estimate the specific radiation sensitivity of embryos and neonates, notely for carcinogenic risk, we examined the consequences of chronic irradiation in Sprague-Dawley rats, upon central nervous system and gonads.

Pregnant mothers and eventually litter mates were exposed to gamma rays of Cobalt 60, at the same dose rate, 5 hours a day, 5 days a week, from the 8th day of pregnancy up to the delivery for group I, up to weaning for group II, up to the 135th day of extra-uterine life for group III; doses varied about from 2.5 to 15 Gy.

400 exposed newborns, male and female, together with mothers and controls were followed for major developmental changes and tumor induction, until their natural death.

After two years, we observed in exposed animals that brain weight was decreased by 25%; by the same time gonads weights were decreased by 85% and all animals were sterile. Tumor incidences in brain and gonads exceeded largely those of controls.

The results substantiate the concept that developing rat brain and gonads are tissues at risk for stochastic and non stochastic effects.

RADIOBIOLOGICAL ARGUMENTS FOR A LINEAR DOSE-EFFECT RELATIONSHIP OF STOCHASTIC EFFECTS AT LOW DOSES

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INTRODUCTION

In the development of our theoretical approach to the analysis of radiation biological effects (1) we have made the association between radiation induced DNA double strand breaks (d.s.b.) and biological effects, such as cell death, mutations and chromosomal aberrations. We further assume that stochastic effects result directly from a mutation in one cell. On the basis of these assumptions we present data which supports the theory and implies that the induction of d.s.b. and thus cellular effects and stochastic effects must be linear with dose at low doses.

THEORETICAL ANALYSIS

In the theory the number of N of DNA d.s.b. per cell induced by a dose D of radiation is defined by

$$N = \alpha D + \beta D^2 \qquad [1]$$

where α is the probability per unit dose that a d.s.b. is induced in one radiation event and β is the probability per unit dose squared that two independently induced single strand breaks combine to form a d.s.b.

If p is the probability that a double strand break leads to cell reproductive death then cell survival

$$S = \exp\,[-p(\alpha D + \beta D^2)] \qquad [2]$$

Equations for mutation frequency (M) and chromosome aberration yield Y can also be derived as

$$M = 1 - \exp\,[-q(\alpha D + \beta D^2)] \simeq q(\alpha D + \beta D^2) \qquad [3]$$

and $$Y = c(\alpha D + \beta D^2). \qquad [4]$$

where q is the probability that a d.s.b. gives rise to a mutation and c the probability that a d.s.b. gives an aberration.
Combinations of equations 2 with equations 1, 3 and 4 give rise to predicted correlations between the biological effects, namely

$$\ln S = -pN;\ \ln S = -\frac{p}{q} M;\ \ln S = -\frac{p}{c} Y.$$

In figures 1 and 2 we present examples of these predicted linear correlations analysed from data with distinctly non-linear dose effect relationships. The figures demonstrate the direct correlation between DNA d.s.b. and cell survival and between cell

survival and mutation frequency. We conclude that these analyses are compatible with the basic assumption of our theory that DNA d.s.b. are the crucial lesions leading to the various effects.

The linear-quadratic form of dose-effect relationships has been well established (see 1) for cell survival, mutation frequency and aberration yield from the analysis of data mainly derived using relatively high doses. There are comparatively few data at low doses but figure 3 presents data for mutation induction in Tradescantia measured for acute and chronic exposures down to 30 mGy which demonstrate the linear-quadratic curve for acute exposure and the linear curve for chronic exposure. Recently, data on aberration induction in human lymphocytes at low doses has come available (2) which is not incompatible with a linear dose relationship through the zero dose point.

Theoretically, we have made calculations, based on track structure simulations, on the variation of the α coefficient as a function of radiation quality (3). These calculations (fig.4) indicate that even for sparsely ionizing radiations 'α' does not become zero because low energy electrons, which have short densely ionizing tracks, contribute to the dose from these radiations.

DISCUSSION AND CONCLUSIONS

The theoretical analysis of the data presented here would appear to support the following conclusions:

1. The crucial molecular lesion which can lead to cell reproductive death or mutations or chromosome aberrations is the DNA double strand break;
2. the dose effect relationship for the induction of DNA double strand breaks appears to be linear-quadratic in general and this is also the shape of the dose effect relationships for cell death, mutation frequency and aberration yield;
3. at low doses the dose-effect relationship is proportional with dose for both acute and chronic exposures and only at higher doses does the acute exposure curve acquires the quadratic function and bend away from the linear chronic exposure curve;
4. the 'α' coefficient for the induction of d.s.b. in one radiation event, which requires two energy depositions close to the DNA with a separation of about 2 nm, defines the linear component of the dose-effect relationship and is never zero for ionizing radiation;
5. the 'α' coefficient varies in absolute value as a function of radiation quality but the densely ionizing tracks caused by low energy electrons ensure that even for sparsely ionizing radiation 'α' is not zero;
6. the ratio of the 'α' coefficients for two types of radiation can be used to define a maximum RBE at low dose for the radiations and this maximum RBE should be related to the Quality Factor.

Two points should be mentioned, namely the repair of DNA d.s.b. is known to occur and we have attributed this to PLD repair in the theory. It is not known to what extent this repair is genetically pure and it seems possible that some mutations and aberrations arise from mis-repair of d.s.b. The second point is that a question still remains about the direct association between

mutation and the induction of cancer (4) which is relevant to the radiological protection philosophy.

ACKNOWLEDGEMENT

This paper is publication number 2438 of the Radiation Protection Programme of the Commission of the European Communities. The work is also supported by Ministries of the Dutch Government.

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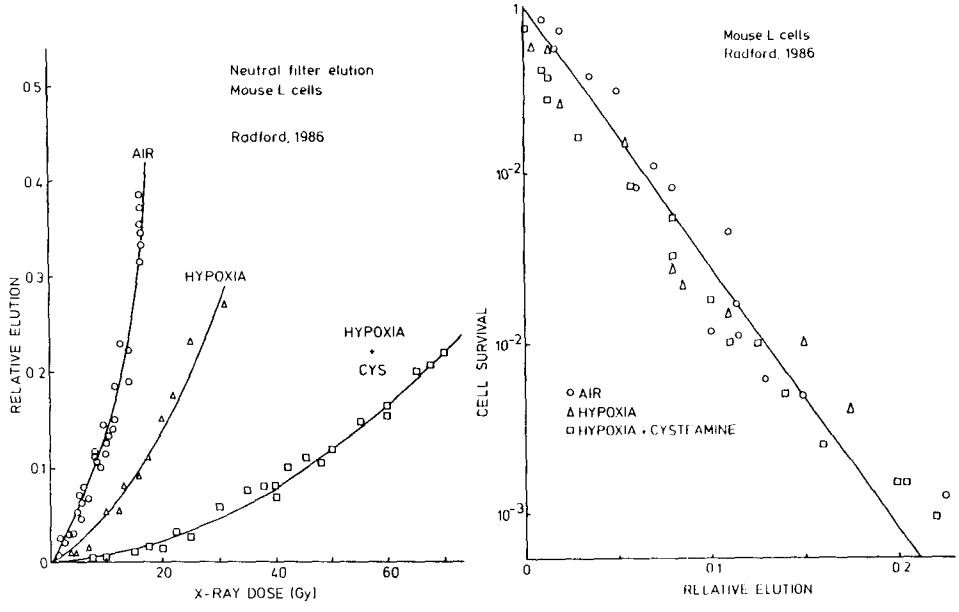

Fig.1. Correlation between cell survival and DNA double strand breaks (neutral elusion) (5, 6).

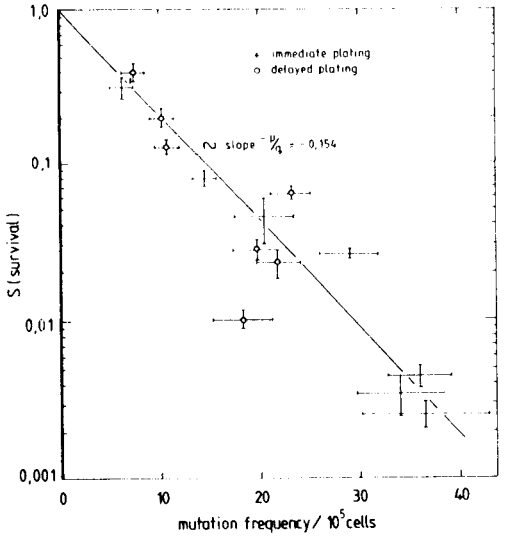

Fig.2. Correlation between cell survival and mutation frequency (7)

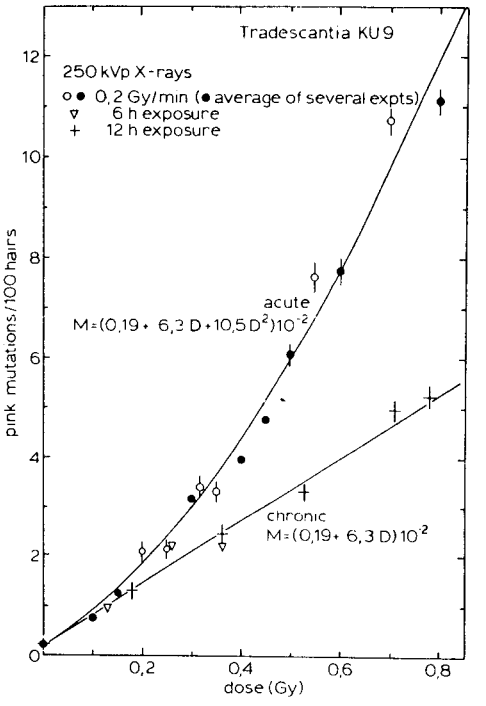

Fig.3. Dose-effect relationship for mutations in Tradescantia for acute and chronic exposure.

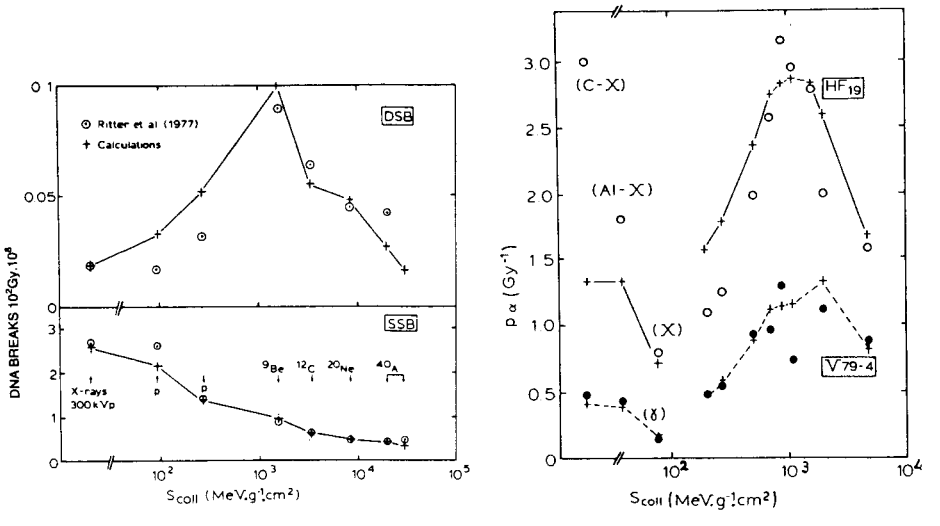

Fig.4. The production of DNA double strand breaks and the 'pα' coefficient for cell survival as a function of radiation quality. Lines are calculated (ref.3), data from refs. 8, 9.

INFLUENCE OF ENERGY METABOLISM AND MITOXANTRONE ON REPAIR PROCESSES FOLLOWING X-IRRADIATION IN CaNT TUMOURS

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Abstract

The response of the levels of adenosine-5'-triphosphate (ATP) and the specific activity of NADP-isocitrate dehydrogenase (NADP-ICDH) following 10 Gy x-irradiation in implanted CaNT tumours in the sternum of CBA mice is presented. Tumour ATP and NADP-ICDH levels appear to have crucial roles in repair processes after radiation damage. Mitoxantrone as an arresting drug is shown in this work to interfere with ATP yields in these tumours, hence it can modify cellular energy status and therefore be an effective way to modulate cellular repair processes after irradiation.

Introduction

Adenosine-5'-triphosphate appears to play a crucial role in repair mechanisms, including cellular repair following radiation damage. A very important and relevant repair process is that of DNA. The relationship of ATP to DNA repair has been studied by several workers, who have shown that some of the enzyme steps require ATP (1). Also, it has been suggested by some authors that ATP is associated with cell membrane repairs following ionizing radiation (2,3).

The oxidative reactions of the hexose monophosphate shunt and extramitochondrial NADP-isocitrate dehydrogenase reactions appear to be very important routes for generating NADPH, in addition to malic enzyme (NADP malate dehydrogenase) in the biosynthesis of fatty acids. These processes depend on the NADP/NADPH ratio which is controlled in part by NADP-ICDH (4). There is presently a great deal of interest in combining ionizing radiation and anticancer drugs in the laboratory, with the aim of providing new strategies in the clinic. Neri et al (5) reported that mitoxantrone inhibited ATP production in rat heart slices. Thus this drug theoretically provides a means of inhibiting the increased energy production after irradiation, which may lead to enhanced tumour cell kill. Also mitoxantrone inhibits RNA and DNA synthesis (6, 7).

The purpose of this work is to investigate ATP content in experimental rodent tumours and the levels of NADP-ICDH related to energy status following x-irradiation solely, or after treatment with mitoxantrone as an arresting anticancer drug.

Materials and methods

Male CBA mice 2 to 3 months old were used. The tumour was maintained by several passage by innoculation of a tumour cell suspension subcutaneously into the sternum area of CBA mice. The CaNT tumour is a poorly differentiated adenocarcinoma and does not show detectable immunogenicity (8). The volume of tumour used in the investigation was between 150 and 250 mm^3. Mice were immobilized during irradiation by an acrylic restraining jig without anaesthesia

and were shielded with 3 mm thick lead, except for the tumour region. A Philips RT 100 x-ray set was used for the x-irradiations, (100 KVp, 8 mA), giving an HVL of 3mm Al and a dose rate of 7.36 Gy min^{-1} at the centre of the field and a total dose of 10 Gy. To prepare samples for intracellular ATP determinations, the tumour was cut away and immediately dropped into liquid N_2. Tumours were quickly weighed, then pulverized in a mortar with frequent additions of liquid N_2. One mℓ $HClO_4$ (6% w/v) was added, ground with the tissue, the mixture was allowed to become fluid and was homogeneized in a Potter-type glass homogeneizer. The homogenate was centrifuged at 17500g for 20 min at 4°C, the supernatant removed and the pH adjusted to 7.5 with 5M K_2CO_3. This was contrifuged at 17500g for 20 min and the supernatant was used for ATP determinations, according to the enzymatic method of Lamprecht and Trautschold (9).

Tumour samples for NADP-ICDH assay were prepared after sacrifice by cervical dislocation. The tumour was excised rapidly, tissue was blotted, weighed and homogeneized in a Potter-type glass homogenizer with ice-cold physiological saline 1:10 (w/v) which was 0.66 mM in EDTA. The homogenates were then centrifuged at 12000g for 20 min at 4°C and the supernatants were used for the NADP-ICDH assay, performed according to the method of Bernt and Bergmeyer (10). Protein content was determined according to the method of Lowry et al (11) using bovine serum albumin as a standard.

Mitoxantrone was administrated by intraperitoneal injection using a dose of 35 mg/m², 2.5h before the x-irradiation. Data were analyzed by using Student's 't' test.

Results

The levels of tumour ATP at different times after x-irradiation (10 Gy) show an augmentation compared with controls (3.42±0.47 nmol ATP/mg tissue, n = 36) which was first noted 45 min after irradiation (Fig 1). The maximum increases was observed 2.5h after receiving the x-rays (3.8 times that of the controls). Then 13h after irradiation, the ATP levels had returned almost to the control values.

After treatment with mitoxantrone in unirradiated tumours, the concentration of ATP was reduced by 0.76 times compared to controls ($P < 0.02$) and the levels of ATP following treatment with mitoxantrone and x-irradiation remained unchanged to within the experimental uncertainty ($P > 0.8$), compared to the values of tumour treatment with mitoxantrone only. The mean activity of NADP-ICDH was 0.46±0.07 µmoles isocitrate converted/min/mg protein as determined in 31 tumours. The activity of the enzyme increased following x-irradiation (10 Gy) to a maximum about 3h after irradiation (Fig 2).

Discussion

Adenosine-5'-trisphosphate has a major role in repair to intracellular damage after ionizing radiation and increased levels are most likely related to homeostatic regulation able to cope with physiological demands which follow radiation injury (12).

Sijens et al (3) have reported the response of a murine mammary carcinoma NU-82 in DBA-2 mice, to 10 Gy of gamma radiation. During the first 8h after this radiation the ratio ATP/inorganic phosphate increased. They hypothesise this effect is related to repair of radiation damage in particular to restoration of membranes. Benova et al (13, 14) reported that the protection of C57BL male mice from genetic radiation damage in germ-cell structures was achieved by using a combination of ATP, aminoethylisothiurorium Br-HBr and

serotonin. This reduced by half, the number of metaphases with translocations observed after 3 and 4 Gy x-rays to mouse spermatogonia, compared with irradiated mice not receiving the compounds. Removal of exogenous ATP from this combination led to a significant reduction (59%) in protective effect. NADP-ICDH mediated reactions are sources of reducing equivalents. The augmentation in the specific activity of this enzyme and its associated reducing equivalents may give protection against radiation damage. The enhanced activity of NADP linked dehydrogenase results from increased anabolic activity following radiation. Modification of the cellular energy supply by using cytotoxic compounds could therefore be an effective way to modulate cellular repair processes (15, 16). Mitoxantrone has been shown to decrease ATP production in rat heart slices in vitro (5) and here it is shown for the first time to decrease ATP production in tumours. Irrespective of the mechanism involved in the phenomena of repair, the present data show that if ATP levels fall as a result of interference from mitoxantrone, repair processes may be markedly inhibited as a result of reduced energy production. If the repair processes are slowed, a greater number of repairable lesions may be converted into lethal and non-repairable forms. Hence, for there to be effective repair processes, a higher ATP production must be superimposed upon the normal metabolic requirement for tumour cells.

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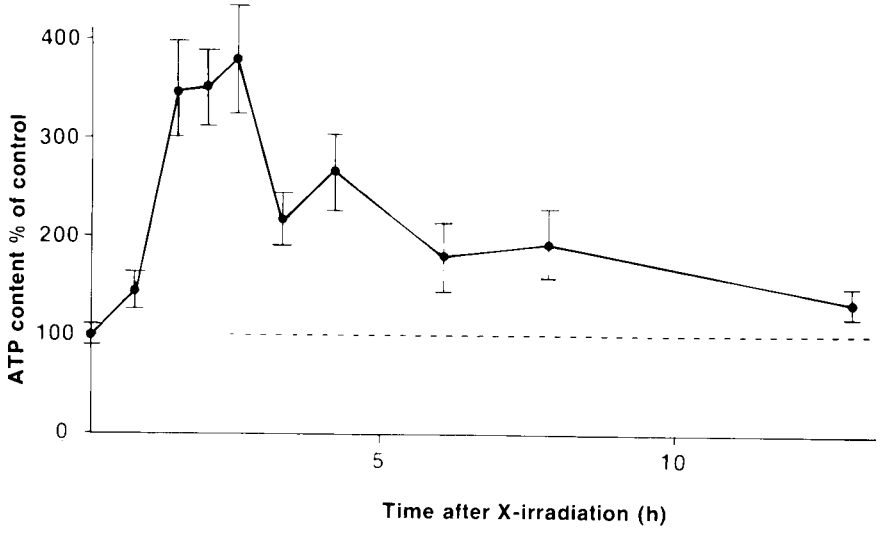

Fig 1 Radiation-related changes in ATP determinations from CaNT tumours in CBA mice at various times after irradiation. Data points represent the mean of at least three determinations ± SEM as indicated by the vertical bars. ATP levels after irradiation were significantly enhanced above levels in control tumours ($P < 0.05$).

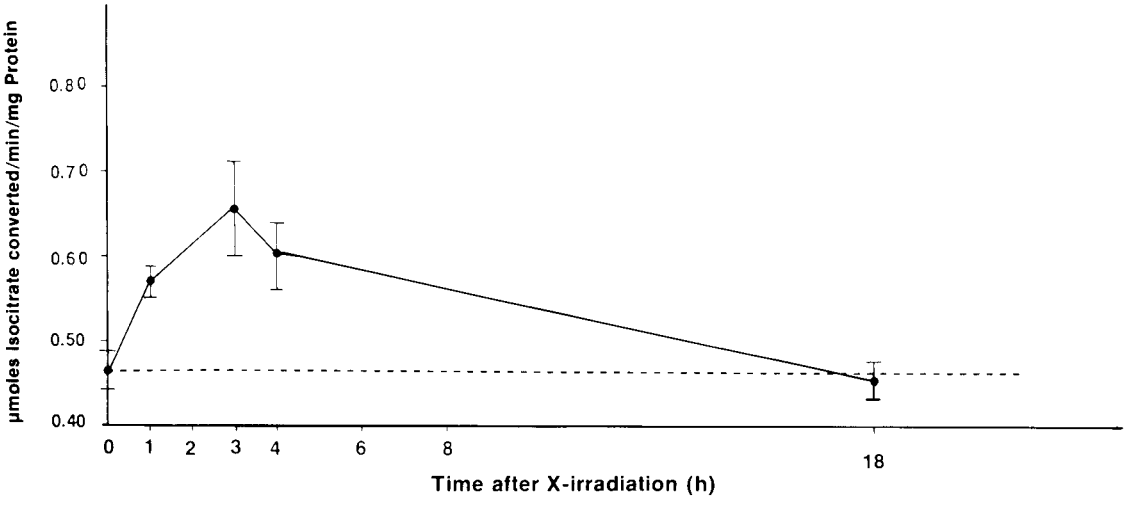

Fig 2 Radiation-related changes in NADP-Isocitrate dehydrogenase activity versus time in CaNT tumours in CBA mice. Each point represents the mean of not less than 5 values. SEM is indicated by vertical bars. This time variation was significantly different from the value obtained from unirradiated control tumours ($P < 0.05$).

PROPOSED TECHNICAL CRITERIA FOR STABILIZATION OF WASTES AND DECOMMISSIONING AND DECONTAMINATION AT FACILITIES USED TO RECOVER HAFNIUM, ZIRCONIUM, AND RARE EARTHS FROM SOURCE MATERIAL

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INTRODUCTION

On January 7, 1983, the United States Congress enacted Public Law 95-425, the Nuclear Waste Policy Act of 1982, to protect public health and the environment from the effects of radioactive waste. In Section 151(c), Subtitle D, of the Act, Congress stated its finding concerning the disposition of the special sites and wastes associated with licensed activities to recover zirconium, hafnium, and rare earths: "... the Secretary (of Energy), upon request of the owner of the site involved, shall assume title and custody of such waste and the land on which it is disposed when such site has been decontaminated and stabilized in accordance with the requirements established by the Nuclear Regulatory Commission (NRC) ...."

The Nuclear Regulatory Commission (NRC) has developed technical criteria for stabilization of wastes and decontamination and decommissioning of these facilities. The radiological characteristics of the typical wastes and the technical criteria are discussed in the following sections.

RADIOLOGICAL CHARACTERISTICS OF WASTES AND POTENTIAL HEALTH HAZARDS

The ores of zirconium, hafnium and rare earths generally also contain source materials such as thorium and uranium. Because of its source material contents, such operations are licensed by the Nuclear Regulatory Commission (NRC) or its Agreement States. Wastes materials generated during the processing of such ores can therefore contain natural uranium and thorium and their respective daughter products. These are the same naturally-occurring radionuclides contained in uranium and thorium mill tailings, although the relative concentrations of certain nuclides in special site wastes may differ substantially from those found in the mill tailings. In general, the thorium plus uranium concentration in the ores range from 0.05 percent to 10 percent depending on the origin of the ores.

At some of these sites, the ores and processing wastes, including most or all of the radionuclides present in the ore, have contaminated the soil and facilities; and small quantities may have been discharged directly onsite to land. The radiation produced by the radioactive component of the contamination is expected to be the primary source of potential hazard associated with these sites.

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Radionuclides in the uranium and thorium decay chains emit alpha, beta, and gamma radiation. The general pathways for radiation dose to person are from direct radiation, inhalation, and ingestion. The most significant pathways are from direct gamma radiation and from the inhalation of radon and its daughter products. The following technical criteria for decontamination and decommissioning (D&D) of sites and stabilization of source materials onsite are mostly emphasizing the reduction of radon emission and direct gamma radiation to acceptable levels. It is also assumed that the daughter products of the uranium and thorium series are generally in secular equilibrium. Any deviation from this assumption will be treated on a case-by-case basis.

PROPOSED CRITERIA FOR DECONTAMINATION AND DECOMMISSIONING (D&D)

Some of the land, structures, and equipment at these sites can be released for unrestricted use after clean-up by the licensee to the following criteria.

1. Structures, equipment to be released for unrestricted use shall be decommissioned and decontaminated so that:

   a. Direct gamma radiation exposure rates shall not exceed the background level by more than 20 μR per hour in any occupied or habitable building. Radon emanation to the interior of buildings shall be controlled in a manner that will ensure the annual average radon and its daughter product concentration within the structure is less than 0.03 working level (WL) and where reasonably achievable, below 0.02 WL. These criteria are taken from the U. S. Environmental Protection Agency's (EPA) standards established in 40 CFR 192 for protection against uranium and thorium mill tailings.[1]

   b. Surface residual radioactive contamination levels are as low as reasonably achievable (ALARA). Facilities and equipment below the levels established in NRC's "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Materials"[2] are considered in compliance with this criterion.

2. The concentration of radium-226 and radium-228 in land averaged over areas of 100 square meters, shall not exceed natural background levels by more than:

   a. 5 pCi/g, average over the first 15-cm surface thickness, or;

   b. 15 pCi/g, average over any 15-cm thickness that is more than 15 cm below the surface.

These limits are from the EPA's clean-up standards specified in 40 CFR 192. These limits are applicable only if the daughter radionuclides in the soil are in secular equilibrium with the parent nuclides.

PROPOSED CRITERIA FOR ONSITE STABILIZATION OF WASTES

Wastes generated at these sites are allowed to be disposed of onsite by stabilization meeting the following criteria:

1. Controls used to meet the stabilization criteria will be designed to be effective for up to one thousand years, to the extent reasonably achievable and in any case, for at least two hundred years.

2. Controls shall be designed to provide reasonable assurance that releases of radon-222 and radon-220 to the atmosphere from residual radioactive material will not:

   a. Exceed an average release rate of 20 pCi/m^2-sec or;

   b. Increase the annual average concentration of radon-222 and radon-220 in air at or above any location outside the disposal site by more than 0.5 pCi/liter.

   The above stabilization criteria are from the EPA's standards as specified in 40 CFR 192.

3. The site must be graded and drained to divert all surface water away from the waste area and located so that the integrity of the waste confinement is not jeopardized by the 100-year flood.

4. Waste shall be stabilized in a manner that provides maximum assurance that slumping or differential subsidence does not jeopardize containment integrity.

5. A multiple-layer cover shall be constructed over all disposed waste.

6. A soil stabilization program shall be implemented to minimize erosion of the cover or other engineered barriers formed of earthen materials and constructed on the surface.

7. An integrated drainage system shall be installed to remove surface runoff and infiltrated drainage water from the waste cover.

8. The cover shall be designed and installed to retard water from percolating into the waste.

9. The waste cover shall include a barrier that is designed to deter living organisms from breaching its integrity.

10. The disposal site should be located so that wastes are not subject to intrusion by ground water.

11. The disposal site should be located so that the integrity of the waste confinement is not jeopardized by geologic hazards.

CONCLUSION

The NRC staff believes that if these special sites are cleaned up and stabilized to the above technical criteria, they can be transferred to and under the custody of the Department of Energy for long-term control and maintenance and should not create an adverse impact to the environment.

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RADIOLOGICAL PROTECTION CRITERIA FOR THE RECYCLING OF MATERIALS FROM THE DISMANTLING OF NUCLEAR INSTALLATIONS

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A considerable fraction of the materials used in the construction of nuclear installations will, on decommissioning and dismantling, be only lightly active or contaminated. These materials could have high economic value and this provides an incentive for recycling or reuse.

The recycling of materials from the dismantling of nuclear installations, either as scrap or as discrete items, would require them to follow conventional processing and marketing procedures which do not provide for further radiological control. Since numerous installations are reaching the end of their working life, and since large quantities of such scrap and items circulate between the various countries the radiological problems of recycling would have to be resolved on an international scale.

On the initiative of the Commission of the European Communities, the Group of Experts set up under the terms of Article 31 of the EURATOM Treaty to advice the Commission on radiological protection standards, set up a Working Party to look into these problems and propose appropriate criteria. The Working Party prepared a report in which the impact of recycling was assessed, criteria directly applicable to steel scrap and equipment from nuclear power plants were proposed and methods described which can be applied for the development of criteria for other valuable metals, such as copper and aluminium, and other nuclear installations. On the basis of this report, the Group of Experts issued a recommendation to national authorities which includes the criteria set out below.

There have been extensive studies of power reactor decommissioning in the US (1,2,3) and Europe (4), but no practical experience has been gained yet. These studies show that the total quantity of steel in the active areas of a large (1000 MWe) pressurized water reactor or boiling water reactor is approximately 10,000 tonnes, of which about half has a potential for recycling with currently available techniques. An advanced gas cooled reactor of the 600 MW(e) class would also contain in its active areas about 10,000 tonnes of steel with a similar potential for recycling (5), whereas the quantity of steel from active areas resulting from the decommissioning of a typical steel pressure vessel Magnox reactor would be about 13,000 tonnes, of which about a third may be suitable for recycling (6 7).

In the study attention was paid to existing examption criteria and limits (8,9); it was concluded that a suitable basis for establishing criteria for recycling was not available. Two

approaches to the problem were explored, one based on defining acceptable individual and collective dose levels, the other based on setting "clearance" levels for the activity concentration of the materials concerned, such that the potential individual and collective doses resulting from recycling would be insignificant. An analysis of current regulatory practice showed that the stipulation of dose limits for the practice of recycling poses practical, and sometimes intractable, problems for both regulators and operators. The requisite criteria were consequently formulated based on activity concentration levels. These are:

- For β/γ radiation 1 $Bq.g^{-1}$ averaged over a maximum mass of 1000 kg, to avoid the inclusion of highly active items within the average mass there is an additional requirement that no single item may exceed 10 $Bq.g^{-1}$.

- To comply with IAEA regulations (10) for the safe transport of radioactive materials the surface activity for β/γ radiation would have to be limited to 0.4 $Bq.cm^{-2}$ for non-fixed contamination on accessible surfaces, averaged over 300 cm^2 or over the surface area if less than 300 cm^2; for fixed contamination the mass activity concentration clearance level is assumed to apply. The surface activity for α radiation is to be limited to 0.04 $Bq.cm^{-2}$ measured over any area of 300 cm^2 of any part of the surface. For both β/γ and α concentrations, if doubt exists for non-accessible surfaces, the activity must be assumed to be higher than the respective clearance level.

The recommended clearance levels are intended to apply to the total activity concentration of all nuclides, i.e. not only those due to residence in a nuclear installation, but also those present in the raw materials that were used in the manufacture of the steels and equipment involved, and those added during the manufacturing process (such as Co-60 used to indicate the wear of furnace liners).

Examination of the question of the boundary at which the criteria would be applied led to the conclusion that it is not necessarily, in every case, either the boundary of the licensed site or processing plant or of the state concerned. The relevant boundary is that at which all control is in fact relinquished, and as such may vary from case to case.

A generic assessment of radiation exposures resulting from the recycling of materials was performed, in which individual and collective doses to workers and the public were evaluated corresponding to various activity levels, in order to appreciate the relative impact of each radionuclide and to identify the most limiting cases. At the clearance levels recommended, the maximal individual dose to workers or members of the public would be less than 10 $\mu Sv.a^{-1}$. The collective dose from the recycling of 10000 tonnes of steel in a year, arising for example from the dismantling of two large PWRs, would be about 1 man Sv. In exceptional circumstances, associated with very low probabilities of occurrence, the dose to the most exposed individual from the direct reuse of scrap or equipment having an activity concentration equal to the level recommended might be greater than 10 $\mu Sv.a^{-1}$, but the individual risk is not expected to increase (11). Similarly, the

recycling of amounts of materials greater than 10000 tonnes per year will result in a linear increase in the collective dose but will have no effect on the magnitude of individual doses.

For materials with levels of activity concentration above the proposed clearance levels, a case-by-case assessment, of the kind currently performed by the competent authorities in accordance with national regulations, may be required for a particular recycling application.

It will be the responsibility of the national authorities concerned to ensure compliance with the recommended clearance levels. It should be noted, however, that appropriate techniques to ascertain such compliance already exist. To this end, a monitoring programme should be designed and applied, once all significant parameters relevant to the components or metals being released have been identified. This programme should include survey plans for surface contamination measurements, sampling techniques and analyses for activation products, selection of appropriate portable field instruments and fixed and mobile laboratory equipment, and quality assurance methods for the documentation and control of measuring and sampling procedures, instructions, processes and equipment.

It should be noted, however, that practical problems of measurement may well arise in the case of the very low mass activity concentration for alpha emitters derived from the above clearance level for surface contamination. In addition, the experience gained to date with materials having a potential for recycling shows that the alpha activity, if present, is in the form of surface contamination. For these reasons a clearance level for the mass activity concentration of alpha emitters, has not been recommended.

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POLICY AND CRITERIA FOR THE RECYCLE AND REUSE OF VERY LOW LEVEL CONTAMINATED MATERIALS FROM MAINTENANCE REFURBISHMENT AND DECOMMISSIONING OF NUCLEAR FACILITIES IN THE FEDERAL REPUBLIC OF GERMANY

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1. INTRODUCTION

In the course of maintenance, refurbishment and decommissioning of nuclear facilities large amounts of materials with very low level specific or surface activities arise, which may be handled either as conventional waste, reused and recycled conventionally or treated as radioactive waste. In any case it is necessary to define criteria which allow to discriminate between the two main options. When establishing such criteria one should start at the basic radiation protection criteria given by ICRP 26.

2. JUSTIFICATION AND APPLICATION OF THE ALARA-PRINCIPLE

It has been estimated that in the OECD-countries the equivalent of two hundred 1000 MW(e) nuclear power plants will be definitely shut down in the period from 1996 to 2010. Studies have shown, that e.g. the total quantity of steel in a large PWR is approximately 10,000 tonns, of which half has a potential for recycling with currently available techniques. (1) That means that recycle and reuse of very low level contaminated or activated steel scrap from decommissioned NPP's offer the potential for extending the life time of valuable natural reserves and is of particular interest for materials which are in short supply in the world and for materials whose extraction from the earth crust is difficult and expensive. In addition an economic benefit from recycling might arise from the recovered value of the recycled material and the savings to be made in costs of conditioning, packaging, storage, transport and especially the disposal of large volumes of very low level radioactive material as radioactive waste.

On the other hand however it must be seen that the release of large amounts of very low level radioactive materials for unrestricted recycle or reuse will cause the exposure of many people to ionising radiation even if only with very low doses. In order to minimize this exposure in the Federal Republic of Germany the first option for recycle and reuse of very low level active materials, especially steel scrap is to restrict the recycle and reuse to the nuclear field as far as possible. How this is already practiced will be described in chapter 4.

3. ESTABLISHMENT OF ACTIVITY LIMITS FOR UNRESTRICTED RELEASE

There seems to be already international agreement that the annual effective dose equivalent to the most exposed individuals

caused by the unrestricted release of materials from decommissioned nuclear facilities should not exceed the order of magnitude of 10 µSv.The IAEA e.g. requested that a source or practice can be exempted from notification, registration and licensing provided that:

(1) The annual effective dose equivalent to individuals of the critical group at no time exceeds 10 µSv, and the annual dose equivalent to skin does not exceed 500 µSv; and
(2) The collective effective dose equivalent commitment from the exempted source or practice is of the order of 1 manSv or less. (2)

There arises the question whether the requirement "at no time exceeds 10 µSv" is not to strict. In order to demonstrate that release limits given as specific activity and surface contamination are in compliance with the dose limits, it is necessary to use exposure scenarios by which the exposure of individuals can be derived from the release of material of given mass and activity. Numerous studies exist in this field. (3, 4, 5, 6, 7) One of the common results is that in most cases the external irradiation by gamma-emitting radionuclides dominates the doses to people exposed to these materials.

But all scenarios which might be established can never be conservative enough that it could not be possible to describe one which is still more conservative.Therefore it seemed desirable however, to have a method available to assess the actual doses to individuals of the general public. Based on this information, one could derive release criteria.The release of radioactive metals from the controlled area for unrestricted use and the processes of recycling have basically a stochastic character. The pathways of these materials cannot be predicted deterministically and the quantities which are relevant for the radiological assessment are fluctuating. Consequently a stochastic simulation of the complete process of scrap release, scrap processing, steelmaking, product manufacture and the use of products including exposure scenario on each of these stages presents an adequate approach. Such a probabilistic model has been developed (8),some of the important input parameters of which are listed in Tab. I.

Tab. I: Basic parameters for the stochastic simulation of scenarios

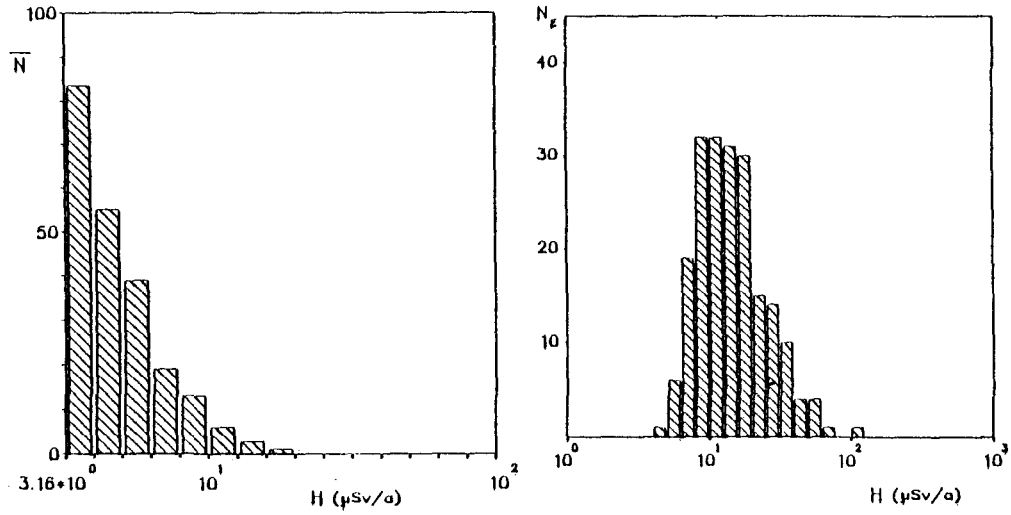
Amount of scrap released	1000 t
Contamination release level	0.37 Bq/cm^2
Release level for "specific total activity" ((surface + bulk activity)/mass))	1 Bq/g
Average fraction of activated scrap	0.1

Note: Only external gamma-exposure taken into account.

The result of each simulation is a distribution of individual doses. Fig. 1 shows the average dose distribution received by multiple repetition of the simulation. The distribution of the maximum doses received after 200 fold repetition of the simulation process is given in Fig. 2. One sees that the requirement of the IAEA that 10 µSv/a is never exceeded is not fullfilled but that the number of cases were this figure is exceeded is rather limited and the maximum doses remain within one magnitude of order above 10 µSv/a.

Fig. 1 Distribution of individual doses. $\overline{N}$: Average number of persons who received doses in a certain range.

Fig. 2 Distribution of maximum individual doses. N_E: Number of events with the maximum dose in a certain range.



$\overline{N}_0$ = 210 (3,16 µSv/a ≤ H ≤ 10 µSv/a)

$\overline{N}_1$ = 10 (H ≥ 10 µSv/a)

These results were used in order to support the establishment of the following release criteria in the Federal Republic of Germany. They are to be applied for the time beeing for steel scrap coming from NPP only:

a) Unrestricted release: Completely unrestricted release is possible if the specific overall activity is not higher than 0.1 Bq/g and the surface contamination does not exceed 0.37 Bq/cm² for beta-gamma-emitters and 0.037 Bq/cm² for alpha-emitters. All single items have to comply with these limits. The surface contamination may be averaged over 100 cm².

b) Release for general melting: The release of scrap material for general melting in a normal steel furnace together with other inactive scrap is possible if the specific

overall activity of each single item is not higher than 1 Bq/g and at the same time the surface activity conditions as for unrestricted release are complied with. The producer of the scrap just has to demonstrate that the scrap material he is going to release is really going into a furnace. The owner of the furnace does not need any licence for handling this material.

c) Controlled recycling: If the specific overall activity of 1 Bq/g is exceeded or it is not possible to measure this because the scrap items are of to complicated geometrical shape or are to small a controlled melting is possible under a special licence according to the German Radiation Protection Ordinance. The only condition is that if the product material is going to be released unrestrictedly the specific activity must not exceed 0.1 Bq/g. In any case the resulting specific activity of the material must not exceed 1 Bq/g. The competent authority can allow that this material is used outside controlled areas if it can be foreseen that nowbody receives an enhanced exposure by this.

In all three kinds of release for any single case a licence by the competent authority is necessary.

4. RECYCLING WITHIN THE NUCLEAR FIELD

As mentioned above, the basic requirement for recycling of scrap is to do this within the nuclear field. This is practiced in Germany by several producers and the Siempelkamp Giesserei GmbH and Co. (9) They started with this idea already in 1981 and up to now about 1000 t of metallic waste were melted and recycled. The way to melt is in short:

- cutting the material and filling into 200 l-drums;
- charging the material into the furnace by using a special developed charging device;
- melting in an induction furnace under low pressure by tight enclosure of the melting bath;
- keeping low pressure by a filter system, which filters all evading radioactive aerosoles coming from the melt.

To be able to melt in this special manner several licenses had to be achieved by the German authorities, who restricted the activity of the contaminated scrap to be melted to not more than 74 Bq/g. All the material being melted is used to produce components for nuclear facilities like waste and transport containers, shielding plates, crane testweights etc.

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APPLICATION OF EXEMPTION PRINCIPLES TO LOW-LEVEL WASTE DISPOSAL AND RECYCLE OF WASTES FROM NUCLEAR FACILITIES[a]

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INTRODUCTION

The International Atomic Energy Agency (IAEA) and other international groups for the past several years have been investigating the possibility of exempting from regulatory control certain radiation sources and practices, initially under the general heading of de minimis. Much of the interest in this topic arises from the recognition that a significant fraction of the wastes from industry, research, medicine, and the nuclear fuel cycle are contaminated to such low levels that the associated risks to health are trivial. Therefore, the application of regulatory processes seems to be unwarranted. Recently, the IAEA work has been conducted by Advisory Groups on two interrelated levels: to establish principles for exemption, and to apply the principles to various areas of waste management. In the second area, the main objectives have been: 1) to illustrate a methodology for developing practical radiological criteria through the application of the IAEA preliminary exemption principles, 2) to establish generic criteria, and 3) to determine the practicability of the preliminary exemption principles. The method used by the IAEA Advisory Groups to develop the criteria relies on a modeling assessment of the potential radiation exposure pathways and scenarios for individuals and population groups following the unrestricted release of materials. The scenarios and models used are necessarily generic; however, an attempt was made by the Advisory Groups to identify the most important pathways based on available literature. The generic scenarios and methods are intended to provide the basic framework for the numerical derivation of generic exempt quantities that would be adequately conservative in most situations. This paper describes the IAEA's assessment methodology and presents the generic results expressed in terms of the limiting activity concentrations in municipal waste and in low-activity materials for recycle and reuse.

PRELIMINARY EXEMPTION PRINCIPLES

As the result of recent IAEA Advisory Group efforts on the subject of exemption, the IAEA published interim general principles for exemption of sources and practices that result in both individual and collective doses of very low significance (IAEA 1987a). The principles are intended to be quite general and applicable to any type of manmade radiation source that gives rise to trivial risks; they do not, however, apply to natural sources of radiation. These interim exemption principles provide a safety margin to account for selected individuals who may be exposed to radiation from several exempted sources and

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(a) This work was supported in part by the International Atomic Energy Agency and the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

to account for the uncertainty of future human activities. With this safety margin in mind, an Advisory Group to the IAEA recommended that the individual doses from a single exempted source or practice should not exceed 1% of the existing individual dose limit for members of the public, or 10 μSv. This dose equivalent is less than 0.5% of the annual effective dose equivalent from natural background radiation and is small compared with the natural variation in background radiation. For skin doses, the IAEA Advisory Group also recommended a dose limit of 1% of the existing limit, or 500 μSv (IAEA 1987a).

The IAEA Advisory Group also recommended that consideration be given to controlling the collective dose to provide additional assurance that many small doses will not add to a significant total and to guard against the possibility that this could occur without the knowledge of controlling authorities for sources which are exempt and, therefore, not subject to notification and registration. The Advisory Group recommended that as part of the "basic case" for exemption, the collective effective dose equivalent commitment from the exempted source or practice should be about 1 person-Sv or less (IAEA 1987a). This does not preclude national authorities from exempting sources that give rise to larger collective doses, but merely establishes a condition below which no further consideration needs to be given on the radiological basis for exempting a source. The Advisory Group considered that sources and practices which comply with the conditions relating to individual and collective dose may be exempted from the normal regulatory requirements of registration and notification, and treated just as if no radiation exposures were involved.

DESCRIPTION OF GENERIC METHODS

Exempt quantities, expressed in units that relate to radiation-detecting instruments, are a more practical expression of the general exemption principles. The steps used by the IAEA Advisory Groups in deriving exempt quantities for a defined source or practice involve: 1) establishing a series of radiation exposure scenarios that account for various exposure pathways and conditions, 2) estimating the resulting radiation doses to individuals and population groups for these scenarios, 3) determining the limiting (highest dose) scenario for each radionuclide, and 4) determining the concentration of individual radionuclides that would result in the exemption criteria (dose limits). In assessing the radiation doses, the IAEA Advisory Group advised that care must be taken in the selection of parameters, assumptions, and data (IAEA 1987b). For their assessments, the IAEA Advisory Group judged scenarios on the likelihood of their occurrence leading to human exposure and the likely magnitude of those exposures. In addition, the potential exposure of a critical population group was necessary.

In their evaluations to date, the IAEA Advisory Groups have considered: 1) disposal of exempt wastes in a sanitary landfill (IAEA 1987b); 2) disposal of exempt wastes by incineration (IAEA 1987b); 3) recycle of contaminated steel, aluminium, or concrete;(a) and 4) reuse of concrete buildings, tools, or equipment.(a) The radiation exposure pathways included external exposure to penetrating radiation, inhalation of airborne material, and ingestion of contaminated food crops or removable surface contamination (through secondary transfer from hands to the mouth). A variety of representative radionuclides were considered to fully explore the radionuclide-dependence of the resulting exemption limits. These radionuclides were chosen to represent alpha emitters (^{239}Pu and ^{241}Am),

(a) As described in a draft working document on "The Application of Exemption Principles to Wastes from Decommissioning and Recycle of Materials from Nuclear Facilities."

high-energy photon emitters (^{60}Co), low-energy photon emitters (^{55}Fe), and pure beta emitters (^{90}Sr and ^{99}Tc).

The potential radiation exposures resulting from different scenarios that may be envisaged have a probability of occurrence that may range from zero to one. Thus, judgment is necessary when including scenarios for the derivation of exempt quantities. As a result, most attention was paid to those scenarios where individuals could have direct contact with the radioactive materials. These scenarios included workers at landfills, incinerators, smelters or recycle centers, and consumers who may use materials made from recycled materials or who reuse released buildings, tools, or equipment. Additional scenarios, such as use of ground water near a landfill or release of volatilized material through the stack at a smelter, were also included to provide an estimate of the likely collective dose.

RESULTS AND DISCUSSION

Example results for the groupings of reference radionuclides and the various types of exemption considered by the IAEA Advisory Group are summarized in Table 1. The results are presented in terms of reasonably expected ranges, based on the various radionuclides in each group, and the expected variation among exposure scenarios. Control of the future fate of exempted materials through unrestricted release is lost; thus, material exempted for recycle or reuse could be disposed of in a landfill, or material exempted to a landfill could be recycled or reused. Because of the lack of future control, the proposal has been made that a single exempt quantity should be established which would cover all alternative future conditions, without placing specific limitations for landfill disposal, incineration, recycle, or reuse. This appears to be possible because of a close grouping of the results shown in Table 1 across most radionuclide and unrestricted-release categories.

TABLE 1. Example Exempt Quantities for Various Exemption Categories

Exemption Category	Alpha Emitters	High-Energy Photon Emitters	Low-Energy Photon Emitters	Pure Beta Emitters
Sanitary Landfill (Bq g^{-1})	1 - 10	0.1 - 10	(a)	300 - 600
Incineration (Bq g^{-1})	0.1 - 100	0.5 - 10	(a)	10^2 - 10^3
Recycle(b) (Bq g^{-1})	1 - 10	1 - 10	10^4 - 10^5	40 - 300
Building Reuse (Bq cm^{-2})	1 - 5	0.004 - 1	10 - 100	60 - 500
Reuse of Tools and Equipment (Bq cm^{-2})	10 - 100	10 - 100	60 - 500	10^2 - 10^3

(a) No radionuclides were considered for the scenarios shown.
(b) For recycle of steel, aluminium, or concrete rubble.

Three potential groupings of released material are: 1) mass concentrations (in units of Bq g^{-1}), 2) surface contamination in buildings (in units of Bq cm^{-2}), and 3) surface contamination on reused tools and equipment (in units of Bq cm^{-2}). Further, it appears that a set of radionuclide groupings could be made by combining the high-energy photon emitters and alpha emitters into a single grouping across all categories. The resulting overall exemption limits are summarized in Table 2.

TABLE 2. Preliminary Exemption Limits for all Release Categories

Limit Category	High-Energy Photon and Alpha Emitters	Low-Energy Photon and Pure Beta Emitters
Mass Concentration (Bq g^{-1})	1 - 10	10^2 - 10^3
Building Surfaces (Bq cm^{-2})	0.1 - 1.0	10^2 - 10^3
Reuse of Tools and Equipment (Bq cm^{-2})	10^1 - 10^2	10^2 - 10^3

Again, ranges of values are shown to denote the potential variations of radionuclides and exposure conditions. For mixtures, the sum of the fractions rule could be applied. The net result for mixtures is that the limit is controlled by the most restrictive radionuclides present. It may be noted that for all scenarios considered, the individual dose criterion is limiting in relation to the exempt mass concentration and surface contamination limits; the collective dose criterion is the determining factor, however, in assessing the total quantity of material that may be buried, incinerated, or recycled. The IAEA work will continue, because further studies are needed to determine if additional practical considerations (including costs and detectability) will change the results or the basic conclusions. At the same time, the preliminary exemption principles will be reviewed again, taking due account of the experience gained through these applications.

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PRE-DECOMMISSIONING ENVIRONMENTAL INVESTIGATIONS AT THE GARIGLIANO NUCLEAR POWER PLANT

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INTRODUCTION

Garigliano power plant is a 160 MWe Boiling Water Reactor with dual cycle. It was operated from 1964 to 1978 generating 12.5 10^9 KWh with a capacity factor of 68%. In August 1978 it was shut-down due to damage to one of the two secondary steam generators. In March 1982 the Italian Electricity Generating Board (ENEL) declared it definitively out-of-service. In December 1982 ENEL approved an "Action Plan" to put the Garigliano plant in safe storage conditions (Stage 1) for at least 30 years. Specific studies (1-3) were started as well to select the best decommissioning alternatives. A feasibility study on dismantling and on the decontamination of all the building components was carried out to start up the decommissioning procedure of the turbine building. The aim of present investigations is to evaluate the radiological status of the environment before any decommissioning programs. Since most decommissioning projects can yield a large volume of low levels of radioactive liquid of effluents, a previsional aquatic model (4) was developed to simulate the radionuclide transport and distribution in the aquatic ecosystem.

RADIOECOLOGICAL INVESTIGATIONS

Two field investigations were carried out in 1985 and 1986. However the present paper reports only the data of the campaign carried out before the Chernobyl nuclear accident.

RADIOACTIVITY DISTRIBUTION IN TERRESTRIAL ENVIRONMENT

The investigated area is located within 15 km from the Nuclear Plant. The following environmental matrices were considered: agricultural and undisturbed soils, well water, and foodstuffs (both vegetable and animal). Only ^{137}Cs was detected in the examined matrices, except for drinking water from wells (Fig. 1). No ^{60}Co was measured above the detection limit (foodstuffs, 0.02 $Bq/Kg_{w.w.}$ and soil, 0.35 $Bq/Kg_{w.d.}$) in the direct gamma ray spectrometry. The activity of ^{137}Cs, mostly due to deposition from nuclear tests in the atmosphere, and gross alpha and beta activity, as obtained from radiochemical separation, decrease according to the following series: soil > vegetables > eggs > milk > meat. By using the derived working limits (DWL_α) it is possible to assign a radiological meaning to the measured activities. An individual exposition of 5 10^{-2} mSv/y was considered and accordingly the following DWL_α were derived: for foodstuffs DWL_α ^{239}Pu = 11 $Bq/Kg_{w.w}$ and DWL_β ^{90}Sr = 55 $Bq/Kg_{w.w.}$; for soil DWL_α ^{239}Pu = 444 $Bq/Kg_{w.d.}$ and DWL_β ^{90}Sr = 185 $Bq/Kg_{w.d.}$.

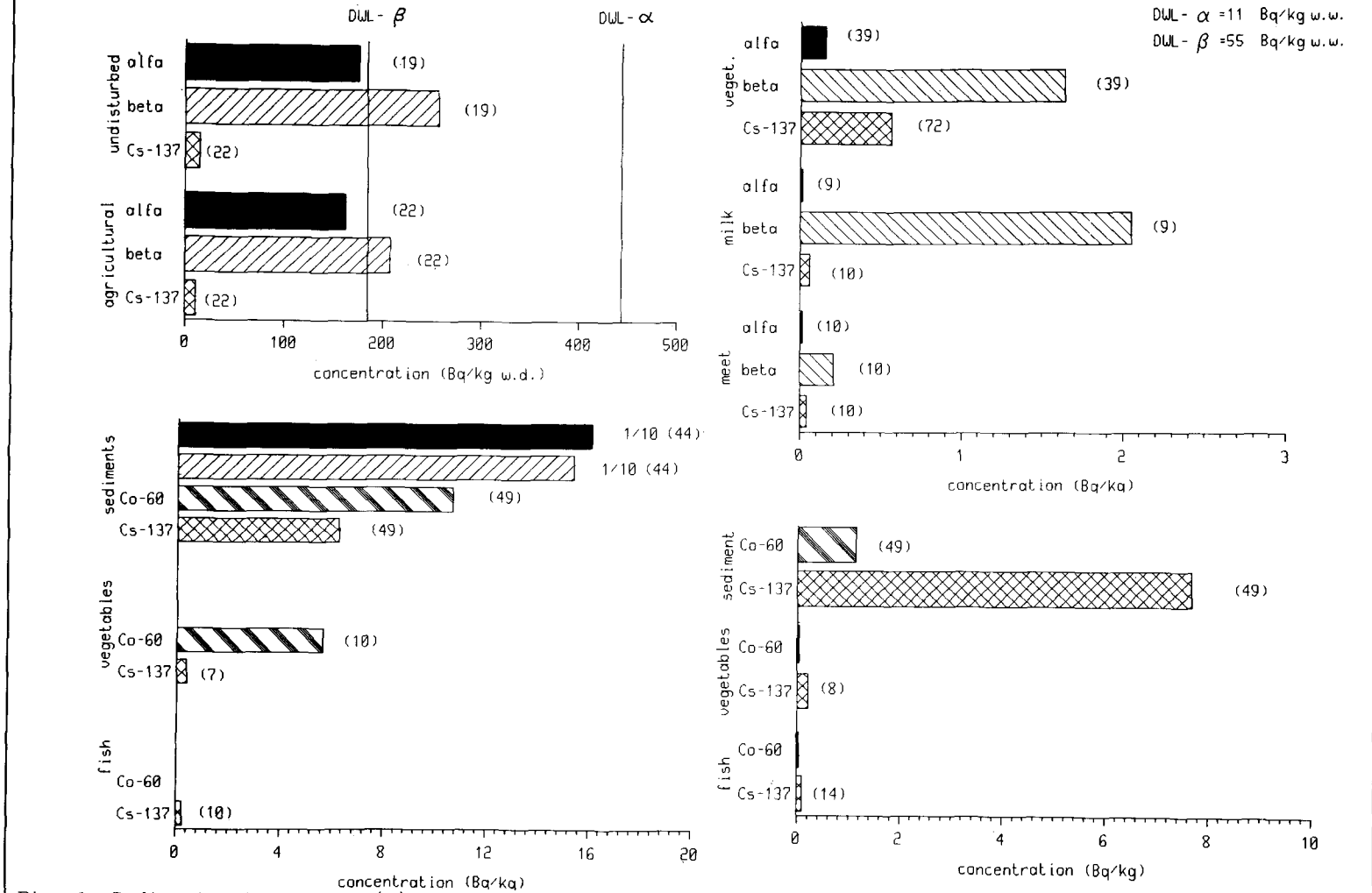

Fig. 1 Radioactivity in soil (A), terrestrial foodstuffs (B), sediment,aquatic vegetable and fish both for riverine (C) and marine ecosystems (D). () number of analyzed samples.

The measured gross alpha and beta activities, however, are 1 to 3 orders of magnitude lower than DWL_β for foodstuffs, whereas in soils the measured beta activity is similar to the DWL_β and the alpha activity is lower by a factor of two with respect to the DWL_α. In addition it is worthwhile to underline that the measured gross alpha and beta activities include a large contribution from natural radioactivity as the used radiochemical separation techniques (5) only partially discriminate between natural and artificial radioactivity.

RADIOACTIVITY DISTRIBUTION IN RIVER AND MARINE ENVIRONMENTS

A reach of the Garigliano river of 12 km from power plant to the mouth was considered, while the investigated marine environment is shown in Fig. 2. In both environments the following matrices were considered: water, suspended matter, sediment, vegetables and fish (edible portion). Artificial radioactivity, due to ^{60}Co and ^{137}Cs, was measured in the following decreasing order: sediment > vegetables > fish (only ^{137}Cs) > water (only ^{137}Cs). The ^{137}Cs activity in river and marine fish and vegetables is similar. Conversely ^{60}Co activity in vegetables is higher in river ecosystem (6 $Bq/kg_{w.w.}$) than in marine macrophytes (less than 0.09 Bq/Kg w.w.). A different distribution of ^{60}Co and ^{137}Cs is shown also in sediments; the $^{137}Cs/^{60}Co$ activity ratio is 0.59 in river environment and 6.7 in marine environment. It must be pointed out, however, that the ^{137}Cs levels are just of the same order of magnitude upstream and downstream the power plant and in the marine ecosystem. The spatial distribution of ^{137}Cs and ^{60}Co are shown in Fig. 2. ^{137}Cs activity appears to be evenly distributed, whereas it is possible to locate points of ac cumulation of ^{60}Co. A deeper reading at the above reported results indicates that: - ^{137}Cs activity is mostly due fallout depositions with respect to the one released from the plant, whereas the contrary happens for ^{60}Co activity; - the different spatial distribution of ^{137}Cs and ^{60}Co may be ascribed either to the different origin and to their different mobility in the environment; - it is necessary to describe, as accurately as possible, the complex transfer processes of radionuclides in the environment in order to identify the final fate of artificial radioactivity releases over large areas.

AQUATIC MODELLING

A mathematical model for Distribution of Radioactivity Material in Marine environment (DRAM) was developed to predict transport and distribution of most relevant fission and activation nuclides. DRAM is a box model which divides the coastal area into three-dimensional zones of uniform physical characteristics. Interzonal, external meteoric and evaporative water fluxes are associated to each zone. Fluxes between compartments (water, particulate, pore water and sediment) in the same zone are calculated assuming that: a) dissolved radionuclides are in equilibrium with suspended particulate radionuclide; b) suspended particulate settles out continuosly causing the bottom of each zone to

rise with constant velocity. The process is associated with movements of radionuclides from the water column to the sediment; c) radionuclides flux across the water column sediment interface, is modelled through a catch-all parameter: the "diffusional transfer rate", to be experimentally determined. The model has been applied to the investigated site. The released radionuclides, initially associated with the liquid effluents, are incorporated into the Garigliano river and transported to sea. In order to obtain the source term, the necessary input for the DRAM model, a model of radionuclide river transportation was developed. The model is monodimensional and takes care of both dissolved and suspended radionuclides and of the tide cycle.

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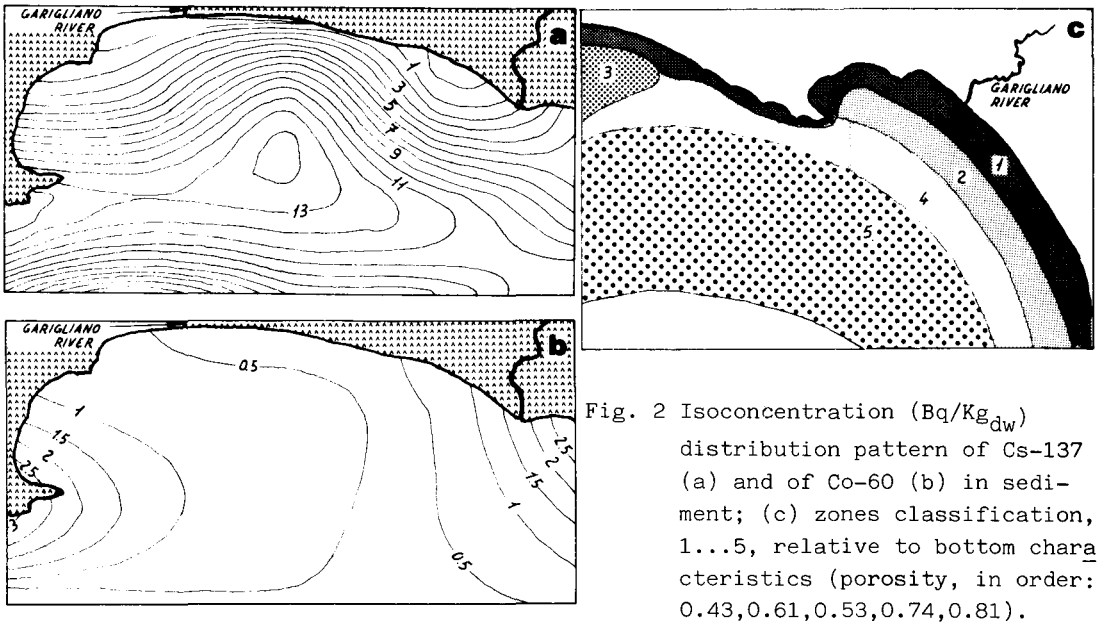

Fig. 2 Isoconcentration (Bq/Kg_{dw}) distribution pattern of Cs-137 (a) and of Co-60 (b) in sediment; (c) zones classification, 1...5, relative to bottom characteristics (porosity, in order: 0.43,0.61,0.53,0.74,0.81).

MANAGEMENT OF WASTE CONTAINING RADIOACTIVE MATERIALS AND CHEMICAL AGENTS

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INTRODUCTION

Human activities generate large quantities of unwanted materials and proper management of these materials is essential in maintaining a healthy and pleasant environment. In the United States, toxic waste is categorized in three groups. Materials associated with the nuclear fuel cycle are covered by the Atomic Energy Act (AEA), naturally occurring or accelerator-produced radioactive materials (NARM) are covered by various other laws, and chemically hazardous waste is covered by the Resource Conservation and Recovery Act (RCRA). For the sake of simplicity, henceforth, these are referred to as AEA waste, NARM waste, and RCRA waste, respectively.

Historically, the first group of waste to be regulated was AEA waste. This occurred informally during the 1950's in the facilities of the then Atomic Energy Commission, and a decade later in the commercial sector. In contrast to the AEA waste, the law mandating a control of RCRA waste was enacted in 1976 and, in a stricter form, in 1984. Currently, NARM waste is regulated only at the State level.

DEFINITIONS

The definition of AEA waste is relatively straightforward. Unwanted materials containing uranium or thorium (source materials), plutonium and certain other transuranic elements (special nuclear materials), and radioactive materials generated as a result of or incidental to the nuclear fission, are defined as AEA waste.[1] All other radioactive materials are covered under NARM. The AEA waste is further subdivided into low-level waste (classes A, B, and C), high-level waste (reprocessing waste), spent nuclear fuel and transuranic waste. Furthermore, U.S. law defines a class of waste referred to as above class C low-level waste or class D waste.[2]

In contrast to the definition of AEA and NARM wastes, the definition of RCRA waste is complex. There is an inherent problem in categorizing the universe of all chemical compounds into two categories, one being hazardous and the other being non-hazardous. Furthermore, the U.S. law defines solid waste as unwanted materials that are "solid, semisolid, liquid or gaseous."[3] The law also regulates man-made materials and has only recently recognized that natural materials can be also toxic, as demonstrated by the recent attempts to control radon in indoor air.

The initial attempts to define a generic toxicity test failed, and had to be replaced with several categories and a process referred to as listing. Currently, a

waste is considered "hazardous" if it fails tests dealing with corrosivity, reactivity, flammability, and the so-called EP-toxicity. This latter test consists of leaching a solid material under specified conditions. If the concentration of the leachate exceeds values identified in the regulations, the waste is considered RCRA waste. The overwhelming majority of wastes are, however, not covered by these tests. Therefore, the regulator has established several lists. If the analysis of the waste indicates the presence of a chemical on that list, the waste is considered RCRA waste.

TECHNICAL PROBLEMS IN MANAGING MIXED WASTE

Differences in the definition of waste would not constitute a major problem in managing a waste consisting of unwanted AEA and RCRA materials. However, there are several technical differences in the waste management approach between AEA waste and RCRA waste. Management of AEA waste is covered by two basic rules. The first rule contains the dose-equivalent limits for the workers and the general public, accepted universally, with the added limitation of an individual dose limit of 0.25 mSv to the population living around a low-level disposal site.[3] The second rule contains the universally accepted ALARA concept. Both of these are performance criteria and provide the designing engineers a great deal of flexibility.

In contrast to these, RCRA waste is basically a design-based system. The designing engineer is not told the required performance limits. Instead, he is given specific design criteria. For example, the law prescribes that a disposal site must have a double liner with leachate collection systems. This prescribed design leads inevitably to the so-called "bathtub effect." Rain water is collected in the liner, and must be removed and processed like effluents from a hazardous waste facility. The law makers have recognized that not all of the water can be removed and thus up to 30 cm of water is permitted to remain in the system.

There are many other areas of technical difficulties in reconciling a performance-based with a design-based system. For example, every RCRA waste package is visually inspected to assure the absence of free-standing water, an unreasonable practice for AEA waste because of the exposure to ionizing radiation. Finally, an AEA waste disposal site is designed for time frames exceeding a century while RCRA waste disposal sites could retain the waste for no more than a few decades.

INSTITUTIONAL PROBLEMS

The AEA waste is principally managed by the Federal government. The regulations governing the packaging, transportation, and disposal of the waste are promulgated by the Nuclear Regulatory Commission (NRC), based on standards set by the U.S. Environmental Protection Agency (EPA). The states can adopt these regulations and enforce them. These "agreement states" act on behalf of the Federal government and can develop additional regulations. For example, many states have expanded the rules of AEA to cover NARM. In general, the states must adopt the Federal regulations, and thus, there is a large degree of uniformity in the regulations dealing with AEA waste. However, the states may develop their own disposal site requirements.

In contrast to the AEA waste, the laws governing the RCRA waste empowers and encourages the states to develop more restrictive requirements than the Federal rules in every aspect of waste management. Accordingly, a mixed waste generator who meets the requirements of one state is not necessarily assured that he is meeting the requirements of the other 49 states.

DE MINIMIS LEVEL OF WASTE

Until recently, the public was led to believe that a toxic material is toxic at any level and a waste that was once declared toxic was toxic forever. Both AEA and RCRA wastes had a "cradle-to-grave" approach by requiring that records had to be kept from the time a toxic material arrived at a place until it left. Not surprisingly, the volume of the waste increased rapidly, with ever fewer and fewer sites to dispose of them. For the AEA waste, the NRC has had an exemption process whereby a material could be removed from the coverage of AEA if the delivered dose equivalent was negligible. The informally adopted dose equivalent was 0.01 mSv with sufficient flexibility to evaluate every case on its own merit. Recently, the NRC has proposed a formalization of the process [4] with dose equivalents between 0.01 and 0.1 mSv considered to be de minimus. The most likely dose limit is, however, 0.04 mSv, a limit permitted under the EPA's drinking water standards.

There is no de minimis process in the RCRA waste system. However, the law permits a "delisting" of a listed waste. The process is, however, costly, cumbersome, and slow. In effect one is asking for a waiver from existing regulations.

OPTIONS FOR MANAGEMENT OF MIXED WASTE

From the foregoing, it becomes apparent that proper management of mixed AEA-RCRA waste is associated with significant regulatory and institutional constraints. Accordingly, the goal of the waste manager should be to assure a separation of AEA and RCRA wastes.

Mixed waste generated in the U.S. can be categorized in three distinct categories. These are 1) AEA waste containing organic RCRA waste, 2) AEA waste containing inorganic RCRA waste, and 3) AEA waste containing organic and inorganic RCRA waste.

The treatment option for the first group is reasonably straightforward. In these cases, incineration is the preferred treatment option. Much of the waste generated in medical and biological research facilities, such as animal carcasses and liquid scintillation waste, can be incinerated advantageously with appropriate off gas treatment techniques. Tritium and carbon-14 are usually released to the atmosphere and other radionuclides, such as isotopes of iodine and phosphorous-32, are collected and stored for decay. The incinerator ash can be then disposed as AEA waste. Probably much of the waste generated in the nuclear power industry would also fall in this category. For example, it is likely that once hydrazine, EDTA, organic solvents, various oils and other organic materials are combusted, much of the mixed AEA-RCRA waste from a boiling water reactor is no longer a mixed waste. A

similar situation exists for pressurized water reactors. In this case, certain materials have to be either replaced or treated. For example, chromium-VI should be converted to chromium-III.

The second group mentioned above is clearly mixed RCRA-AEA waste. One should avoid generating it and should attempt to reduce its volume by evaporation, incineration, compaction, and other techniques common in the radioactive waste technology.[5]

The third group can also be treated like the first group. However, the product resulting from the treatment may remain RCRA waste. Again, here, one can only try to reduce the volume of the waste by avoiding its generation and treating it once it has been produced.

One should be aware that the costs of disposal of AEA waste are increasing rapidly, not only in the U.S., but also in many other countries. In the past, the chemical industry implied that it could not afford to follow the rules of AEA waste because of economic constraints. Meanwhile, the cost of disposal of RCRA waste has increased to a level that it significantly exceeds the disposal of AEA waste.

Immobilization techniques used in the radioactive waste technology were designed to contain inorganic materials. In general, they are poor for immobilization of the numerous organic compounds occurring in the mixed waste. The most commonly used agent, cement, is the example of solidification material not readily useful for immobilizing organic materials.

CONCLUSIONS

Institutional and legal constraints are the primary cause of the problems encountered in the U.S. in managing mixed AEA-RCRA waste. The strategy for solving the mixed waste problem should consist of several elements:

1. A concerted effort to replace RCRA materials with other materials
2. Waste minimization
3. Separation of RCRA waste from AEA waste
4. Volume reduction
5. Development of performance-based waste management options for mixed waste
6. Legislative and Regulatory efforts to remove current institutional and legal constraints for a risk-based waste management system

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RADIOLOGICAL IMPACT OF DISPOSING OF HIGH LEVEL WASTE IN A GRANITIC REPOSITORY IN ARGENTINA

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INTRODUCTION

The Argentine nuclear programme is based on natural uranium and heavy water reactors. Besides, it is scheduled the use in these reactors of very low enrichment uranium and recycling of plutonium produced in this fuel cycle.

Although no significant amounts of liquid high-activity wastes have been produced in Argentina, their production is expected as from the operation of a fuel element reprocessing plant that is now in an advanced stage of construction.

The repository must satisfy the needs of six nuclear stations operating during 30 years, which will generate about 80 GW.a of electrical energy. The wastes resulting from the reprocessing of spent fuel elements for the generation of that energy will require approximately 3000 containers of about 0.6 m in diameter and 1.6 m high.
Detailed studies in a preselected site, at Sierra del Medio, Province of Chubut, are being carried on in order to obtain specific design parameters for the repository construction.

RADIOLOGICAL SAFETY CRITERIA

Radioactive waste disposal involves isolation of the radioactive material from the human environment over a sufficient period to permit its decay to clearly acceptable values. Any exposure of man that could still occur would be due to mechanisms that may be described as normal scenario or random disruptive events. These ones may cause releases before there is sufficient decay to make the exposures acceptable.

Criteria for Normal Scenario:

The radiological safety criteria for normal scenarios adopted by the Argentine authorities are in agreement with the ICRP recommendations (1).The main objectives are to keep individual doses below appropiate limits and to reduce the total radiological impact from the disposed wastes as far as it is reasonably achievable. In order to limit the contribution from any particular source, an "upper-bound" is fixed, which should be only some fraction of the ICRP dose limit for individual doses. For planning purposes, the Authority has established this value at 0.1 mSv in a year as a maximum dose in a hypothetical critical group.

In the case of waste repositories, and owing to the low doses that are expected to be involved, the total radiological impact or detriment can be taken to be proportional to the collective dose commitment from the disposal (2). Because of the presence of very long lived nuclides, an assessment of the complete collective dose commitment will be highly speculative. Therefore, the incomplete collective dose commitment integrated over 500 years was used for optimization purposes, following procedures similar to those used by the UNSCEAR (3). The Argentine authorities also established as a design aim of the repository that the total collective dose commitment from high level radioactive waste disposal should be of the same order as that resulting from the other steps in the fuel cycle.

Criteria for disruptive events

The Argentine criteria implies as a basic objective that the maximum individual risk to a member of the public due to potential disruptive events in a repository does not exceed the risk due to the "normal" scenario.
The upper bound of 0.1 mSv in a year corresponds to a risk of the order of 10^{-6} in a year. Therefore, the proposed objective was stated as R max.$\leq 10^{-6}$, where R max. is the maximum value of the probability of lethality commited for one year. If about ten disruptive event sequences could be identified, then their individual risk contribution should not exceed a value of Ri=10^{-7} in a year.

MULTIPLE BARRIERS CONCEPT

To comply with the radiological safety requirements, proper isolation of the high level radioactive wastes from the biosphere will be provided by means of engineered and geological barriers, independents and redundants, in such a way that any potential failure of one of them does not interfere with the performance of the others. Engineered barriers consist in the incorporation of the wastes in a vitreous matrix of the borosilicate type (leach rate below 4.10^{-4} g cm^{-2} a^{-1}), with a content of fission products and actinide oxides in the glass mass of 10 wt %. The vitreous matrixes into stainless steel containers will be covered by 10 cm lead. An additional engineered barrier will be formed with the final sealing of the repository holes. These barriers will provide an initial isolation period of the order of 10^{4} years, time enough to ensure low individual risk to members of the public. Also, to reduce the global detriment, the geological barriers of the site should retard the arrival of the radionuclides in the biosphere for a period of the order of 10^{5}–10^{6} years. To achieve such isolation, the containers will be placed within a granitic formation with low hydraulic conductivity (< 10^{-9} m s^{-1}), several hundred meters deep (4).

MAXIMUM INDIVIDUAL DOSES

a) Normal scenario

The most likely release scenario implies leaching of radioactive wastes and radionuclide transport by groundwater up to the biosphere. An initial isolation period of 1000 years will be assured by design, covering the canisters with 10 cm of lead. After that, nuclide leaching from the vitreous matrix by groundwater will occur, and significant quantities of radioactivity will reach the biosphere (5). Backfill retardation was disregarded.

The model used for the nuclide migration through the geosphere was a one dimensional transport model, which includes axial dispersion, geochemical retardation and radioactive chain decay, with input parameters similar to those used by other authors for granitic formations (6,7). The recipient surface water body was assumed to be a very low water flow river (0.1 $m^3 s^{-1}$), located near the edge of the repository. Estimation of radiological consequences needs to define the location, habits and metabolic characteristics of the hypothetical individuals concerned. Because of the long times involved in waste disposal, the location and time at which individuals lived was assumed to be those where and when the doses were maximum. The habits of the individuals were defined on the basis of present human life needs using conservative but realistic assumptions (8).

Pathways that have shown to be important are internal contamination via ingestion of water , fish and contaminated food from irrigation practices. Among the actinides, Np-237 was identified as the most important nuclide, being its contribution to the maximum individual dose of the order of 4.10^{-4} Sv/a, but this dose would be incurred about 5.10^{5} years after leaching process begins. The contributions due to Ra-226 and Pu-239 would be about 10^{-6} Sv/a and 10^{-11} Sv/a, occurring around 10^{6} and 10^{8} years after leaching begins, respectively. Long-lived fission products such as I-129 and Tc-99 would deliver doses of 10^{-8} Sv/a each, a few hundred years after leaching because the corresponding retardation factor in granite is estimated to be 1. A rough optimization analysis shows that the construction of the repository 500 m deep would reduce the detriment as low as reasonable achievable.

b) Disruptive events

Important damages on hard crystalline rock, at 500 m deep, could be produced only if violent natural events, such as large meteorite strikes or volcanism occurred. Assuming total failure of the geological barriers as a consequence of one of such highly improbable events (probability of occurrence< 10^{-13} per annun), individual doses up to several $Sv.a^{-1}$ could be incurred, depending on the period of occurrence of the causing event. However, these high doses could be reduced to acceptable levels by avoiding water consumption from the affected area. Tectonic activity, giving rise to faulting or fracturing of the granitic formation, has been estimated for the region of Sierra del Medio to have a probability of occurrence of 10^{-8} to 10^{-7} per annun, and transport models applied to these conditions

(fractured rock) in previous works gave as resulting annual doses in the most exposed individuals of some mSv (7).

TOTAL COLLECTIVE DOSE COMMITMENT

The total collective dose commitments from the Argentine repository for delay periods of 10^4, 10^5 and 10^6 years, were estimated to be $8.2.10^4$, $4.6.10^3$ and $6.8.10^2$ man Sv, respectively (2).

CONCLUDING REMARKS

The radiological consequences of high level radioactive waste disposal from the Argentine programme should imply doses to an hypothetical critical group of the order of 10^{-4} Sv/a, due mainly to Np-237. These doses would be incurred some 10^5 years after the waste disposal.
The total collective dose commitment associated with the expected delay time in reaching the biosphere of 10^5-10^6 years for the radioactive wastes, represents at least one order of magnitude less than the corresponding value for the operation of the nuclear plants that generate these wastes (2).

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OPTIMIZATION OF OPERATIONS INVOLVED IN ASSUMING RESPONSIBILITY FOR LOW ACTIVITY WASTES IN FRANCE

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ABSTRACT

In addition to the basic nuclear installations generating the largest amounts of radioactive wastes as far as both volumes and activities are concerned, many smaller producers of such wastes also exist (university - hospitals establishments, research centers,...) presenting the following characteristics :

- high degree of geographical dispersion,
- generation of small quantities of wastes that have to be collected periodically,
- generation of wastes of very varied natures (physicochemical properties and chemical and biological risks associated with radioactive risks),

All the expenses involved in assuming responsibility for these wastes have been analyzed in detail in order to define and dominate the problems likely to occur.

Different sources of expense having led to the adopting of special technological or methodological solutions are analyzed :

- accounting for dangerous materials and the management involved in every thing from assuming responsibility to invoicing clients (our choice : computerized management),
- packaging dangerous materials and pre-storage prior to controlled disposal (our choice : standardization),
- collecting radioactive waste packages by road vehicules (our choice : specialization).

The operational results obtained from both the safety and financial points of view demonstrate that the choices made have led to an efficient optimization of all operations involved in assuming responsibility.

HANDLING OF RADIOACTIVE LIQUID WASTE

ANCARANI ROSSIELLO LAURA
ENEA

INTRODUCTION

For the decontamination of radioactive liquid wastes and extraction of radionuclides, we propose the "foam separation" method. Studied and illustrated in a previous paper (1), this method has a high efficiency, in very dilute solution, in those cases where traditional methods meet with economic and practical limitations. For many years, in both chemical and nuclear applications, the "foam separation" method has been used with success.

The proposed method permits the selective extration of only one ion or the extension of the extraction to a large spectrum of ions. Moreover, the extracted ions are concentrated in extremely small volumes (the reduction of total volume of treated liquid is about 1/1000).

EXTRACTION TECHNIQUES

In order to effect extraction, for solutions containing various ions in very low concentrations, it is sufficient to introduce a surfactant, a substance capable of adsorbing such ions at the liquid-air interface. A stream of gas passed through the liquid causes the surfactant to foam: the foam thus formed is the technical means which permits the collection of this mono-layered liquid.

The efficiency of estraction by means of foam separation is conditioned by numerous parameters, among which the most important are: the pH value and the surfactant concentration.

EQUIPMENT AND RESULTS.

The basic equipment required is very simple: Fig. 1 (1).
For extraction with recycling: Fig.2 (2).
Some results obtained employing the "foam separation" method are reported in Table 1.

RISKS-BENFITS AND CONCLUSIONS

The final destination of low-level radioactive liquid wastes is the acquatic environment; this involves negatives effects to the ecologic system.

There exist, as above mentioned, studies to establish either acceptable limits within which such discharges can be carried out, or dilution processes. In any case, in a hydrobiological system, radionuclides can at first be dispersed in surface waters. From these, through suspended materials, sediments, acquatic plants and animals, the radionuclides can be transmitted to the terrestrial environment and to man. For example, figure 3 (3) reports the ways through which radioactive substances dispersed in deep water

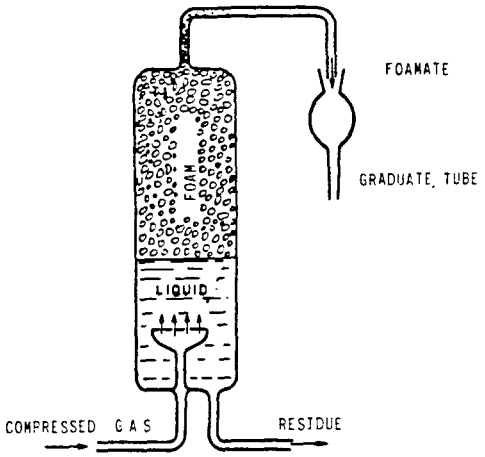

Fig. 1(1) Schematic diagram of a single stage apparatus

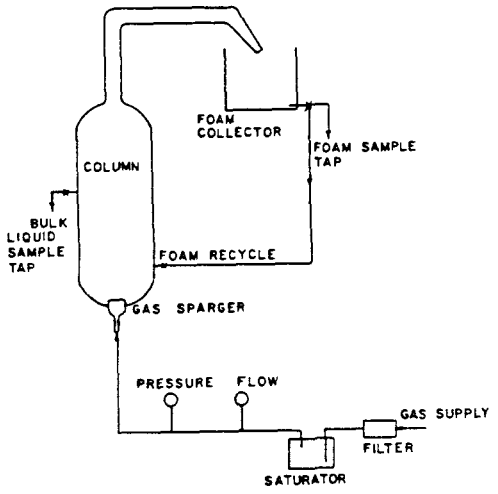

Fig. 2. Schematic apparatus of Schnepf *et al.* (2). The mechanical foam breaker, not visible in the picture, is mounted in the collector.

Fig 3(3). *Simplified diagram showing the ways in which radioactive substances dispersed in ground waters or surface waters (ocean included) reach man. (1) radioactive substance; (2) direct radiation; (3) air; (4) aquatic plants; (5) sand and sediments; (6) fishing vessels and leisure facilities; (7) surface or ground waters; (8) aquatic animal; (9) man; (10) irrigation water; (11) radioactive substances; (12) land plants; (13) ground; (14) land animals; (15) ingestion.*

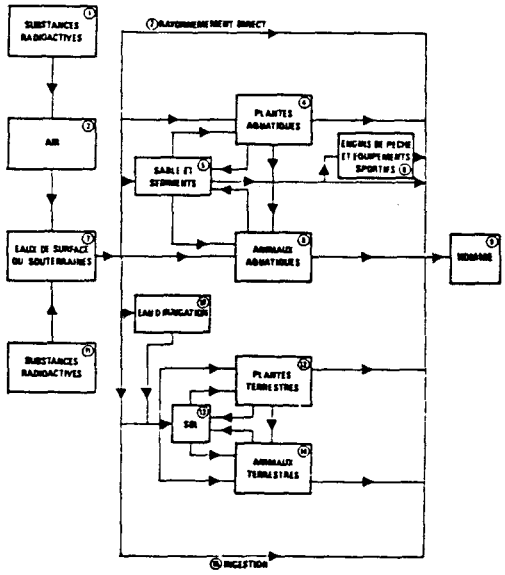

Table 1. Some results obtained employing the "foam separation" method.

Ions	Surfactant	Extraction %	Reference
^{90}Sr	Sodium salt of dodecylbenzyl-diethylene-triamine tetraacetic acid (DBDTTA)	99	4
^{144}Ce		98	
^{137}Cs		10-20	
^{95}Zr	By precipitation in $Fe(OH)_3$ e $Co(OH)_2$, and floated using sodium oleate		5-6
^{95}Nb	- when a single operation was used	98	
^{106}Ru	- with four successive flotation on the same solution	90	
^{106}Rh			
^{141}Ce			
Zn^{2+} ^{65}Zn	dodecylimino-dipropionic acid (Deriphat 160)	92	7
Ca^{2+} ^{45}Ca as tracers		99	
Sr^{2+} ^{89}Sr		91	
Ba^{2+} ^{139}Ba		95	
Cr^{6+}	sodium dodecylsuphate after reduction to Cr^{3+}	97	8
Pb^{2+}	sodium lauryl suphate	99	9
Cd^{2+}	4-dodecyldiethylenetriamine	98-99	10

(including oceans) or in surface waters can reach man.

It is indispensable therefore to try as much as possible to reduce the level of pollution by treating the waste effluents prior to discharge into the acquatic environment.

The technique proposed and implemented by us, as previously mentioned, resolves, perhaps entirely, the problem.Infact, most of the methods currently used for the treatment of radioactive liquid wastes have the strong disadvantage of eliminating one pollutant by the employment of another pollutant.

Our method, instead, produces none or very small quantities, of wastes. Therefore, the benefits to be derived are evident.

The potential utility of the "foam separation" method arises from technical and economic feasibility.

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PHYSICAL AND CHEMICAL PROPERTIES AND COMPOSITIONS OF LIQUID EFFLUENTS OF A NUCLEAR POWER REACTOR

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Behaviors of the released radioactive materials from a nuclear power plant to the marine environment depend largely on their physical and chemical properties. It is therefore important to grasp the compositions of the liquid effluents. From this point of view, some kinds of drainwater of a BWR plant were sampled(Figure 1). These samples are the laundry drain water at charcoal bed inlet and outlet, and the floor drain water. Suspended solids were separated from these samples physically with the step filtration method and ions chemically with the ion exchange resin. Radioactivity of Co-60, Mn-54 and Cs-137 was measured with γ-spectrometer. Suspended solids of the laundry drain samples were also analysed with the instrumental methods(differential thermal analysis, X-ray diffraction analysis and X-ray fluorescence analysis).

It was found that the majority of radionuclides Co-60 and Mn-54, in lalundry drain water, were suspended in the form of solid, while that of radionuclide Cs-137 solved in the liquid. The radioactivity on the particles of about 10μm-size was high, and the particles were identified as the organic matter, the mineral and so on. In the liquid phase, radionuclide Cs-137 was found as mineral compound(Figure 2 and Figure 3).

Decontamination factor and decontamination characteristics for the laundry drain water were also investigated in order to verify the performance of the filtering system. It was found that the decontamination factor of the charcoal filter was about from 3 to 10, and radionuclides Co-60 and Mn-54 in the liquid was removed to some extent while radionuclide Cs-137 was removed little(Table 1).

In the samples of floor drain water, radionuclides Co-60 and Mn-54 were found both in the form of the suspended solid of about 10μm-size and in the form of the mineral compound in the liquid phase(Figure 4).

A chemical analysis method was also established in order to measure the extremly low concentration of radionuclide Co-60 in the sea water(1000 ℓ). The radioactivity of 0.2~0.7pCi/1000ℓ could be detected with this method at the frontal sea area of the small bay where the liquid effluents of BWR plant are released.

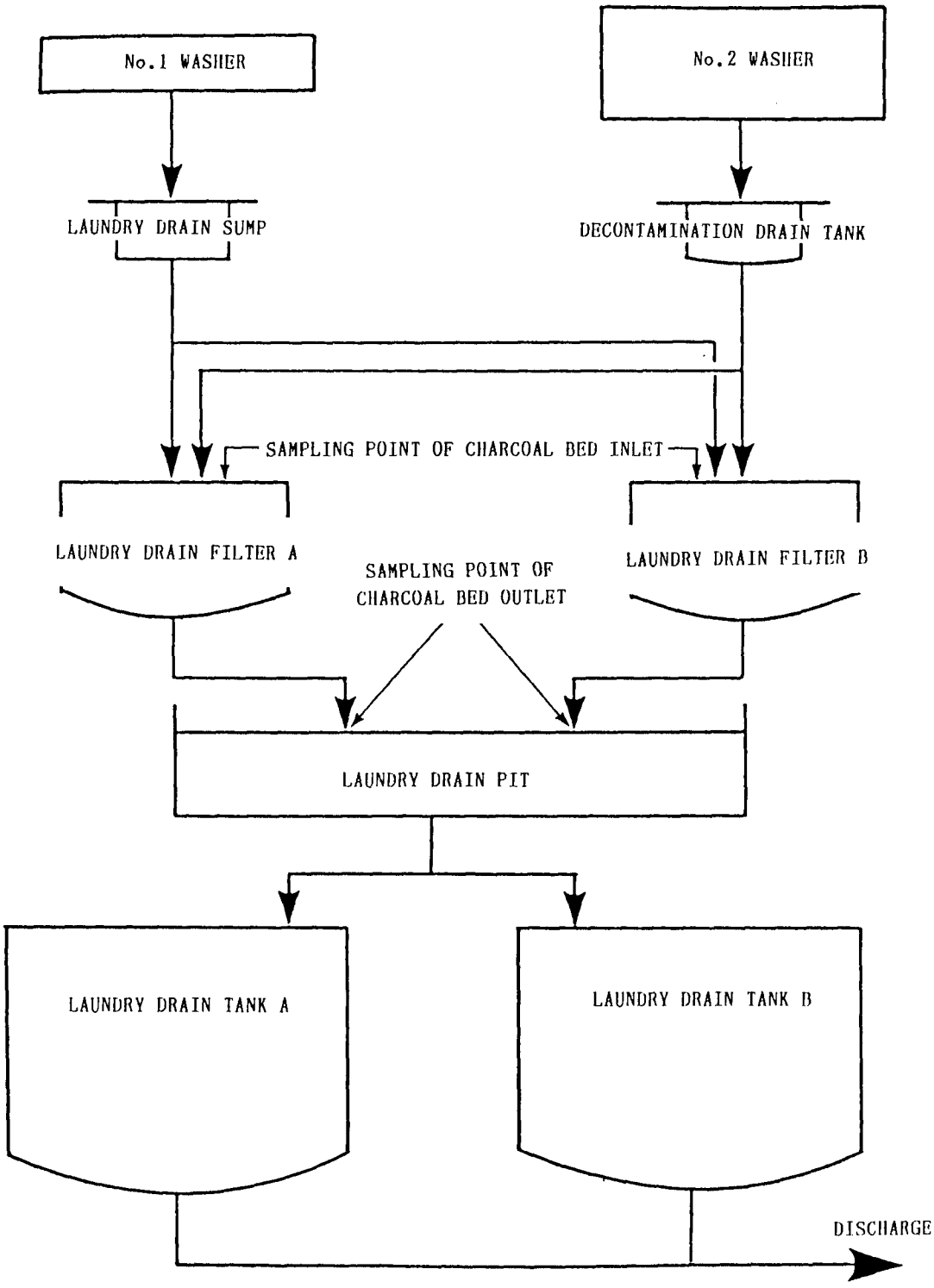

Figure 1 Sampling points of laundry drain water

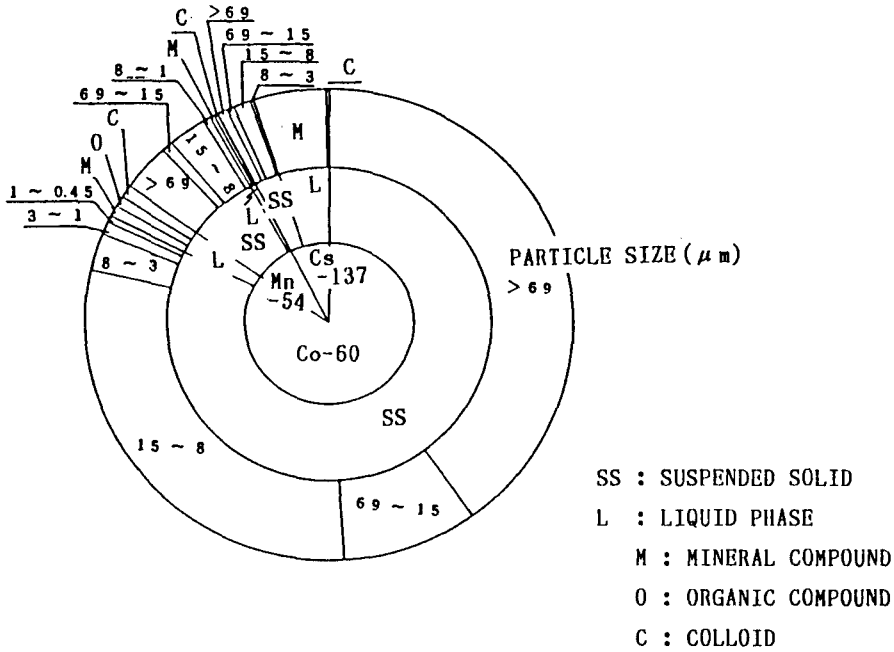

Figure 2 Composition of laundry drain water charcoal bed inlet

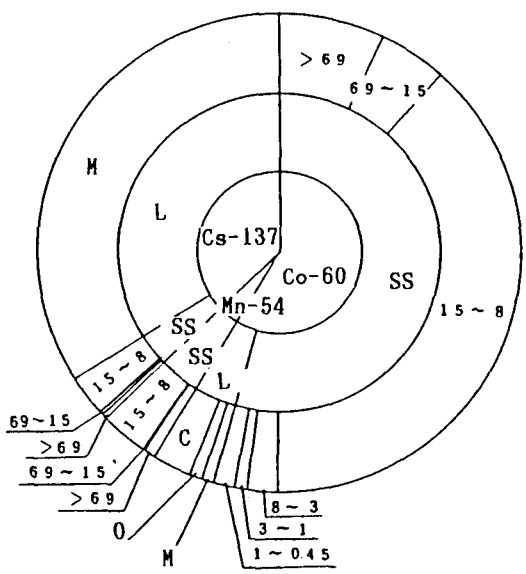

Figure 3 Composition of laundry drain water charcoal bed outlet

Table 1 Radioactive concentration and decontamination factor of laundry drain charcoal bed

		Co-60 (pCi/l)	Mn-54 (pCi/l)	Cs-137 (pCi/l)	Total (pCi/l)
Normal operation (typical data)	Charcoal bed inlet	1327.7	107.4	37.7	1472.8
	Charcoal bed outlet	105.4 (DF 1/13)	5.0 (DF 1/21)	30.8	141.2 (DF 1/10)
Maintenance outage (averaged data over typical two weeks)	Charcoal bed inlet	445.1	42.8	68.2	556.1
	Charcoal bed outlet	138.3 (DF 1/3)	11.4 (DF 1/4)	58.7	208.4 (DF 1/3)

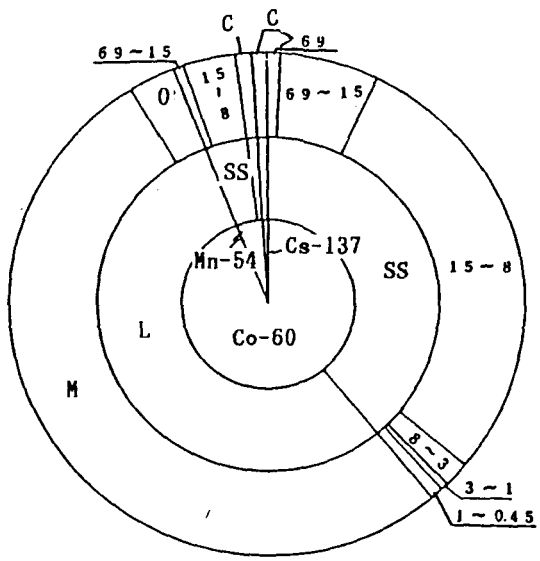

Figure 4 Composition of floor drain water

ESTIMATION OF DOSES TO WORKERS AND THE PUBLIC IN THE JPDR DECOMMISSIONING

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INTRODUCTION

Japan Power Demonstration Reactor (JPDR, BWR, 90MWt) went into power operation in 1963 and finally shut down in 1976. It had played an important part as a pilot of Japanese nuclear power plants till the end of service. The dismantlement of the JPDR started on December in 1986 and will take about 7 years to complete. The purposes of the dismantlement are to demonstrate the complete dismantlement of JPDR using the developed techniques and to accumulate practical data which will be beneficial to dismantling of commercial power plants in the future. This paper describes the estimated results of the collective dose to workers and the maximum individual dose to the public, [1],[2] including the calculation method based on the dose rates in the working area and the environment, man power and the dismantling methods.

ESTIMATION OF COLLECTIVE DOSE TO WORKERS

The dismantlement of reactor internals, reactor pressure vessel(RPV) and radiation shielding concrete will be very noticeable works from the view point of the radiation control. The dose rates were calculated to be maximum 3.6 R/h around the RPV (with water), 0.2 R/h inside the radiation shielding concrete (after removal of the RPV) and 1 R/h around the primary cooling pipes. In the JPDR decommissioning, " remote underwater cutting "," disk cutter " and " controlled blasting " will be adopted in order to reduce the dose to workers and the released amount of radioactivity to the environment. The local exhaust devices with HEPA filters and/or temporary containment rooms will be prepared in the places where air contamination might occur. The radioactivity inventory data of activated components such as the reactor internals and the others were taken from the calculated results [3] and the measured values, respectively.

Collective external dose

Collective doses in each working area were estimated by multiplying the average dose rate in the area by man power (man•day). The average dose rates in the working areas were calculated for the dismantling of the RPV and the radiation shielding concrete by QAD-CG code. The collective dose obtained is shown in Table 1. In the dismantling of the reactor internals about one half of the collective dose (0.17 man•Sv) results from decontamination and transport of waste packages. And in the RPV dismantling about one half of it (0.26 man•Sv) also results from preparation of the RPV cutting.

This work was performed by the JAERI under contract from the Science and Technology Agency of Japan.

Table 1. Collective Dose to workers

Building	Dismantled component or structure	Manpower (man•day)	Collective dose (man•Sv)		
			External	Internal	Total
Reactor Building	Reactor Internals	9,200	1.7×10^{-1}	2.3×10^{-4}	1.7×10^{-1}
	RPV	4,100	2.6×10^{-1}	2.5×10^{-4}	2.6×10^{-1}
	Radiation Shielding Concrete	4,500	7.9×10^{-2}	1.3×10^{-4}	7.9×10^{-2}
	Others	17,700	1.5×10^{-1}	3.7×10^{-4}	1.5×10^{-1}
	Total	35,500	6.6×10^{-1}	9.8×10^{-4}	6.6×10^{-1}
Others	Whole installations	37,500	3.6×10^{-1}	5.4×10^{-4}	3.6×10^{-1}
Total		73,000	1.0×10^{0}	1.5×10^{-3}	1.0×10^{0}

Collective internal dose

The dispersed amounts of radioactivity in each dismantling work were calculated using the radioactivity inventory data, the R&D data and so on [4],[5]. The parameters used for safety analysis are shown in Table 2. The collective internal dose was estimated by multiplying average internal dose rate by man power. For the estimation of average internal dose rate the airborne radioactivity concentration was determined mainly based on the DAC of ICRP publication 30 and the above parameters. Then it was assumed that a worker wore a half-mask during operation. The results are also shown in Table 1. The collective internal dose will be 1.5×10^{-3} man•Sv through the JPDR decommissioning, and it will be reduced to a negligible level by the above air contamination protections compared with the collective external dose.

ESTIMATION OF INDIVIDUAL DOSE TO THE PUBLIC

The doses to the public during dismantling are classified as follows.

- internal dose by inhalation and external dose by radioactive plume arisen from gaseous effluent
- internal dose by intake of seafoods grown in ocean where liquid effluent was discharged.

Dose due to gaseous effluent

The released radioactivity from the stuck would be estimated to be 1×10^{6} Bq(3×10^{6} MeV•Bq) through the JPDR decommissioning by using parameters shown in Table 2. The maximum individual dose to

the public was estimated at a point where the maximum dose was expected, assuming that total activity was released into environment at a constant rate through a year. The individual dose obtained is shown in Table 3.

Table 2. Parameters used for safety analysis.

Item of parameter		Value
Dispersion factors for radiation shilding concrete cutting with	abrasive water jet	0.4g/min
	diamond saw	0.06g/min
	controlled bluster	0.03g/kg
Dispersion factor for reactor internals cutting with underwater plasma arc torch		0.05~5g/m
Dispersion factor for RPV cutting with underwater arc saw		0.2kg/m^3
Dispersion factor for equipment & piping cutting with gas torch		3cm^2/kg
Dispersion factor for decontamination of tanks with water jet		0.01g/m^3
Dispersion factor for decontamination of contaminated concrete with planner		0.08g/m^2
Resuspension factor for decontamination of floor		$5\times10^{-6}cm^{-1}$
Leakage of half mask		0.1
Leakage of temporary containment room		0.1
Penetration of HEPA filter for exhaust attached to temporary containment room		5×10^{-4}
Penetration of HEPA filter for exhaust attached to facilities		1×10^{-2}

(1) Internal dose by inhalation

The annual average air concentrations were calculated with all the 16 direction sectors and the maximum value of them was used for dose calculation. If a release rate of 2×10^{2}Bq/h continues for a year, the maximum value of annual average radioactivity concentration in the air would be 2×10^{-8}Bq/m^3in the residential district. Deriving from these values, the internal individual dose to the public at the site boundary would be estimated to be 2×10^{-13}Sv by inhalation.

(2) External dose by radioactive plume

If a release rate of 4×10^{2}MeV•Bq/h continues for a year, the maximum annual dose would be estimated to be below 7×10^{-14}Sv in the residential district outside the site boundary.

Table 3. Individual doses to the public

Effluent	Pathway	Radioactivity	Dose
Gaseous	Inhalation	1×10^{6}Bq (2×10^{2}Bq/h)	2×10^{-13}Sv
	External	3×10^{6}MeV•Bq (4×10^{2}MeV•Bq/h)	7×10^{-14}Sv
Liquid	Intake of seafood	1×10^{9}Bq (1×10^{5}Bq/h)	7×10^{-8} Sv
Total			7×10^{-8} Sv

Internal dose due to liquid effluent

Total radioactivity discharged into ocean would be 1×10^{9}Bq through the JPDR decommissioning project. If a discharge rate of 1×10^{5} Bq/h continues the internal dose by intake of seafoods (seaweed, fish, non-vertebrate animal) would be estimated to be 7×10^{-8}Sv.

CONCLUSION

In the JPDR decommissioning project, the collective dose to workers and the maximum individual dose to the public would be 1man•Sv and 7×10^{-8} Sv, respectively. From the dose estimation, we believe the JPDR decommissioning will be performed with radiological safety.

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ANALYSIS OF OCCUPATIONAL RADIATION EXPOSURE IN REPROCESSING PLANT OF PNC, TOKAI

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INTRODUCTION

Reprocessing plant of PNC in Tokai-mura (Tokai-plant) began the operation in 1980, however the operation was suspended in 1983, because of the failure of acid recovery evaporator and the pinholes appeared at dissolvers.

While repairing the dissolvers with remote welding method and the acid recovery evaporator with direct maintenance by workers under high radiation exposure in cell, PNC decided to install the third dissolver to ensure the stable operation of the plant.

After the installation work of the third dissolver performed in 1984, the plant operation was started again in 1985. The cumulative amount of spent fuel reprocessed reached to 347 tons as of May 1987.

The averaged number of radiation workers engaged in operation and maintenance work is around 2,000 persons/year and the averaged annual collective dose is around 0.8 man·Sv.

In this paper, the authors give an empirical formula for the annual individual dose distribution of Tokai-plant, which was acquired through the analysis of past annual individual dose distributions.

RADIATION CONTROL

The radiation control of the plant has two major aspects. The one is the routine radiation control for the routine radiation works associated with regular plant operation. The other is the non-routine radiation control for the non-routine radiation works such as maintenance works and reform of process equipments.

Non-routine radiation work is further divided into two categories, namely special radiation work (SRW) and second grade

radiation work. The non-routine radiation works in which the pre-operational estimated dose for whole body, extremities or skin could exceed respectively 3 mSv, 15 mSv or 6 mSv per week are to be classified as SRW to be put under more strict radiation control.

Non-routine radiation works other than SRW are classified as second grade radiation work. Safety regulation of the plant prescribes some radiation protection criteria to keep individual dose as low as reasonably achievable. Table 1 shows a part of the dose limitation system for whole body.

Table 1 Dose limitation System at Reprocessing plant of PNC, Tokai for whole body

Investigation level	Action level	Dose limit
3 mSv/3 months	13 mSv/3 months	30 mSv/3 months

ANNUAL INDIVIDUAL DOSE DISTRIBUTION

The individual dose distribution of the year in which the plant was operated relatively smooth (Operation-year), namely 1981, 1982, 1985 and 1986, form Long-normal distribution, as is seen in Fig. 1. Year denotes fiscal year and this is the same hereafter.

On the other hand, the individual dose distribution of the year in which large scale SRW's auch as maintenance of the acid recovery evaporator (E30-Work) and installation of the third dissolver (R12-Work) were performed (LSRW-year), namely 1983, 1984, has sharp increase in higher dose region, therefore the individual dose distribution dose not form Long-normal distribution. This is shown as curved distribution in Fig. 2. Radiological data of E30-Work and R12-Work are presented in Table 2 to help realize the magnitude of these radiation works.

Table 2 Radiological data of E30-Work and R12-Work

	Period [Day]	Workers [person]	Average dose [mSv]	Collective dose [man•Sv]
E30-Work	190	320	4	1.2
R12-Work	150	340	4	1.3

ANALYSIS OF INDIVIDUAL DOSE DISTRIBUTION

Individual dose distributions of E30-Work and R12-Work were precisely examined.

As a result, it was found in each case that there is a group of workers with higher individual dose. And that the individual

dose of the group forms *Normal* distribution as is shown in Fig. 3. Substracting this group from the whole workers who received higher dose than detection limit (curved distribution in Fig. 2), we had the same Log-normal distribution (dotted distribution in Fig. 2) as in Fig. 1.

This leads us to conclude that the distribution of LSRW-year has two components, namely Log-normal distribution and Normal distribution. The latter is specific to LSRW-year.

CONCLUSION

The empirical formula for the annual individual dose distribution of Tokai-plant (F) is given as follows:

$$F(x) = \frac{\beta N}{\sqrt{2\pi}} \int_0^X \left[\frac{(1-\lambda)}{\sigma_f X} \mathrm{Exp}\left\{ -\frac{(\ln X - \mu_f)^2}{2\sigma_f{}^2} \right\} + \frac{\lambda}{\sigma_g} \mathrm{Exp}\left\{ -\frac{(X-\mu_g)^2}{2\sigma_g{}^2} \right\} \right] dx$$

Table 3 Parameters for the annual individual dose distribution function

x	Individual dose	-	[mSv]
N	Total radiation workers in a year	1,500∿2,900	[person]
β	Ratio of workers with measurable dose to N	0.31 ∿ 0.46	
σ_f	Standard deviation of Log-normal distribution	1.0	[mSv]
μ_f	Mean of Log-normal distribution	0.6	[mSv]
σ_g	Standard deviation of Normal distribution	2.7	[mSv]
μ_g	Mean of Normal distribution	10.0	[mSv]
λ	Ratio of workers with Normal distribution to βN	Operation-year : 0 LSRW-year : 0.1	

REMARKS

As for the specific radiation work that could give rise to higher individual dose, it is one of the conceivable choices to increase the number of workers to be employed in the work in order to reduce the number of workers who would receive higher dose.

However this could possibly cause the increase of collective dose as a whole.

This is the subject left to be discussed from the view point of the Optimization of Radiation Protection.

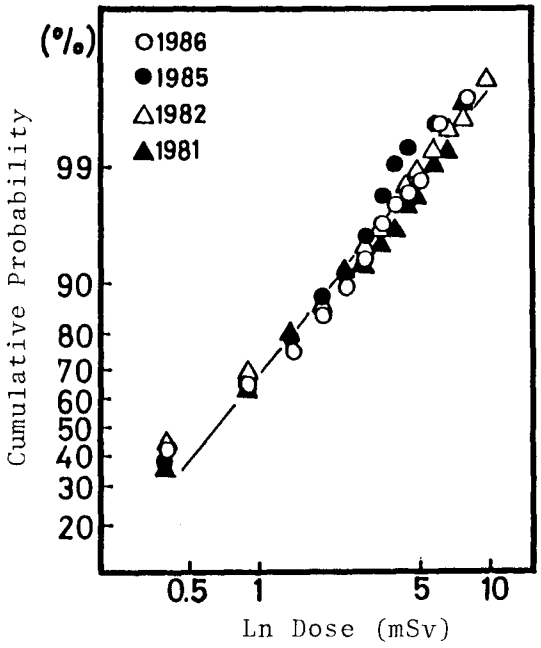

Fig. 1 Annual individual dose distribution of Operation-year*.

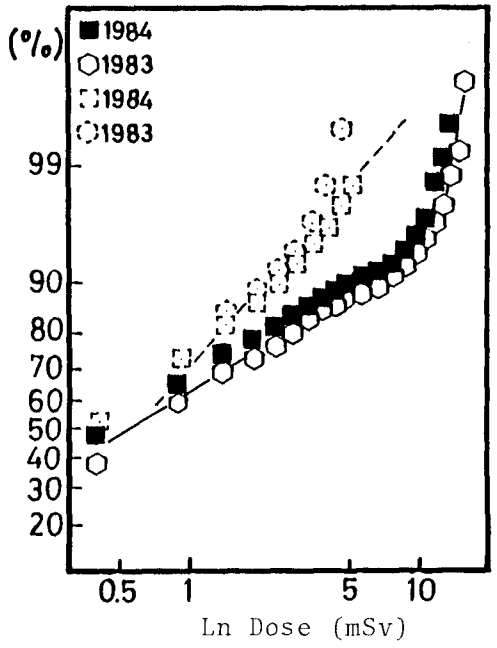

Fig. 2 Annual individual dose distribution of LSRW-year* (Curved) and modified distribution (Dotted).

* Definition is give in the text.

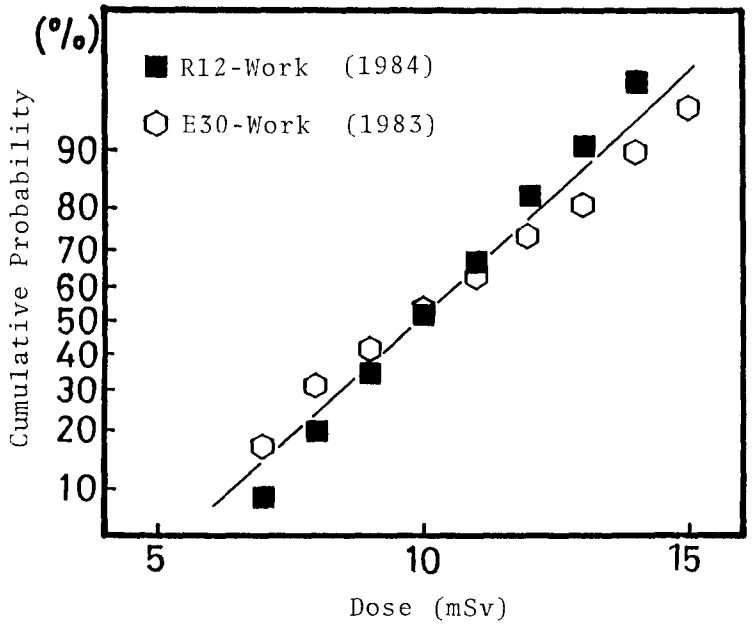

Fig. 3 Annual individual dose distribution of workers who received higher dose than other workers in R12-Work and E30-Work.

OPTIMISATION DE LA PROTECTION DES REACTEURS DE FUSION, CAS DU TRITIUM

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Summary :

The purpose of this study is to contribute to the achievement of the protection systems for fusion reactors, and to look for the best options to protect from tritium, which is the major readioactive element issued from fusion reactors. In this paper a computer tool, the TRITO code is presented. It defines the detritiation system, function of tritium release (gas HT or water HTO). It respects limitations on workers and public doses ; it calculates deposits and water and air concentrations of tritium in the different containments, it estimates also the resulting costs, with regards to wastes, worker and public doses. It allows thus to compare different options, to assess the less costles hall volume and source of tritium (coming into the primary circuit of cooling water), or moreover to find the more interesting flow rate of water sampling in this circuit. Furthermore, it looks for the best operating system of dryer, ventilation, and water and air detritiation.

1. INTRODUCTION

Le Tritium des Tokamaks réalisant la fusion du Deutérium-Tritium peut être un des risques majeur de contamination pour les travailleurs, le public et l'environnement, la première étape consiste à définir le chemin de cette contamination. Les barrières interposées entre les sources de contamination et le public sont imposées soit par le procédé lui-même (paroi de la chambre torrique), soit par d'autres critères de sûreté (protection contre l'irradiation externe, par les neutrons ou les rayonnements γ). Un des buts est de définir, si ces méthodes de confinement sont suffisantes, pour limiter les doses possibles aux travailleurs et au public, et si le gain obtenu, en équivalent de dose effectif, avec des barrières suplémentaires est "rentable". En effet, une surprotection peut devenir très coûteuse, pour une réduction minime des équivalents de doses.

Le code TRITO (TRItium Transfer Optimization) optimise, c'est-à-dire choisit, les différentes options de protection possibles, correspondant au coût-bénéfice le plus avantageux en respectant les limites des équivalents de doses travailleurs/ public. Ce code compare différentes installations du hall avec une ou deux enceintes, recouvertes ou non d'une couche d'acier. Il définit quel est le meilleur fonctionnement des systèmes de détritiation de l'eau et de l'air, du sécheur et les débits de ventilation. De plus, TRITO peut optimiser les fuites de routine ou les fuites résultant d'une situation accidentelle.

2. METHODOLOGIE

2.1. Radioprotection

La première règle est de **limiter les concentrations en tritium de l'air** (C1)

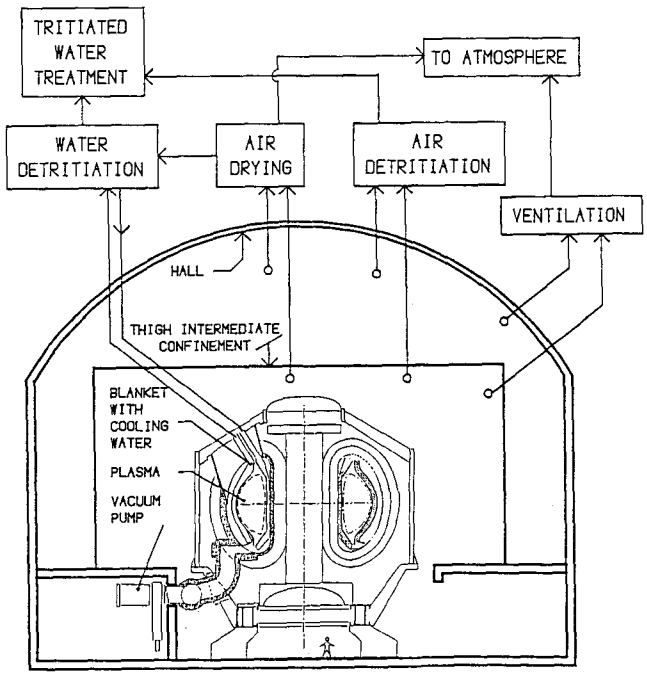
que peuvent respirer les travailleurs, à une valeur choisie comprise entre 1 et 100 CMA (1 CMA = 8.14 10^5 Bq/m^3). La deuxième règle vise à **limiter les rejets dans l'environnement.** Le modèle TRITO limite les rejets atmosphériques à une certaine valeur, qui est ici de l'ordre de 3.7 10^{13} Bq/an (1 000 Ci/an). La troisième règle est plutôt un **critère de sélection, qui consiste à diminuer au maximum les coûts** du système de protection, en respectant les deux premières règles.

2.2. Identification du système de protection et évaluation des coûts

Le système considéré pour l'optimisation de la protection est le suivant (figure 1):

a. Confinement suplémentaire T.I.C. (Tight Intermediate Confinement).
b. Système de détritiation de l'eau du circuit primaire (W.D.S.).
c. Système de détritiation de l'air (A.D.S.).
d. Système de séchage (D.S.).
e. Revêtement en acier du mur d'enceinte (S.C.S.).

Figure 1 : Coupe d'un réacteur de fusion selon le projet NET (1985).



Les coûts de ces systèmes sont décomposés en coûts d'investissement et de fonctionnement. Les **coûts d'investissement** comprennent, la construction du T.I.C. et le revêtement du T.I.C. ou du hall S.C.S. (fonctions de la taille du T.I.C. et du

hall), et les systèmes de détritiation W.D.S. et A.D.S. où interviennent les débits.

Les **coûts de fonctionnement** résultent des systèmes d'épuration A.D.S. et D.S. ; qui sont fonction des débits. Viennent aussi les coûts de décontamination, des déchets et des doses travailleurs, coûts qui dépendent de la concentration en tritium de l'air. Le coût des doses délivrées à la population est introduit en partant du principe que les effluents d'air tritiés peuvent contaminer le public. Cette contamination peut se chiffrer par un équivalent de dose collectif. S'il lui est associé un coût de l'homme-Sievert, il en résulte un coût global relatif aux rejets. On considère que 3.7 10^{10} Bq/an (1 Ci/an) correspond à une dose collective de 10^{-4} h.Sv, soit 100 FF.

L'équation de conservation de la masse du tritium, appliquée à l'intérieur de chacune des enceintes de confinement, permet de déterminer les concentrations en tritium. Le système est supposé en équilibre avec des taux de fuite ou de ventilation constants. Un scénario d'accident introduit consiste en un rejet instantané de tritium gazeux HT, dans le hall ou dans le T.I.C. s'il existe. Les parois du hall ou du T.I.C. ne sont pas endommagées par l'accident. On suppose qu'au bout d'un renouvellement d'air, (dû aux diverses ventilations possibles) de l'enceinte contaminée, la moitié seulement du tritium introduit par l'accident sera évacuée. En maintenant une ventilation constante, on peut calculer le nombre de renouvellements (N) nécessaires pour retomber à une concentration, dite de routine, c'est à dire inférieure à un seuil donné estimé à 10 fois la concentration maximale autorisée (CMA). Partant de cette valeur de N, il est possible de déterminer la période de confinement T nécessaire pour que les N renouvellements puissent s'effectuer.

2.3. Traitement informatique

La **réalisation informatique** du calcul du coût de la protection le plus faible est prise en charge dans le **code TRITO** (TRItium Transfer Optimization). Il s'agit d'un code de calcul conversationnel, écrit en APL et exécutable sur micro-ordinateur. **L'intérêt de TRITO** est que le code ne se réduit pas au seul calcul du coût de la protection. Il détermine aussi la valeur des débits de ventilation pour la configuration qui lui a été introduite.

TRITO peut sur option, optimiser un autre paramètre au choix de l'utilisateur, dans une gamme de variation donnée. Il peut ainsi déterminer le volume du hall ou du T.I.C., la source de tritium contaminant le circuit primaire de refroidissement, ou encore les taux de fuite du T.I.C. ou du circuit primaire entrainant les coûts les moins élevés.

3. TYPES DE RESULTATS

Il est possible d'avoir des informations sur le fonctionnement des systèmes d'épuration les plus économiques. Par exemple, **la détritiation de l'eau** de refroidissement du Tokamak **n'est pas** toujours **indépendante de la détritiation de l'air.** Ou encore, à propos de la détritiation de l'air, en situation de routine, on constate que le **surcoût** nécessaire pour assurer la **protection** des travailleurs et du public, **est directement lié** à deux éléments : le **volume du hall** et **la fuite en tritium** dans le hall en provenance du circuit primaire de refroidissement. Dans le

cas où on ajoute un confinement suplémentaire T.I.C. entre le réacteur et le hall, seul compte alors le **volume du T.I.C.**.

En cas accidentel, ne pas réaliser les investissements nécessaires, pour lutter contre un accident, aurait pour conséquence d'augmenter la durée de la détritiation du réacteur et aussi les coûts de fonctionnement pendant la période accidentelle. Si les coûts d'investissement diminuent avec la période de détritiation, il n'en est pas de même des coûts de fonctionnement qui augmentent beaucoup plus vite, d'où il y a nécessité de détritier le plus rapidement possible. Faire face correctement à un rejet accidentel de 10^7 Ci, selon les coûts de protection les plus bas, demande un investissement suplémentaire de 8×10^6 FF/an par rapport à l'investissement nécessaire pour la routine de 4×10^7 FF/an, dans la solution avec T.I.C.. Cet investissement permet de réduire au maximum les coûts de fonctionnement durant une situation accidentelle, soit 10^5 FF/accident. Mais il est tout à fait envisageable de ne pas réaliser d'investissements supplé mentaires, auquel cas la résorption de l'accident durera 4 jours au lieu de 1 et coûtera 3×10^5 FF au lieu de 10^5 FF.

4. CONCLUSION

Une telle étude permet, en laissant au code TRITO, le soin de gérer les limites de dose pour les travailleurs et les limites de rejets dans l'environnement, de faire ressortir les éléments essentiels et les plus coûteux du système de protection. L'optimisation du coût global de la protection résulte d'un équilibre entre d'une part, le coût des détritiateurs d'air et d'eau, et d'autre part le coût des déchets, avec parfois le coût des travailleurs. Elle permet de quantifier les rejets atmosphériques et les concentrations en tritium à l'intérieur du hall. Cette étude met en évidence l'intéret, tant en limitation des doses qu'en coût financier, de surconfiner autour des sources potentielles de rejets. Au fur et à mesure que le "design" du réacteur sera precisé, les confinements intermédiaires seront dessinés autour de chaque point faible de la machine (vannes, raccords, soudures, injecteur de "Pellets", récupération du tritium, etc...). En introduisant dans TRITO, les paramètres des nouveaux systèmes de confinement (volume, ventilation...), on pourra juger de leur intérêt.

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ORGANIZATION AND OPERATION OF A MULTIHOSPITAL DEPOSIT FOR RADIOACTIVE WASTES DERIVING FROM MEDICAL USE

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The increasing use of not sealed radionuclides in medicine for "in vivo" and "in vitro" diagnostics and for therapy, presents the problem of the management of the radioactive wastes in order to not pollute the environment.

In fig. 1 & 2 are shown the amount of radionuclides used for diagnostics during the last 14 years in Bologna, Italy (population 500,000; 5 hospitals; 4700 beds, 2 Nuclear Medicine Dpts, 2 Metabolic radiotherapy dpts; 5 Radioimmonoassay Labs; 2,400,000 determination for year; 9,500,000 US$/year the cost of the radionuclides).

In 1987 more than 6 GBq of ^{125}I and more than 5 TBq of ^{99}Mo has been used while the use of longer lived isotopes (< 9 d) is decreasing.

The strategy which has been adopted in Bologna in order to manage the radioactive wastes is the following:

a) collection of the wastes from all the users of the city;

b) storage in a unique deposit for an appropriate number of half lives;

c) disposal, by means of conventional system of waste treatment, when the amount of radioactivity is below the limits which are defined by national laws.

The limit of disposal into the environment are the following (with reference to the radionuclide mostly used in hospital).

Nuclide	MBq max activity per year	Bq/ml max concentration (liquid wastes)	Bq/g max concentration (solid wastes)
125 I	3.7	0.037	3.7
131 I	37	3.7	3.7
57 Co	37	1850	37
201 Tl	370	1110	37
67 Ga	3.7	0.037	0.37
99 Mo	370	0.74	37

Operation

The single user must dispose the radioactive solid wastes into 25 l drums provided of safety lock. At the moment of locking some data must be written on the cover such as: isotope, activity, weight and date.

Periodically an authorized car collects from the users the drums and brings them to the deposit, a building which lies 30 km far from the city in a poorly inhabited area.

The plan of the building is shown in fig. 3 and it is possible to distinguish three main zones.

A. It is an area with an office, a radiometry laboratory and decontamination facilities.

B. In this area the wastes are received, catalogued and then submitted to a volume reduction procedure which consists in a compression within a 100 l oil drums by means of an idraulic press based system. In this way, about 15 little drums can be pressed into one big drum.

C. In this area are stored all the drums containing radioactive wastes. They are identified by a label and their content is recorded in a data base operating on a personal computer.
Moreover a computer program determines the time of storage inside the deposit of any single drum in order to dispose them when the radioactivity is below the limit already listed.
An other area is obtained inside the previous one where wastes containing scintillating liquid for beta counting are stored.

The main features of the building are the following:

Total area: 650 m^2

Capacity: 1760 100 l drums (useful to store the wastes deriving from the city hospitals for about 5 years without turnover).

Safety precautions:

a) Everything, inside the deposit, is fire-proof. Notwithstanding this, a fire detection system is permanently connected to the fire-brigade station.

b) The deposit has been built using criteria against the risk of inondation.

c) All the liquids used into the deposit for washing are collected into two containers in order to prevent pollution in case of radioactive contamination.

d) Instrumentation for the detection of radioactive contamination on the surfaces and on the operators is present in the deposit as well as a multichannel analyzer connected to a well - type NaI detector for γ spectrometry.

Conclusions:

After one year experience in the management of the deposit it is possible to outline the following:

1) All the city users are set free from the problem of the radioactive wastes deriving from medical use.

2) Not allowed disposal are avoided and all radioactive wastes are under control.

3) No accident has been pointed out.

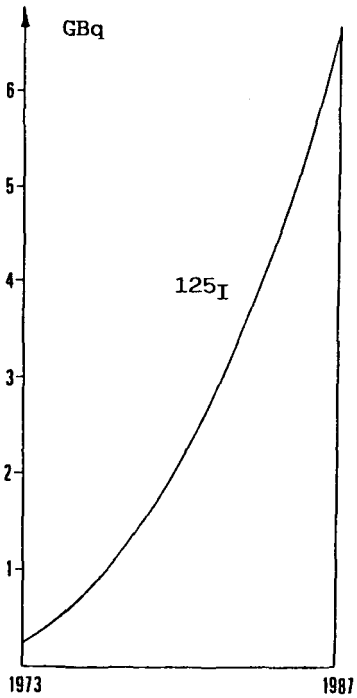

fig. 1

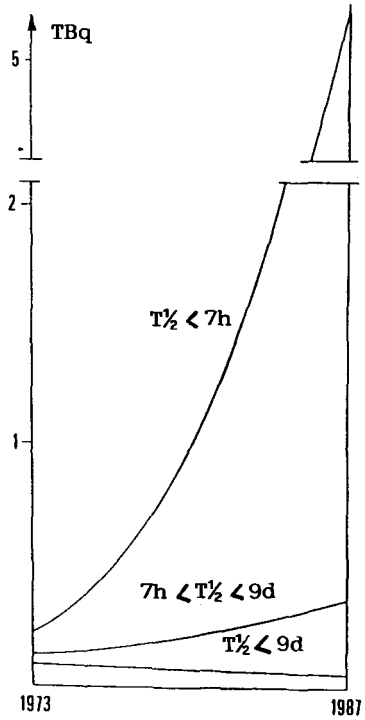

fig. 2

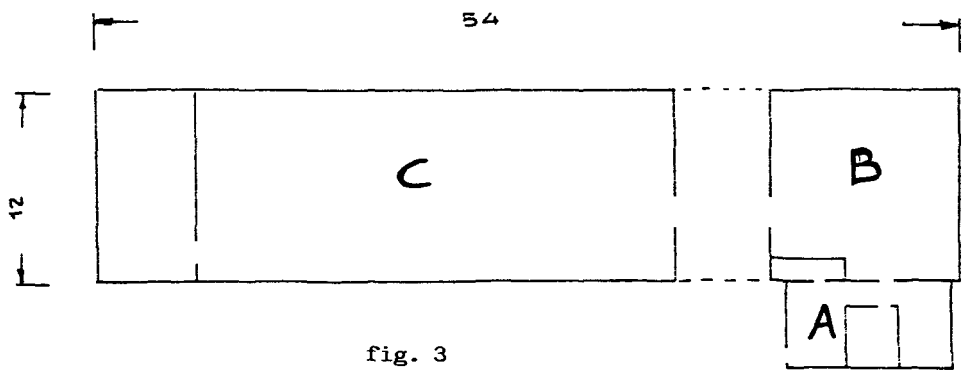

fig. 3

COMPARISON OF U.S. AND INTERNATIONAL STANDARDS FOR RADIATION PROTECTION INSTRUMENTATION

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INTRODUCTION

The quality and performance of radiation protection instruments are extremely important in providing conservative protection to radiation workers. International standards for radiation protection instrumentation, published by the International Electrotechnical Commission (IEC), specify general performance, type, and acceptance test requirements. The recent draft American National Standards Institute (ANSI) standards on radiation protection instrumentation provide more specific performance requirements and definitive performance tests.

The IEC Publication 395, Portable X or Gamma Radiation Exposure Rate Meters and Monitors For Use In Radiological Protection, was published in 1972 and is a recommendation that specifies general characteristics; general test procedures; radiation characteristics; and electrical, mechanical, safety, and environmental characteristics.[1] The recommendation applies to portable instruments intended to measure exposure rate due to x- or gamma radiation of energy between 50 keV and 3 MeV for the purposes of radiation protection.

Draft ANSI N42.17A-D8 (May 1987), "Performance Specification for Health Physics Instrumentation - Portable Instrumentation for Use In Normal Environmental Conditions," is a standard that establishes minimum acceptable performance criteria for health physics instrumentation for use in ionizing radiation fields.[2,3] The standard was written in 1981 by a task group that included manufacturers and users of these instruments as well as representatives from the regulatory bodies. As in IEC Publication 395, draft ANSI N42.17A-D8 specifies general, radiation, electrical, mechanical, safety, and environmental characteristics along with test procedures for each characteristic.

SIMILARITIES AND DIFFERENCES

The purposes of the two standards are essentially the same. However, IEC Publication 395 is specifically for portable x- or gamma radiation detection instruments, whereas draft ANSI N42.17A-D8 includes portable rate and integrating devices for beta, photon, and neutron radiations and monitors for surface contamination (alpha, beta, and photon). Both standards use similar sets of test conditions.

A noticeable difference between the two standards is the classification of instruments. IEC Publication 395 addresses limits of variation of indication for Class I, II, and III instruments. For example, the coefficient of variation for a

Class I instrument is <10%, and for a Class II and Class III instrument is <20%. In draft ANSI N42.17A-D8, a Class A instrument meets all the applicable requirements in the standard, a Class B instrument must meet only those requirements in specific sections, and a Class C instrument meets the requirements specified by the purchaser or user group.

Table 1 compares some of the test characteristics of the two standards and lists some of the requirements found in only one of the standards. The requirements found in IEC Publication 395 and draft ANSI N42.17A-D8 are similar. However, in most cases, the ANSI standard provides more specific guidance on methods for performing tests.

TESTING OF DRAFT ANSI N42.17A

The Pacific Northwest Laboratory (PNL) has evaluated the draft of ANSI N42.17A in terms of applicability and practicality and has also evaluated the performance of the instrumentation used in the study with respect to the requirements in the proposed standard.[2,3] Selected data are presented in Table 2. The testing of instruments against the draft ANSI N42.17A standard took place over a 2.5-year period and included procedure development, verification and instrument testing on more than 100 instruments. Five groups of instruments were tested including ionization chambers, Geiger-Mueller (GM) detectors, alpha detectors, neutron monitors, and others.

From the evaluation of the standard, a number of recommendations were made to the ANSI N42.17 working group regarding the requirements of the draft standard. Some major changes that were based on the testing program are: change in the coefficient of variation test requirement; change in the photon-radiation energy dependence test requirement; inclusion of an equilibration time for temperature, humidity, and ambient temperature tests; increasing the intensity for the magnetic field test; and a decrease in the acceleration level applied to instruments in the vibration test.

CONCLUSIONS

Standards such as the two discussed above are extremely valuable in assuring that quality and performance of radiation protection instruments are adequate to protect the health and safety of workers using the equipment. Evaluating the standards by testing instruments against the requirements in the standards can be beneficial to ensure that the provisions of the standard are applicable and can be applied to the appropriate instrumentation. As radiation protection instruments change and improve with advances in technology, the standards must be updated to reflect these changes.

ACKNOWLEDGMENTS

This work was performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

TABLE 1. Comparison Of Standards Requirements

Test Characteristics	Draft ANSI N42.17A-D8	IEC 395
Alarm Threshold	Alarm threshold shall be given as % of scale or decade of adjustment	Adjustment should be stated
Stability	±6% of reference for: 3 hrs: battery 24 hrs: A.C. power	Not addressed
Geotropism	Within ±6% of reference	±10% of reference orientation
Accuracy	±15%	±10% (C1) ±20% (C2) ±40% (C3)
Photon Energy Dependence	±20% from 80 keV to 1.25 MeV	±25% from 50 keV to 3 MeV, ±15% from 300 keV to 3 MeV (C1)
Photon Radiation Overload	Instrument shall continue to operate and be offscale	Remain full range if exposure over range
Extracameral Response	Not greater than 5%	Not addressed
Interfering Ionizing Radiation Response	Not greater than stated by manufacturer	Shall be designed to limit influence of other ionizing radiations
Temperature Dependence	Reference 22°C: ±15%: 0-40°C ±20%: -10-50°C	Not addressed
Vibration	±15% following harmonic loading of 2 G/15 min w/freq. 10-33 Hz	Not addressed
Ambient Pressure	±15%; 70-106 kPa reference 101 kPa	Effects of variation should be indicated
Mechanical Shock	±15% following 50 G 18 msec sinusoid shock on 3 axes.	Withstand shock of 30 G which is 18 msec sinusoid, any direction

TABLE 2. Selected Test Results

Test	Instrument Type	Number of/Number Failures/ Tested
Stability	Ion Chamber	1/9
	GM	1/35
Geotropism	Ion Chamber	0/10
	GM	0/24
Accuracy	Ion Chamber	1/9
	GM	7/19
Photon Radiation Energy Dependence	Ion Chamber	3/9
	GM	14/20
Temperature	Ion Chamber	9/17 (0° to 40°C)
	GM	0/23 (0° to 40°C)
Vibration	Ion Chamber	1/4
	GM	0/7
Ambient Pressure	Ion Chamber	0/10
	GM	0/5
Mechanical Shock	Ion Chamber	0/3
	GM	0/8

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A MODIFIED INDEX FOR MONITORING OF RADIATION PROTECTION PROGRAMMES WITH SPECIAL REFERENCE TO NUCLEAR MEDICINE INSTITUTE.

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In India, Institute of Nuclear Medicine & Allied Sciences (INMAS) was established with the main objective of developing radioisotopic methodologies for diagnosis, therapy, biomedical research and training. The steady progress in India's Atomic Energy Programme gave impetus to rapid growth in the application of nuclear medicine techniques at the Institute and now it is one of the largest users of unsealed radiation sources in the country. Right from the beginning adequate attention has been devoted to the radiation safety of the staff and personnel management techniques have been successfully employed to assess performance of the radiation protection staff. A significant positive correlation ($p < 0.02$) between performance and job satisfaction of radiation protection personnel of the Institute has been demonstrated (Gupta 1976). Efforts are also directed to evolve a parameter that may throw light on the effectiveness of radiation protection staff and subsequently that may be helpful to monitor radiation protection programme of any other establishment also.

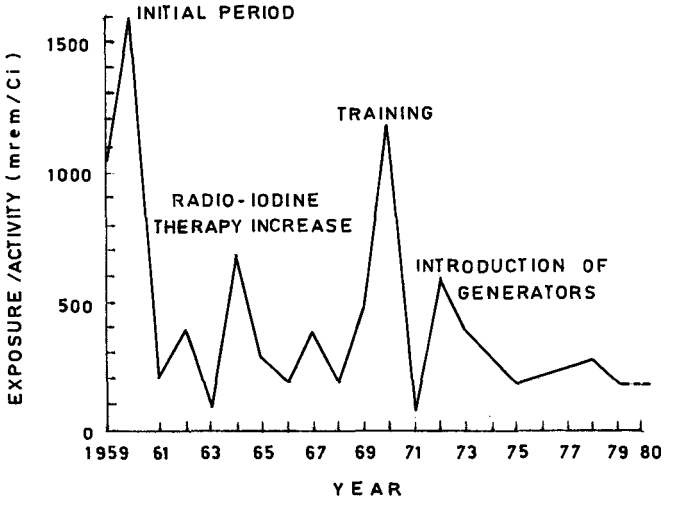

Fig. 1. Normalised annual collective dose equivalent to radiation workers at INMAS.

The rapid development at the Institute has also resulted in increasing radiation exposure to the staff that is usually demonstrated by Annual Collective Dose Equivalent (ACDE) of the workers. However the Annual Collective Dose Equivalent reflects the situation rather inadequately. It does not take into account the quantitative and qualitative changes that had occured in the utilisation of radioisotopes at the Institute nor it takes cognizance of change in staff pattern during the last two and a half decades (Gupta et al 1984). Thus unless the level of radioactivity handled by radiation workers of the Institute which had caused a particular level of ACDE, is also taken into consideration, ACDE's themselves maynot reflect the real health and full efficacy of the

functioning of radiation protection system. For that purpose, annual dose equivalent of each radiation worker of the Institute for a particular year is added and the sum is divided by the total number of radiation worker in that year. This process gives the average annual dose equivalent for that year. Now, if this average is divided by the total quantity of radioactivity (in curies) handed at the Institute in that year, we obtain a Normalised Annual Collective Equivalent (NACDE).

This modified index was applied to investigate radiation protection programme of the Institute for the period from 1959-80. The figure 1 shows chronological variation of NACDE (m rem/Ci) for all the radiation workers of the Institute. Wirh a minor variation, the curve has four distinct maxima. While each maximum in the NACDE value represents modification of the then existing procedure or addition of a new function at INMAS, its subsequent lowered values demonstrate the existance of a vigil and dynamic radiation protection organisation at the Institute.

The same modified index modified index is also applied to identify the group of radiation workers subjected to maximum levels radiation exposure. For the sake of discussion, the NACDE values for the workers in Health Physics Division, Medical Division and Radiation Chemistry Division are taken. The analysis shows that NACDE values in the initial period of INMAS formation, is highest for workers of Health Physics Division. Afterwards, due to extensive use of generator produced short lived radionuclides, the workers in Radiation Chemistry Division got the maximum exposure.

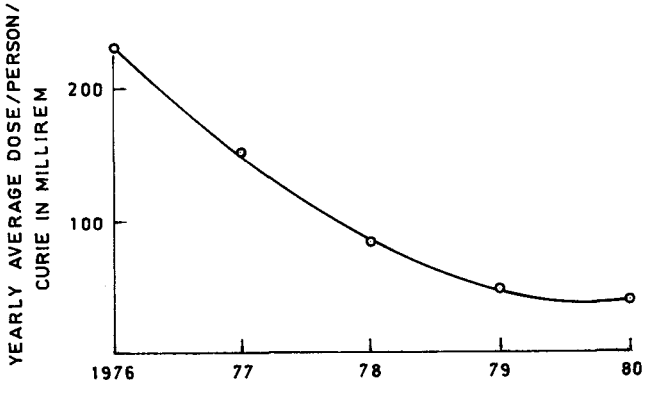

Fig. 2: Annual average dose equivalent per curie of radioactivity handed to the workers in the Radiation Chemistry Division (1976-1980).

The effectiveness of radiation protection is further evidenced by figure 2. In the figure, the annual average dose equivalent per curie handled by the workers of radiation chemistry group (NACDE for that group) is shown for 5 years (1976-80). Although the activity handled during this period represents an increase of more than 25%, the NACDE values have been progressively decreasing.

The usual annual collective dose is a good indicator of radiation protection situations of an Institution, however, the modified index (NACDE) gives more information about the establishment as well as about effectiveness and efficacy of its radiation protection set up. In this typical case it has

also given a reassuring picture of the exposure situations of the most vulnerable group of the Institute.

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HEALTH SURVEILLANCE OF RADIOLOGICAL WORK

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1. Introduction

Shielding X-ray devices and issuing filmbadges to radiological workers in 1936 can be considered the start of the radiological protection in the Philips enterprises in the Netherlands. Shielding and equipment were constantly improved based upon the dosimetry results of the filmbadges. The problem of radioactive waste led to the foundation of a central Philips Committee for radiological protection in 1956, which in 1960 also issued an internal license system in order to regulate the proper precautions to be taken: workplace design and lay-out, technological provisions and working procedures. An evaluation of all radiological work in 1971 learnt that a stricter health surveillance program was needed to follow up the precautions issued by the licence. On one hand a health surveillance program was established and on the other hand all types of radiological work were classified. In this way an obligatory and optimal health surveillance program was issued for each type of radiological work.

2. Health Surveillance Program (HSP)

In 1972 the following 7 items were defined:

1. Periodical inspection and radiation measurement of switchable sources;
2. Periodical inspection, radiation measurement and smeartest of radionuclide sources;
3. Personal dosimetry with thermoluminescent dosimetry badges or filmbadges (neutrons, e.g.)
   Environmental monitoring was dictated in the licence;
4. Preemployment and exit medical examination;
5. Periodical medical examination;
6. Periodical bio-assay analysis, mostly urine samples;
7. Periodical total or partial body counting.

The periodicity was set for a minimum of once a year. The highest frequency was applied for the personal dosimetry: every month.

3. Classification of types of Radiological work

Based upon the experience sofar the following classification of radiological work and the according appropriate health surveillance program (HSP) items were made in 1972:

- work in one place with a radiation emitting device with a small chance of external radiation (less than 5 mSv per year: HSP 1 (per. inspection), HSP 3 (pers. dosimetry) and HSP 4 (preempl. & exit examination).
- work in one place with a radiation emitting device with a higher chance of external radiation (5-50mSv per year): HSP 1 (per. inspection), HSP 3 (pers. dosimetry), HSP 4 (preempl. & exit examination) and HSP 5 (per. examination).

- work in different places with radiation emitting devices (mostly service and maintenance): HSP 3 (pers. dosimetry), HSP 4 (preempl. & exit examination) and HSP 5 (per. examination).
- work in one place with a well shielded device emitting parasitic X-rays: HSP 1 (per. inspection), HSP 3 (pers. dosimetry) and HSP 4 (preempl. & exit examination).
- work in one place with a well shielded radionuclide source with no emission of any importance: HSP 2 (per. inspection, smeartest).
- work in one place with a shielded radionuclide source emitting gamma radiation with a small chance of external radiation (less than 5 mSv per year): HSP 2 (per. inspection, smeartest), HSP 3 (pers. dosimetry), HSP 4 (preempl. & exit examination).
- work in one place with a shielded radionuclide source emitting gamma radiation with a higher chance of external radiation (5-50 mSv per year): HSP 2 (per. inspection, smeartest), HSP 3 (pers. dosimetry), HSP 4 (preempl. & exit examination) and HSP 5 (per. examination).
- work in different places with shielded radionuclide sources: HSP 3 (pers. dosimetry), HSP 4 (preempl. & exit examination) and HSP 5 (per. examination).
- work in one place with radionuclides not emitting gamma radiation (radionuclide laboratory): HSP 2 (per. inspection, smeartest), HSP 4 (preempl. & exit examination), HSP 5 (per. examination), HSP 6 (bio-assay analysis).
- work in one place with radionuclides emitting gamma radiation (radionuclide laboratory): HSP 2 (per. inspection, smeartest), HSP 3 (pers. dosimetry), HSP 4 (preempl. & exit examination), HSP 5 (per. examination), HSP 6 (bio-assay analysis) and HSP 7 (per. body counting).

In this classification in 1972 the next considerations were taken into account.
The standing order that every incident should be immediately reported proved to be insufficient. Especially unexpected periodical checks on internal contamination are important when working with radionuclides. Periodical checks of working conditions and achieved doses are necessary. In case of an unexpected (over)exposure one ought to have information about the preliminary health condition. Persons with an existing health risk should be withheld from additional risk due to radiological occupational exposure. Because of the longterm effects of radiological exposure a good record keeping of all occupational exposures is necessary.

4. Evaluations

In the above described way it turned out to be possible to improve the safety of the radiological work. In 1982 and 1987 (and in between to cope with new legal regulations) we studied 4 main items:
- dosimetry results
- bio-assay results
- body counting results
- medical examination results

A. Dosimetry results

An exposure of neutron radiation was never recorded on a film-badge. The thermoluminescent dosimeters (TLD) recorded the exposure to Xrays and gammaradiation. According to the readings we found from 1972 to 1978 in 12 cases an exposure, exceeding more than 50 mSv. After 1976 exposures between 15 and 50 mSv did not occur anymore. After 1983 exposures between 5 and 15 mSv did not occur anymore. Since 1984 all by TLD measured exposures were less than 5 mSv per year. The population of radiological workers had gradually increased from 996 in 1972 to 1642 in 1986.

B. Bio-assay results

From periodical urine tests a number of unknown internal contaminations were registered. 82 out of 523 on tritium (^{3}H) tested urine samples turned out to be positive. The permissible limit of 20 µCi/l was exceeded in three cases. Highest value appeared to be 22 µCi/l. Also urine tests were carried out with regard to ^{14}C (and ^{32}P). One ^{14}C sample appeared to be positive (0,2% of the permissible concentration).

C. Body counting results

The periodical body counting revealed a number of unknown internal contaminations. Fortunately only small quantities of radiounuclides were found. The highest contamination turned out to be one with ^{65}Zn reaching 5% of the max. permissible body burden. Of course the positive findings of the body counting led to improvement of the working conditions and procedures.

D. Medical examination results

The Dutch Philips enterprises have a well established occupational health service. Health impairment of the workers can lead to readjustment of their work if necessary. So it could happen that a radiological worker was transferred to another job due to his state of health, which was not related to radiological exposures. Only once an applicant for radiological work was rejected in the preemployment examination because of a serious Morbus Bechterew. Abnormal blood-counts (leucocytes, thrombocytes) led to a withdrawel of a number of applicants, while others were admitted with no hematological problems during the follow-up. In the periodical medical examinations 6 radiological workers were rejected, 5 because of pregnancy and one because of renal failure, who retired soon afterwards due to his invalidity. In accordance with these results and new european legal regulations, our health surveillance program was adjusted with effect from 1984:

- no medical examination at all if the estimated chance of exposure is below 5 mSv a year;
- only a preemployment and exit medical examination if the estimated chance of exposure is in the range of 5-15 mSv;
- a preemployment, periodical and exit medical examination if the estimated chance of exposure is in the range of 15-50 mSv or radionuclides are handled.

5. A new development

Licenced radiological work and a health surveillance program have provided a good protection. As a side product irregularities and (small) incidents were detected, which of course asked for immediate action. This combined with the growing awareness about the risks of ionizing radiation revealed a weak spot in our system: an incident/accident checklist was necessary next to a rapid access to relevant data concerning the radionuclides present in the enterprise.
The checklist is dealing with the description of the accident or incident with relevant data about what has happened, the number of possible victims, the present radionuclide or radiation source, the estimated amount of radiation and present health symptoms, followed by actions to be taken. The radionuclide database gives relevant fysical and radiobiological data of the present radionuclide like: radiotoxicity, energies, halflifes, ALI, DAC followed by limits of action, necessary medical examination programs and possible therapy.

6. Conclusions

Classification of types of radiological work and an appropriate health surveillance program for them has proven to be effective in view of the substantial drop in exposures. The contribution of medical examinations was less than expected and they could therefore be reduced. On the other hand the risk management (however small the risk might be) of working with radionuclides needs improvement by using a direct on the spot available checklist and data base.

USE OF COST BENEFIT ANALYSIS IN THE FIELD OF INDUSTRIAL RADIOGRAPHY

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INTRODUCTION

Over the past decade NRPB has had a programme of work on the development of cost benefit analysis (CBA) techniques in the optimisation of radiological protection. A provisional framework for including suggestions for assigning a value to unit collective dose was published for consultation in 1981/82 and after various interim statements this process culminated in formal advice in 1986[1]. As part of this work, and as part of a project for the Commission of the European Communities (CEC) the NRPB has carried out a number of case studies to demonstrate the practical implementation of ALARA or optimization of protection using CBA. These techniques, used in conjunction with ALARA Audits, covered in an associated paper[2], are now in general use in the NRPB's Radiation Protection Adviser Service. They have been used for a variety of medical and industrial situations, but mainly in industrial radiography as this is the part of the non-nuclear sector where occupational exposure problems predominate. The three cases below are given as representative examples.

CASE 1 : DESIGN OF AN X-RADIOGRAPHY FACILITY

This was an 'a posteriori' study[3] of how optimisation considerations might have provided an input to the design of an existing X-radiography facility (see Fig. 1).

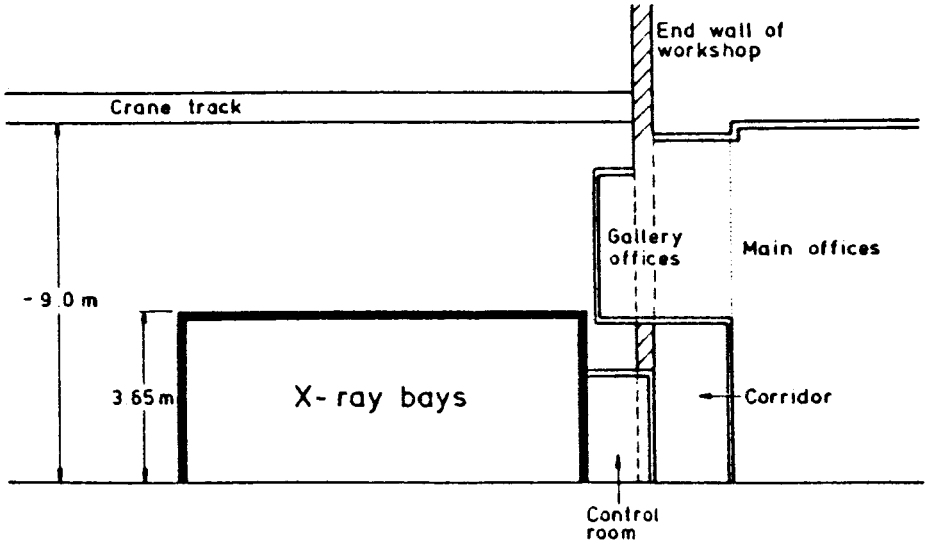

Figure 1 Elevation view of radiography facility

The facility consisted of twin adjacent bays situated within a factory workshop, designed to each accommodate a 200 kVp, 5 mA directional X-ray set used to radiograph a variety of pipework. The useful beam could be pointed in all directions at or below the horizontal. The facility was built from lead sheets affixed to a steel framework. The walls were 7 mm thick lead, designed to ensure that under the worst possible condition of operation the external dose rate would not exceed 7.5 $\mu Sv.h^{-1}$. There were sliding roofs to permit the cranage of workpieces and to protect the overlooking offices during use. The roofs incorporated 4 mm thick lead.

It was decided to limit the problem to optimising the wall thicknesses, using steps of 0.5 mm of lead, and to analyse the protection options of no roof, a shielded roof and a shield on the front face of the gallery offices. Although judgments on the allocation of overhead costs and maintenance costs had to be made, it was relatively easy to identify the protection costs for each protection option. These were in 1980 prices and to cover temporal distributions the costs were annualised over an assumed 20 year useful life using a 5% discount figure. To cost the detriment, the NRPB's provisional costing framework was used and required assessments of the individual and collective doses arising for each protection option. This had previously not been assessed and required modelling of the work pattern, associated dose rate distribution and occupancy distribution, taking into account both transmitted and airscatter radiation. As well as a realistic model reflecting the best estimates of dose, a 'design capacity' model to reflect future changes in work patterns was produced and used.

A cost benefit analysis was carried out using both total cost and incremental cost methods. The technical solution, or 'ALARA solution' was that for the realistic and design capacity models, the optimum wall thicknesses were 2.25 and 3.0 mm lead respectively. Also, for both models the optimum solution for the roof was to dispense with the shielded roof, but to provide a shield to the front of the offices. The sensitivity of these conclusions to variations in a wide range of parameters was tested and the results were found to be robust with variations not exceeding 0.5 mm lead thickness in the walls.

The study used the structured approach of the ALARA procedure developed by Webb et al[4]. In this a clear distinction is made between the ALARA solution and the final decision. It requires identification of factors which can be quantified and included in the analysis and other factors that are relevant to the decision maker but which cannot be quantitatively included. In this case the latter included regulatory factors, reaction of the workforce and accident scenarios. Where there is a quantitative element to factors not directly included in the analysis it is useful for these data to be made available to the decision maker. In this instance for each protection option the following data were supplied: capital cost penalty for shielding to greater than the optimum, maximum individual dose, the maximum instantaneous and time averaged (over 8 hours) dose rates. The latter would be relevant to avoiding the regulatory need for establishing a controlled area outside the X-ray rooms.

CASE 2 : LOCAL SHIELDING FOR GAMMA RADIOGRAPHY

The second case was an 'a priori' study that arose out of a protection problem identified by an ALARA Audit[2]. Within an open top radiography facility inside a workshop, the Company carried out gamma radiography of steel pipework using an Ir-192 source of 185 GBq maximum activity. It was always used without collimation, i.e. panoramically, either inside a pipe for circumferential welds or external to pipes in double wall single image (DWSI) methods. Assessment of the doses involved indicated annual collective doses of 33 and 11.4 man.mSv associated with the DWSI and circumferential methods, respectively. Local shielding in the form of a directional collimator (for DWSI) and a 360° collimator and/or a shielding 'saddle' (circumferential) were considered as protection options. The collimators were commercially available but the shielding saddle would have to be purpose-built and would require to be craned into position. Experience suggested that the collimators, particularly for the DWSI work would be cost effective, but the option of the saddle was not so clear. Therefore it was decided to carry out a simple cost benefit analysis (shown in Table 1) to act as an input to the decisions. A useful life of 10 years was assumed for each option and 0% discount was used.

Date and Element of Analysis	Protection Step			
	DWSI	Circumferential		
	Base Case to Collimator	Base Case to Collimator	Base Case to Saddle	Collimator to Collimator and Saddle
Incremental Cost (£)	500	500	1000	1000
10y dose savings (man mSv)	287	76	86	29
Incremental Cost Effectiveness (£/man Sv)	1.7k	6.6k	11.6k	35k

The annual individual doses were initially of the order of 1 mSv and using the most recent NRPB advice[1] a figure of £15,000 per man Sv was considered appropriate. This analysis therefore confirmed that the collimators were cost effective. The use of the saddle (after implementing the collimator option) would not appear to be cost effective, but possible variations in parameters such as doubling the useful life would make the option more desirable. Also there were other inputs to the decision which were not quantified, e.g. additional production costs due to cranage and advantages from its use in another area of work, open shop radiography. Overall the decision maker decided to implement the saddle option.

CASE 3 : TORCH TYPE CONTAINERS

For many years 'torch' type containers have been used in industrial radiography. These require manipulation of typically 0.4 TBq Ir-192 on the end of a shielded 30 cm 'torch'. They require proximity of the operator and in many cases have been gradually replaced with remote exposure containers, the use of which gives rise to significantly lower doses. The introduction of new Regulations placed more emphasis on ALARA and caused two Companies to consider whether or not it would be reasonably practicable to replace their existing torch containers with remote exposure containers. Table 2 provides a summary of the results from a cost-benefit analysis of these two situations.

Table 2

	No. of workers	Individual dose range (mSv)	Annual Collective dose (manSv)	10 y dose saving (manSv)	Protection cost (£)	Cost Effectiveness (£/manSv)
A	4	0.4 - 2.7	0.007	0.07	7.0k	100k
B	12	5.0 - 18	0.13	1.3	38k	30k

The NRPB advice[1] recommended a general figure of £15k/man.Sv for occupational exposure to cover the objective health detriment and individual risk aversion. However, it specifically recommended an additional multiplier where individual doses were a significant percentage of the dose limit. For Company A the individual doses were small (due to a low workload) and thus £15k/man.Sv was the appropriate figure to compare with the incremental cost effectiveness. For Company B the doses are a significant percentage of the dose limit and indeed would have been above the ALARA Investigation Level of 15 mSv. At this dose level an extra multiplier of 5 was considered appropriate giving £75k/man.Sv. It was concluded that for Company A it was not reasonably practicable for them to purchase remote exposure containers until the end of the useful life of the torch containers. However, for Company B it was concluded that in order to achieve ALARA, investment in the new containers should be made.

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DOSIMETRY INDEX: A USEFUL CONCEPT IN OPERATIONAL RADIATION PROTECTION

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INTRODUCTION

The aim of dose assessment, including dosimetry measurements in operational radiation protection is to provide information which is needed:
- to demonstrate that individual dose equivalents are in compliance both with the ALARA approach as well as with dose limits
- to signalize and elucidate trends in exposure levels in different working conditions and radiological practices
- to organize and manage effective and efficient routine and operational monitoring programs.

In conditions of uniform external exposure to penetrating radiation, the dose assessments in terms of dose equivalent to total body will be adequate since dose limitation for total body is determant.

However, in working conditions that cause a substantial non-uniformity of exposure for different organs or tissues, it may be necessary to evaluate organ dose equivalents in comparison with respective organ dose equivalent limits.
To facilitate these evaluations we use the concept of «dosimetry index».

The quarterly dosimetry index has proven to be a useful concept to define and applicate a coherent system of classification of working conditions. Also the management of person related arrangements for radiation protection, for dosimetry and for medical supervision can effectively be based on individual index values and frequency distributions of indexes within groups of persons with the same working conditions.
In our practice, dosimetry index is also used in dose registration.
In this paper the concept and practical application of dosimetry index will be discussed.

DOSIMETRY INDEX

The operational quantity «dosimetry index» is defined as the ratio (per cent) between the value H(t) of the individual dose equivalent in a periode of time (t) and the pro rata fraction (f) of the relevant annual dose equivalent limit (H_{lim}), corresponding to that periode of time.

$$\text{Dosimetry Index} = \frac{H(t)}{f \cdot H_{lim}} \cdot 100$$

This definition fulfills the conceptual requirement that an index value shall be dimensionless.

UNIFORM EXPOSURE

In working conditions which are characterized by uniform external exposure to penetrating radiation, assessment of dosimetry index per monitoring period is based on the dose equivalent limit for total body. The values for individual dose equivalent are derived from

- Ambient dose measurements. This approach is considered satisfactory as long as dosimetry index for the monitoring period is not above 3.
- Individual dosimetry with one basic dosemeter worn at the trunk of the body. These dosemeters are calibrated to Individual Dose Equivalent as recommended by ICRU [1].

  Only in cases where dosimetry index derived so far, are significantly in excess of 30, a better assessment is made in terms effective dose equivalent using conversion functions [1,2].

NON-UNIFORM EXPOSURE

Non-uniformity of exposure for different organs, tissues or body parts can be the result of a variety of causes:

- Mixed radiation fields, different radiation quality will result in highly different pattern of energy absorption. Non-penetrating radiation effect mainly superficial tissue such as skin and eye lens. Depth dose distribution for neutrons is highly dependent on neutron energy.
- Partial external irradiation due to narrow beam configuration or due to shielding or protective clothing.
- Combination of external exposure and internal contamination, especially when internal contamination is from radionuclides that are highly concentrated in specific organs or tissue.

Dose assessment for non-uniform conditions are complicated by the fact that dose equivalents for superficial organs are non-additive and are not included in the concept of effective dose equivalent. In addition dose limits for internal organs, skin, eye lens and extremities differ from dose equivalent limits for total body and effective dose equivalent. Dosimetry index is a helpful tool in the evaluation of dosimetry measurements (or calculations) by applying the concept to each of the non-additive components.
When none of these respective index values for the monitoring period exceeds 10, no further dose evaluation or more precise assessment is made. Doses are recorded as measured. In working conditions where at least one of the non-additive dosimetry indexes in a monitoring period exceeds the value of 10, individual dosimetry will be focussed on measuring the dose equivalent (or related quantity) for related organ or tissue. In many practical situations this means that in addition to the basic individual dosemeter, other dosemeters are worn to measure exposure of skin, eye lens or extremities. In case of internal exposure, monitoring programs are organized in such a way that, where possible, intakes can be assessed in terms of committed effective dose equivalent.

In general, the highest index value is determinative for the operational radiation protection approach.

REFERENCE LEVELS

Reference levels for record keeping, investigation and intervention are related to dosimetry index. Dosimetry measurement results, corresponding to index values below 10 for the monitoring period are recorded as such, but no special attention is paid to the accuracy or validity of each result. Further investigation by radiation protection staff will only take place in those cases, where zero measurements were to be expected.
Dosimetry measurements below threshold of detection are treated as zero for dose assessment purposes, but are recorded in such a way that they can be distinguished from non-measurements.

Investigations levels and intervention levels are layed down in a monitoring program in relation to both dosimetry index per monitoring period as well as dosimetry index per quarter. Investigation levels are by no means used as derived limit. Where excess was not planned or expected it is a trigger level for radiation protection staff to evaluate the practice, operating procedures and working conditions.
Where excess was expected, investigation is aimed at confirmation that working conditions are consistent with planned practice.
Since exposure levels in normal conditions are quite different for working condition A and working condition B, numerical values for investigation levels are different also (see table 1).
Where investigation levels on a quarterly base are exceeded for working condition A, attempts are made to assess actual effective dose equivalent and/or organ dose equivalent more accurate.

TABLE 1 REFERENCE LEVELS EXPRESSED IN TERMS OF DOSIMETRY INDEX

		Monitoring period	Quarter
Investigation	A	100	30
	B	30	10
Intervention	A	300	100
	B	100	30

CLASSIFICATION

In international recommendations and national legislation reference levels for classification of working conditions or persons are related to criteria, whether individual occupational exposure during a year «might exceed» or «is unlikely to exceed» specified fraction of respective annual dose equivalent limits. For several reasons, evaluation of actual or projected dose equivalents is better be done over shorter time periods:
- In many practices (e.g. research and education) persons are not continuously involved in radiological work. Dose evaluation on a yearly base can lead to underestimation of the potential exposure.
- In general, doses are not received at constant rate (e.g. hot jobs, maintenance).
- Radiation protection management must provide necessary dose-allowance for the rest of the year to prevent conflict with ALARA-concept or annual dose limits.

DOSE DISTRIBUTION

In literature, information about dose distributions in occupational exposure is generally related to dose equivalent for total body [3]. Use of dosimetry index on quarterly base provides a mean to present distributions of potential or actual exposure for non-uniform exposure conditions. Examples for five types of

practice are given in fig. 1. In all case the highest value of non-additive indexes is determinative.
For example: in cardiac catheterization we have learned that dose equivalent to the eye lens is far out the higher in proportion to annual dose limits [4]. For comparison, collected data about medical X-ray diagnostics are presented. The same type of distribution is given for radiological research in our university. Separately the distribution is given for working conditions A in cyclotron with radionuclide production.
N.B. The number of observations for each distribution is given, expressed in number of persons•quarter.

Fig. 1 DOSE DISTRIBUTION FOR DIFFERENT WORKING CONDITIONS

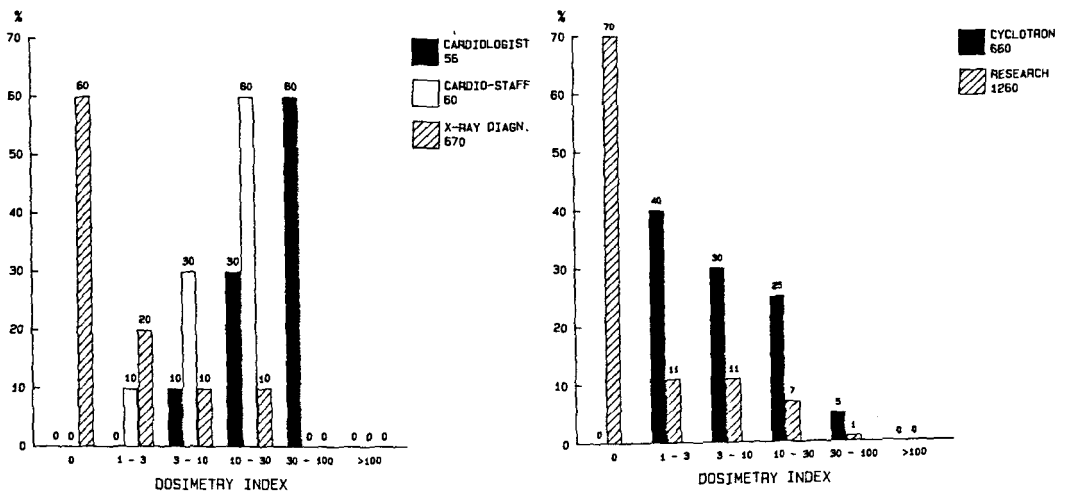
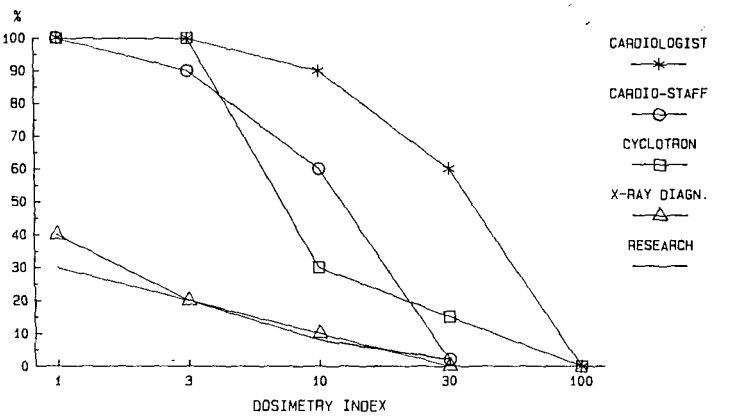


Fig. 2 PROPORTION OF WORKERS EXCEEDING GIVEN DOSE LEVEL PER QUARTER



A useful characteristic of a dose distribution is the proportion of persons exceeding a given dose level. For the same practices this information is displayed in fig. 2. This type of information is extremely useful for planning purposes in radiation protection management.

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QUALITATIVE AND QUANTITATIVE DECISION AIDING TECHNIQUES APPLICABLE IN RADIATION PROTECTION

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1 - RADIATION CONTROL PRINCIPLES

EDF's exposure reduction program comprises a three pronged attack on the radiation field buildup process, based on modification of PWR's design features, and operating conditions.

1.1. Reducting the sources of radioactive fields

More than 75 % of the occupational radiation exposure is due to deposite activity on the out-of-core surfaces. So it is very important to minimize these deposites.

There are three ways in witch such contamination can be reduced :

- use of alternative materials with less corrosion product, input a lower activity potential (e.g. to eliminate Cobalt 59).
- primary chemistry control
- optimizing conditions under witch large-scale corrosion product transfert and migration occur during cold shutdown, due to changes in coolant physical and chemical characteristics, as fast flow clean up of primary coolant during cold shutdown.

1.2. Reducing equivalent dose rates

Biological protection refers to measures aimed to protect workers against ionizing radiation, i.e. use a protective shields, and location of work areas at reasonable distances from radiation sources.

Such action must necessarily consider both the radiation level prevailing in the vicinity of plant systems and equipment, and the time required for workers to carry out relevant operating procedures or maintenance activities.

It is thus necessary, at the very first stage in plant design, to provide for :

- rational equipment lay out, i.e. in large members of differents rooms, whereby high and low activity systems are separated ;

- Installation of permanent biological shields, as required by plant lay out

- use of remote valve control devices

- limiting neutron leakage by confinement of neutron particle emission to the reactor pit.

Other actions are implemented during plant operation. Work planning studies are performed on a case-by-case basis, to identify requirements for any additional protection devices or remote handling tools, needed to restrict work activities to low exposure.

1.3. Reduction of exposure times

In nuclear power plants, such reduction are a vital part of radiation control strategy. Current programms call for the tiered action aimed at :

- reducing stay time
- limiting the number of workers subjected to exposure.

Reduction is achieved firstly, through plant construction features, geared to :

- easier equipment acces,

- use of sophisticated handling equipment (wich entails appropriate design of relevant plant room)

- easy equipment dismantling

- improved reliability, i.e. less frequent maintenance operations, according to the regulatory guides.

These features are supplemented by a program of permanent actions aimed at reducing radiation worker stay times, through :

- better training, wich may, in certain cases (very high radiation levels, complex operations...) include use of non radioactive mock-up equipment, as an adjunct to traditional course material.

- better work planning, based on experience acquired in the same or other similar plants, through generation and use of detailed plant histories.

1.4. In-plant radiation protection

Controlled access area

All areas of the plants in with average radiation level is likely to exceed 0,25 mrem/h, and those involving constant risk of surface contamination, or

airborne radioactivity, are subject to access control restrictions

- green area : radiation level up to 0,75 mrem/h and lower than 2,5 mrem/h
- yellow area : 2,5 to 200 mrem/h
- orange area : 200 mrem/h to 10 rem/h
- red area : more than 10 rem/h

Safety of contractor workers

French law clearly defines the responsabilities between the plant manager and the contractors.

The plant manager must provide the contractor with any information required to assume his assigned responsabilities.

The contractor is responsible for the safety of his employees, and for any hazards generated by their activity.

He is responsible also for the individual medical surveillance and the individual dosimetry.

Personnel dosimetry program

The personnel dosimetry program involves two separate but interelated means of surveillance

- electronic dosimeter and computer processing
- statutory individual photographic film badge.

2 - USE OF FEEDBACK

2.1. - Main parameters influencing the collective dose

It is clear that the main parameters influencing the collective dose are, for a plant wich is running (except the work organisation) :

- the deposite activity on the out of core surfaces

In french NPP, primary coolant treatment is called "with coordonate lithium". It optimize both, the mass of corrosion products deposited, and the location where they are deposited.

- the way to decrease the temperature of the primary coolant

The french procedure uses an anticipated oxygenation, as early primary coolant temperature is about 120°C, without oxygene peroxyde, but only with an injection of air.

In this way, the advantage of the control of corrosion products, during the cycle is preserved, according to the respect of the planning of work.

2.2. Qualitative and quantitative decision aiding techniques

Activity index

About 12 hours after the reactor has been stopped, the radiological protection service performs a serie of measurements of dose rates, at the contact of the primary pipes, and determines an "activity index", in mrem/h.

It is obvious it exists a correlation between the value and the collective dose will be taken during the shutdown. More the activity index is high, higher will be the collective dose, and more the maintenance teams will must be carefull.

It will perhaps be interesting to perform complementary measurements, to know where are the highest dose rates. Thus, maintenance team will be aware and will can protect itself with shield, organizes its worksites to save more time and personnel to decrease the collective dose.

Knowing the "potentiel collective dose", the maintenance team can give itself a target.

Also, it will be interesting, for the future cycles, to know why, for example, the activity index is higher than the average value of all the NPP (anormal deposite, bad primary coolant treatment, failure on purification systems, etc...)

- Analyse of dosimetry during outage

For all the maintenance works during a shutdown, the collective dose is known. Everybody entering in nuclear island bears an electronic dosimeter, and it exists a computerized counting of the doses.

Works are divided in groups :

- reactor
- steam generator
- primary circuit
- auxiliary circuits RCCS, ...
- in service inspection
- electric works
- cleaning, scaffolding

At the Radiation Protection Department, all the values are collected, and the most probable value is evaluated, according to the experience.

Each power plant must be analyse its results and compares at these values.

All the differences will be explained, and it is probably a big progress factor.

To perform this job, we have centralised all the data concerning :

- collective doses (job related)
- dose rates
- deposite activity

on a date base called TIGRE RP.

3 - RESULTS OF FRENCH NPP

The graphics showing theses results are note joigned to this paper. These will must include the doses of 1987.

The themes wich will be presented are :

- collective dose (per reactor, MWyear, ...)
- comparizon EDF and contractor workers,
- international comparizons
- outlook for the future
- exposure data for EDF personnel.

Nevertheless, the most important results we already can show, are :

- about 130 reactor year cumulated in operation

- for 1986 :

  - 2,3 man.Sv/reactor (reactors having an outage during the year)

  - 3 10^{-3} man.Sv/MW year generated for the three last years.

EXPERIENCE WITH ALARA AUDITS IN THE NON-NUCLEAR POWER INDUSTRY

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INTRODUCTION

The Ionising Radiations Regulations 1985 which apply to all work with ionising radiations in the U.K, give statutory significance to the ICRP concept that all exposures should be kept as low as reasonably achievable (ALARA). Under certain circumstances they require the employer to appoint a Radiation Protection Adviser (RPA) to provide advice on compliance with the Regulations. The National Radiological Protection Board (NRPB) acts as RPA to some 600 organisations (largely industrial, medical and research) and in support of this role has established a structured approach (the ALARA Audit) to help determine whether the ALARA principle has been implemented. This approach has been integrated into a documentary package, the RPA Operational Files, which provides an up to date picture for each establishment of the uses, the radiation protection programmes and the resultant dose profiles.

DEVELOPMENT OF ALARA AUDITS

The use of ALARA Audits evolved from a survey, carried out by NRPB, on a random sample of establishments, to determine if conscious efforts were being made to implement the ALARA principle. One conclusion was that establishments tended to give a high priority to meeting regulatory reference levels, but little thought to the actual levels of dose involved or the ALARA principle. Also it was identified that their radiation protection programmes had generally grown in an ad hoc fashion, without any overview of the areas covered, or the effectiveness of the programme. It was concluded that the systematic collection of all data in a structured format to provide a complete picture of the use of ionising radiation and the radiological protection programme was a pre-requisite to any judgment on the attainment of ALARA.

One way of identifying all the points that need to be covered in an ALARA Audit is to produce an analytical tree representing all the components that make up an ideal radio-logical protection programme. A simplified portion of such a tree is shown in figure 1. Such a structured approach has obvious advantages: (a) it focuses attention on radiological protection which in itself may lead to dose reductions simply through improved practice, and (b) it is a mechanism for providing the basic data to be used in more quantitative analyses using, for example, cost benefit analysis techniques(1,2).

INTEGRATION INTO THE RPA SERVICE

The ALARA Audit was one of the fundamental elements on which the RPA Service was built and provided operational advantages for NRPB staff in carrying out their duties as RPA's. To be successful the system of data collection and most importantly its

presentation had to be efficient, easy to use and flexible. Much of the information to be recorded is ideally suited to storage on a word processor, which in turn lends itself to ease of updating. From this a hard copy file can be obtained, structured to cover the up-to-date situation, the basis of previous decisions, statutory records and any analysis. In formatting the structure of the RPA Operational File, it was necessary to consider both our ALARA Audits and the audiences which the file would serve. Principally these would be the user and the RPA. However it was also considered that one objective of the file should be to demonstrate, as far as possible, the user's compliance with the regulations, so a further audience would be the controlling authorities.

FILE STRUCTURE

The RPA Operational File has ten main sections with a number of sub divisions. The first five sections form the core of the ALARA Audit. The essentials are given below.

1. **USES & WORK PATTERNS : Equipment** - sources, X-ray sets, collimators; **Facilities and design based safety** - inherent protection, safety and warning systems, inbuilt emergency capability; **Use** - methods, workloads, procedural safety; **Maintenance work** - what, when, where and by whom.

2. **SAFETY STRUCTURE : Methods of operation and responsibilities** - safety committees, RPA's, Radiation Protection Supervisors, nominated persons (eg authorised to purchase sources); **Training** - typical schedules, formal and 'on job' training received.

3. **PROTECTION MEASURES : Designated areas** (controlled and supervised) - basis, delineation, control of access, consequent actions; **Personal dosimetry** - basis of designation form and schedule of monitoring, dose trigger levels; **Workplace Monitoring** - instrumentation (availability, maintenance and testing), monitoring schedule, trigger levels; **Safety system evaluation** - specification, routine checks (what, how and when).

4. **LOCAL RULES** : Procedures for routine use, maintenance and emergencies; Description of safety system and responsibilities; extent of designated areas, control of access (classified workers and written systems of work).

5. **DOSE ANALYSIS : Assessment of hazards** (from accidents) - prevention, mitigation of consequences, emergency procedures; **Dose analysis** - review of individual group and collective dose profiles (present and trends), link to workload profiles, identification of problem areas.

Essentially the next four sections might be loosely grouped under the heading of 'records', namely: **Dose Records, Other Formal Records** (source records, leakage tests, instrument tests, environmental monitoring results, safety checks, details of maintenance etc), **RPA Reports** (visits, memoranda, investigations, annual reports etc), and **correspondence with Enforcing**

Authorities. The final section of the file is the working/ administrative section for the Company and RPA.

EXPERIENCE OF USE

ALARA audits and the RPA Operational Files have now been in routine use for two years and points of interest arising from this experience are noted below.

(i) **Maintenance/non-routine work:** the area providing greatest scope for staff doses and accidents; but previously lacking attention.

(ii) **Safety structures:** clearly defined responsibilities and communication routes have often been lacking.

(iii) **Training:** receiving increasing attention both for formal training and on the job training by RPA.

(iv) **Site specific trigger levels:** (dose, dose rate etc) rarely used previously; now helping to identify problems at an early stage.

(v) **Planned preventative maintenance schedules:** routine checks (e.g. source accountancy, safety systems, workplace monitoring etc) have been most successful when integrated into such existing schedules.

(vi) **Sources disposal:** upsurge in sources for disposal, previously they had languished unused in source stores or unrecognised in hazardous situations.

At the end of the day it is the resulting levels of dose that matter. Analysis of individual, group and collective dose profiles are carried out and their variation with time is also monitored. It is difficult to disaggregate effects due to the use of ALARA audits and the more general impact of the Regulations, but it is clear that the former are having a positive effect in terms of dose reduction. Perhaps the most striking example is that of an industrial radiography company which was surveyed in the initial project. At that time the annual collective dose was 1.26 manSv, with 20% of the workforce over 15 mSv. Last year, for a similar workload, the comparable figures were 0.34 manSv and 0%.

ALARA Audits are often thought of as having only the function of dose reduction. Whilst this is the end point in many cases, ALARA Audits should, and in our experience often have, identified purported protection measures that are patently not cost effective and which require elimination from radiation protection programmes. For example, we have found that expensive shielding has been installed around some machines to limit dose rates from beta radiation to specified levels, even though there is no associated reduction in dose. Thus ALARA Audits should be thought of more as an element of attaining appropriate resource allocation in radiation protection rather than as a simple dose reduction mechanism.

CONCLUSION

Optimisation, or ALARA, is universally accepted as a primary objective of radiation protection. However, for some it remains an abstract concept. From our experience described above it is suggested that ALARA Audits, linked with quantitative analytical techniques (such as CBA) and the structured presentation of these, provide the mechanisms to translate the concept into reality.

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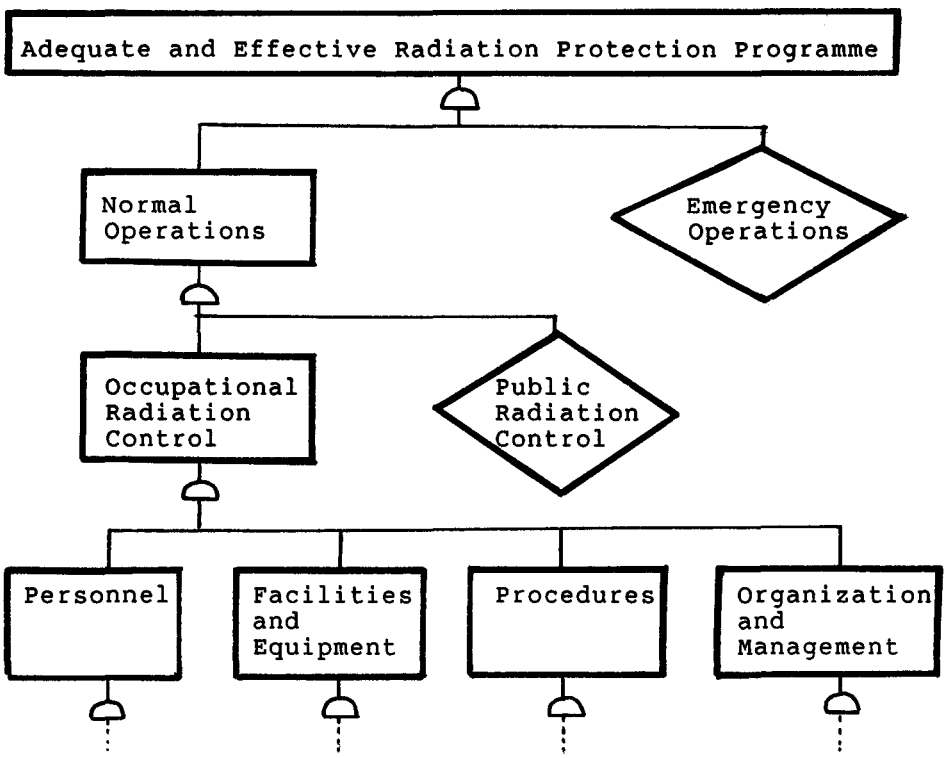

Fig. 1 Simplified portion of an analytical tree.

A SYSTEM OF PERMANENT CODE FOR RADIATION WORKERS

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INTRODUCTION

The International Commission on Radiological Protection (ICRP) has recommended [1] that individual monitoring of radiation workers should be carried out in case the annual doses are likely to exceed 15 mSv (1.5 rem); and that the personal records of all such workers be properly maintained [2]. Workers may receive doses from external radiation or from internally deposited radionuclides. Personnel monitoring badges using dental-size x-ray films (for x- and gamma-rays), nuclear emulsions or solid-state nuclear track detectors (for neutrons), photoluminescent glass or thermoluminescent dosimeters, are used for monitoring doses from external radiation. Analysis of excreta and whole-body radioactivity counting is conducted for assessment of internal dose. ICRP publication [3] gives values of annual limit of intake (ALI) and derived air concentration for this purpose.

PROBLEMS OF CONSOLIDATING DOSE RECORDS

In the present system, when a worker is assigned radiation work in one installation, he is issued one (or more) personnel monitoring badge(s), which he must return after work in that installation is completed or after a specified period (usually 1 to 2 months), whichever is earlier. With this system, it is probable that a person working in several installations may receive doses less than the permissible level in each installation, but the total dose may sometime exceed the annual limit.

Moreover, since different numbering systems are being followed at present by individual installations for the personnel monitoring badges, when a person works in several installations during the same period, he may have different personnel monitoring badge numbers. The determination of the total dose of such an individual, although theoretically possible, is simply impractical on a routine basis.

Hence there has been a long-felt need for a unique system of identifying radiation workers, so that the total dose received by an individual exposed to external or internal sources, in one or several installations, can be routinely and efficiently aggregated, and measures can be taken to avoid over-exposures.

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THE CONCEPT OF PERMANENT RADIATION IDENTIFICATION NUMBER

To solve this problem, it is proposed that a permanent identifying number may be allotted to each radiation worker, which remains invariant throughout his life, irrespective of his place of work. With this arrangement, all the exposures received by him in various installations can be easily entered in a single record of a file, from which the cumulative value as well as the complete radiation history of the person can be readily obtained.

The permanent identification number must reveal particulars such as year of birth, sex, and it should be possibe to allot it on the spot by any local office without reference to the central agency. Moreover, it should be self-checking, so that any mistake in writing, or feeding to computer, may be immediately detected. Since it has to be invariant throughout the life of a person, it should be short and easily memorisable. Any identification formula based on decimal numbers will be too long to be easily memorisable, or to be referred to on every occasion. And if it is not extensively used, the main purpose of integrated accounting of radiation data will be defeated.

APPLICATION OF PHONETIC NUMBERS

The problem of long numbers in respect of personal identification can be easily solved by the application of Phonetic Numbers, recently developed in Bhabha Atomic Research Centre [4]. Phonetic Numbers have radix 128. Their basic numerals of decimal value 0 through 127, are obtained by modulating 16 soft consonants into 8 clear vowel sounds. Hence their every numeral is soft and clear monosound. Phonetic Numbers sound mostly like meaningless words. But because of their short structure and soft sound, they are easlily memorisable and they can be easily referred. They can be written in International Phonetic Alphabet (IPA). But when they are written in any Indian script, they acquire extreme simplicity because each phonetic numeral is represented by a single letter of the alphabet (Table 1). Moreover, the sequence of sound of phonetic numerals is same as the sequence of consonants and vowels of the alphabet. Hence for phonetic numbers, numerical order is same as the alphabetical order in any Indian language. Therefore, all papers labelled by phonetic numbers can be arranged or referred without going into the mathematical intricacies. A computer programme has been already developed in our Institution to work with phonetic numbers in respect of file management [5].

BADGE IDENTIFICATION

Based on the concept of phonetic numbers, a four-digit Permanent Phonetic Radiation Code (PPRC) formula has been evolved for the identification of radiation workers [6]. The first digit (from the left) indicates year of birth. For this purpose, the year 2000 AD has been taken as reference point, and

years of 20th and 21st century are counted backward and forward respectively. By this scheme, the phonetic year code recycles after every 128 years, making the scheme perpetual. The second digit indicates the office of allotment. The third digit can be randomly taken. The vowel sound of the last digit indicates male/female, and its consonant is a check-code which has mathematical relationship with the whole code. By this formula, it is possible to account for 128 years of human age, 128 offices of allotment,and in each office 512 males and 512 females corresponding to any year of birth. In all we get 16,777,216 unique PPRCs.

RELEVANCE IN THE INTERNATIONAL CONTEXT

The scheme of PPRC now being introduced for the first time for radiation workers in India, has the potentiality of extension to all the radiation workers in the world. If the PPRC digits are increased to 6, the number of unique phonetic identification codes, corresponding to a given year of birth, will be more than 2,000 million; and there can be provision to identify nations and laboratories. Under such an International scheme, even in the case of global transfers or migration, the exposure data and radiation history will retain unbroken continuity.

CONCLUSIONS

The adoption of PPRC proposed here will facilitate correlation and entry of all radiation dose data for an individual in a single record of a file. This will prevent inadvertent over exposure of radiation workers, and will also be valuable for biomedical studies of long-term effects of low-level radiation.

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TABLE - 1

FORMATION OF THE BASIC NUMERALS OF THE PHONETIC NUMBER SYSTEM OF RADIX 128 BY SINGLE SOUND CHARACTER OF HINDI SCRIPT AND THEIR EQUIVALENTS IN INTERNATIONAL PHONETIC ALPHABET AND DECIMAL

HINDI	IPA	DEC.	HINDI	IPA	DEC.	HINDI	IPA	DEC.	HINDI	IPA	DEC.	HINDI	IPA	DEC.	HINDI	IPA	DEC.	HINDI	IPA	DEC.
क	ka	0	च	tʃa	16	ज	dza	32	थ	tha	48	न	na	64	ब	ba	80	र	ra	96	स	sa	112
का	ka:	1	चा	tʃa:	17	जा	dza:	33	था	tha:	49	ना	na:	65	बा	ba:	81	रा	ra:	97	सा	sa:	113
की	ki:	2	ची	tʃi:	18	जी	dzi:	34	थी	thi:	50	नी	ni:	66	बी	bi:	82	री	ri:	98	सी	si:	114
कू	ku:	3	चू	tʃu:	19	जू	dzu:	35	थू	thu:	51	नू	nu:	67	बू	bu:	83	रू	ru:	99	सू	su:	115
के	ke	4	चे	tʃe	20	जे	dze	36	थे	the	52	ने	ne	68	बे	be	84	रे	re	100	से	se	116
कै	kai	5	चै	tʃai	21	जै	dzai	37	थै	thai	53	नै	nai	69	बै	bai	85	रै	rai	101	सै	sai	117
को	ko	6	चा	tʃo	22	जो	dzo	38	थो	tho	54	नो	no	70	बो	bo	86	रो	ro	102	सो	so	118
कौ	kau	7	चौ	tʃau	23	जौ	dzau	39	थौ	thau	55	नौ	nau	71	बौ	bau	87	रौ	rau	103	सौ	sau	119
ग	ga	8	छ	tʃha	24	त	ta	40	द	da	56	प	pa	72	म	ma	88	ल	la	104	ह	ha	120
गा	ga:	9	छा	tʃha:	25	ता	ta:	41	दा	da:	57	पा	pa:	73	मा	ma:	89	ला	la:	105	हा	ha:	121
गी	gi:	10	छी	tʃhi:	26	ती	ti:	42	दी	di:	58	पी	pi:	74	मी	mi:	90	ली	li:	106	ही	hi:	122
गू	gu:	11	छू	tʃhu:	27	तू	tu:	43	दू	du:	59	पू	pu	75	मू	mu:	91	लू	lu:	107	हू	hu:	123
गे	ge	12	छे	tʃhe	28	ते	te	44	दे	de	60	पे	pe	76	मे	me	92	ले	le	108	हे	he	124
गै	gai	13	छै	tʃhai	29	तै	tai	45	दै	dai	61	पै	pai	77	मै	mai	93	लै	lai	109	है	hai	125
गो	go	14	छो	tʃho	30	तो	to	46	दो	do	62	पो	po	78	मो	mo	94	लो	lo	110	हो	ho	126
गौ	gau	15	छौ	tʃhau	31	तौ	tau	47	दौ	dau	63	पौ	pau	79	मौ	mau	95	लौ	lau	111	हौ	hau	127

SOME EXPERIENCES FROM THE PRACTICAL APPLICATION OF ALARA PRINCIPLE

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1. INTRODUCTION

One of the basic principles of the ICRP Recommendation,Publication 26 is well known requirement that each exposure to ionizing radiation should be kept as low as reasonably achievable. This requirement is known as the ALARA principle,which serves as the background,for the optimization of radiation protection.(1).This optimization must be applied to both,technical and organizational measures.As such ALARA principle is incorporated,with some modifications, into the radiation protection legislature of the majority of countries where sources of ionizing radiation are being used.

In this paper we discuss incorporation of ALARA principle into Yugoslav radiation protection legislature,as well as an example of its practical application for radiation sources used in metallurgy.

2. ALARA PRINCIPLE IN YUGOSLAV RADIATION PROTECTION LEGISLATURE

Yugoslav legislation which regulates radiation dose limitation consists of two parts: A Law for Protection against ionizing radiation and particular safety measures in nuclear energy use,and B.Regulations.

The law,which is voted by the Federal Parlament,treats problems of radiation protection in a rather general sense,defining radiation protection goals,sources of ionizing radiation,general measures and actions in the practice of radiation protection, etc. Part of the law deals with safety of nuclear instalations and nuclear materials. As far as the ALARA principle is concerned it is nowhere explicitly stated in the law, but it could be recognized in few articles of the law.

In fourteen regulations, which are defined by the law, in details are regulated different aspects of radiation protection in all variety of the use of radiation sources. As far as ALARA principle is concerned the most important of these regulations is one titled "Limits above which population at large and individuals working with sources of ionizing radiation must not be exposed ....". In article 5 of this Regulation it stands:

"Dose for individuals working with the sources of ionizing radiation,members of the public and population at large, from all individual radiation sources,or from all sources which are being used in any practice,are limited by:

1) Justification of the application of a given type of radiation source and justification of use of radiation sources in a given practice.

2) Optimization of the protection from ionizing radiation.

3) Limits of the effective equivalent dose for an individual.

This article elaborated further in the mentioned Regulation, as well as in all other Regulations,serves as the bases for all measures and actions which are to be undertaken in practical work in the field of radiation protection in Yugoslavia.

3. AN EXAMPLE OF THE PRACTICAL APPLICATION OF ALARE PRINCIPLE

In the text to fallow it will be elaborated one example of the practical application of ALARA principle. The example is concerned with radiation protection optimization in the use of radiation sources in metallurgy,where,as it is known,are being used different radiation sources in a considerable number.

One,firstly,has to go through the justification procedure, which in our case is being done according to the following procedure: One first identifies all sources and whose application should be justified. The second step in the process of justification is to consider and meet all general requirements of radiation protection, after which one estimates the price one has to pay due to use of the radiation source in the considered application. This estimation includes nonradiation protection too. In the estimate of total benefit one consideres measurables as well as nonmeasurable social benefits. If one finds that net benefit 0 then application of the source is justified and oposite. Even if 0, one has to do still another step before deciding about justification of the source use-it is nessary to carry out so called relative estimate of the practice, comparing justification of the alternative,nonradiation,practice,with the same benefit.

Step 1. Application which is to be optimized

- Level measurers (137-Cs);
- Thickness measurers (241-Am);
- Thickness measurers with X-ray generators;
- X-ray generators for industrial radiography;
- Ionization smole detectors;
- Neutron sources.

Step 2. Parameters of Optimization Identification

- Shielding (statical and portable);
- Shape of the source container;

- Planing of the operational procedure;
- Definition of the time one spends near the source (regulated by internal regulation and for each working place separately);
- Establishement of the controled areas and limitation of time one can spend in them;
- Automatization of operation,and manipulation from the distance;
- Excercise of the procedure without source;
- Adequate management of the source layout and of the radioactive wastes.

Step 3. Estimate of the Optimal Protection

Two approaches are to be used: a)quantitative and b)qualitative.

(a) One, by the established procedures, carries out an estimate of the equivalent dose below which it is not justified to go with any further reduction.

(b) This approach is applied in those cases where we already have operation with mounted sources. On the bases of Regulations, experience and subjective judgement one estimates the optimization protection.

Step 4. Selection of the Optimal Protection

On the bases of the so far discussed steps one selects the optimal radiation protection for a given use of radiation source.

By using the shown optimization procedure it was possible to reduce considerably the doses of irradiation for individuals employed in a steel plant we treated separately. With smaller modifications this optimization proceudre could be used in many other applications of the sources of ionizing radiation.

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QUALITE ASSURANCE APPLIED TO RADIOLOGICAL PROTECTION

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ABSTRACT

Quality Assurance is required for most industrial activities. In France, a by-law was passed concerning Quality Assurance for Safety of Nuclear Facilities (Code of Practice 50-C-QA of IAEA). What are the advantages of introducing these rules in Health Physics ?

Of the three fundamental recommandations given by the ICRP (Publication 26), Quality can bring effective help to the optimisation of exposure (ALARA) and to the control of exposure limits. It leads to an extensive analysis of the activity concerned, and of the different working positions and conditions in Health Physics. A great deal of exposure can be avoided or minimised, cheaply, if work conditions were considered in more detail. Optimisation can then be reached, at the same time improving workers' security. A list of important elements for Health Physics can be set up, and so optimise technical and human means, thanks to a risk "hierarchisation".

Applied to Health Physics, Quality could be considered in the following way : census and quantification of risks, study of the "cost-advantage" of the different measures it is possible to undertake, study of each working position and optimisation. Of course, Quality can also be applied in a more classical way to all that concerns measuring devices, protection equipment, formation of personnel, projects, etc.

The existence of written documents would allow for a perfect continuity in Radiation Protection, and a good transmission of knowledge. It would ensure a rigourous control of statuary laws and audits would guarantee Quality, which could only improve the protection of workers. The introduction of Quality would fundamentally change the work of Health Physicists. There is no reason why Health Physics should not benefit by modern analysis techniques and Quality Assurance.

Existing rules have incited us to develop essential steps for Quality Assurance practices to be applied to Radiation Protection. In utilizing the safety guideline 50-SG-QA2 of the IAEA, an example can be shown by bringing out a records system. We are therefore studying the development of a new way to approach and solve Health Physics problems.

SHIELDING REQUIREMENTS FOR DIAGNOSTIC AND THERAPEUTIC X-RAY APPARATUS IN SOUTH AUSTRALIA: REGULATORY REQUIREMENTS, SPECIFICATION AND ASSESSMENT OF X-RAY ROOMS

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ABSTRACT

In South Australia the shielding requirements for diagnostic and therapeutic X-ray apparatus depend on the maxiumum power output of the equipment regardless of workload, usage or occupancy.

The method of calculating the required shielding is discussed and the procedure for assessing the degree of radiation protection of the X-ray room once the apparatus has been installed is outlined.

INTRODUCTION

In South Australia, ionizing radiation is controlled by the Radiation Protection and Control Act, 1982 and its Ionizing Radiation Regulations, 1985. Under the Act, X-ray apparatus must be registered with the South Australian Health Commission. However, the Commission cannot register the apparatus unless it has been constructed, shielded and installed in accordance with the Regulations.

The most widely accepted methodology for the calculation of the radiation shielding is that contained in the NCRP Report No.49. Using the equations in Appendix B of this report, the thickness of the barrier for primary radiation, scattered radiation and leakage radiation can be computed as a function of the distance from the radiation source and the product L = WUT, where W, U and T are, respectively, the workload in mA min/week, use factor and occupancy factor.

For new X-ray installations specific information on the value of L may not be available and is usually estimated. An underestimate can lead to inadequate shielding specifications which are expensive to correct after the room has been completed, while overestimates lead to unnecessary costs.

REGULATORY REQUIREMENTS

The Regulations require that diagnostic apparatus be shielded such that:

1. "The air kerma rate:

   (a) 50mm from any wall, door, window, floor or ceiling outside a room, space or enclosure in which the apparatus is installed, being:

      (i) an area continuously occupied by a radiation worker; or
      (ii) a corridor, walkway, lift, stairway, carpark, toilet or other area that is normally occupied by a member of the public for a short time; and

   (b) 50mm from behind a protective screen,

   must not exceed 25 microgray per hour when the apparatus is operated at its maximum rated X-ray tube potential and one half of its maximum continuous tube current at that potential", and

2. "The air kerma rate 50mm from any wall, door, window, floor or ceiling outside a room, space or enclosure in which the apparatus is installed, being an area occupied by a member of the public for other than a short period of time, must not exceed 2.5 microgray per hour when the apparatus is operated at its maximum rated X-ray tube potential and one tenth of its maximum continuous tube current at that potential."

For therapeutic apparatus the shielding requirements are similar to those for diagnostic installations, except that in (2) above one half of the maximum continuous tube current is used instead of one tenth.

The philosophy behind the development of the shielding requirements expressed above was that an X-ray machine should be shielded so that in areas continuously occupied by radiation workers and by members of the public, the air kerma rate should be such that the annual dose limits for radiation workers and general public would not be exceeded when the machine is operated at its maximum capacity. However, calculations of barrier thicknesses indicated that application of this concept was causing overshielding and, therefore, unnecessary costs.

A comparison of the shielding design between the method used in NCRP 49 and that discussed here indicated that overshielding could be prevented if factors of "one half" and "one tenth" of the maximum continuous current at the maximum rated kV were used in the calculation of shielding for radiation workers and members of the general public, respectively. The comparison was made for an existing X-ray room in a large hospital where the workload could be determined accurately. In the calculations made using the NCRP 49 method a weekly exposure of 100 uGy was used for radiation workers. NCRP 49 reports that the cost of shielding for typical diagnostic and therapeutic X-ray installations will only increase approximately 25% if the shielding design for radiation workers is based on a weekly exposure of 100 uGy rather than the maximum design value of 1000 uGy. In other words a tenfold increase in radiation protection could be gained for a 25% increase in cost. An analysis of the shielding materials available locally and their costs indicated that in South Australia for these more stringent shielding conditions the average increase in the cost of X-ray installations is consistent with that mentioned in NCRP 49.

The shielding concept espoused in this work has the advantage that it eliminates contention between the legislative body and the owner of the installation as to whether the equipment has been adequately shielded. It is also consistent with the ALARA principle in which exposure to radiation is as low as practicable without large increases in the cost of shielding.

SHIELDING SPECIFICATIONS

In specifying the structural shielding requirements equations 3c and 6f, Appendix B of NCRP 49, are used where the product WUT is substituted by L calculated as discussed above. It has been found in practice that it is not necessary to consider leakage radiation separately as the shielding requirements calculated for the scattered radiation are also adequate for leakage radiation.

SHIELDING ASSESSMENT

The adequacy of shielding is assessed once the installation is completed using a water phantom with a depth of 200 mm and field area the same as that used in the shielding calculation. A tube voltage as close as practicable to the maximum rated kV of the tube and a charge between 50-200 mAs are used.

A radiation survey is then made with a 100 cm^2 ion chamber. Various critical locations are surveyed, e.g. behind the operator's protective screen, door and joints in wall partitions. The air kerma rate at these locations is then calculated using the appropriate fraction of the maximum continuous current at the kV at which the exposures were made.

SUMMARY

In South Australia the shielding required for diagnostic and therapeutic X-ray equipment depends on the maximum power output of the apparatus only.

The method discussed above has proven to be practical and does not lead to overshielding of X-ray installations and is consistent with the general objective of the ALARA principle incorporated in the legislation. It has also eliminated contention between the legislative body and the owner of the installations over workload, usage and occupancy.

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A MULTI-COMPARTMENT SYSTEM APPROACH FOR OPTIMAL ORGANIZATION OF RADIATION SAFETY

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Years of experience in radiation safety at universities and research institutes revealed the laboratory classification, radiation protection regulations and supervision practice usually applied, to be non-optimal. They are too rigid when applied to specific types of radiation work and are sometimes not rigid enough to enable the use of existing general purpose facilities for other types of radiation experiments.

In light of these considerations, a new and more specific approach to radiation protection organization, supervision and optimization is presented. It is based on a detailed classification of radiation sources of all kinds, which takes into consideration a large number of physical, chemical, biological, technical and quantitative parameters of the source and the mode of its use. Consequently, all other factors involved in radiation work and safety, such as the technical requirements concerning laboratories and control areas, supervision and monitoring, etc. on the one hand, and factors related to radiation workers like personal protection and monitoring, medical supervision, instrumentation, etc. on the other hand, are influenced by this classification. The factors involved in radiation work and safety are mutually dependent and form a multi-compartment system with linkage parameters between its components. A schematic block diagram of the successive information flow and the decision making is presented in fig. 1. Once the system is well defined, any variation in one of the components causes an iterative-like process, until the system is again in "steady state", resulting in an optimal procedure for safe radiation work and proper safety organization. This process enables an individual treatment for each case, and optimizes the safety demands from both the safety and economical aspects. The methodology of establishing the system is based upon a detailed analysis of those sub-blocks of "Personal Radiation Safety" and "Areal Radiation Safety" in fig. 1.

Suppose a block denoted S, which is responsible for a certain aspect of the radiation safety. There are N different means which can contribute to its goal, so that $\{S\} = S_i$, $i=1,N$. At this point all the combinations C_N^k of the S_i's are calculated. Each combination is examined as to its ability to be sufficient for facing a certain radiation source at a certain physical, chemical, biological or technological state. At the end of this process, a new group $\{R\}$ is formed, of radiation sources, where $\{R\} = \{Rj,\ j=1,M\}$. M is the number of different radiation sources that the block S can handle. This procedure is being carried out for each of the blocks which form the multi-compartment model, resulting in an enlarged group $\{R\}$ consisting of a fully detailed variety of radiation sources. A flow chart of the methodology is schemed in fig. 2.

Finally, based on professional experience and judgement, $\{R\}$ can be reduced to a more practical group $\{R^1\}$. Consequently, the blocks "Radiation Workers" and "Radiation Areas" are determined. At this stage, the system is ready for operation. It should be mentioned that the content of each of the sub-blocks of "Personal Radiation Safety" and "Areal Radiation Safety" should

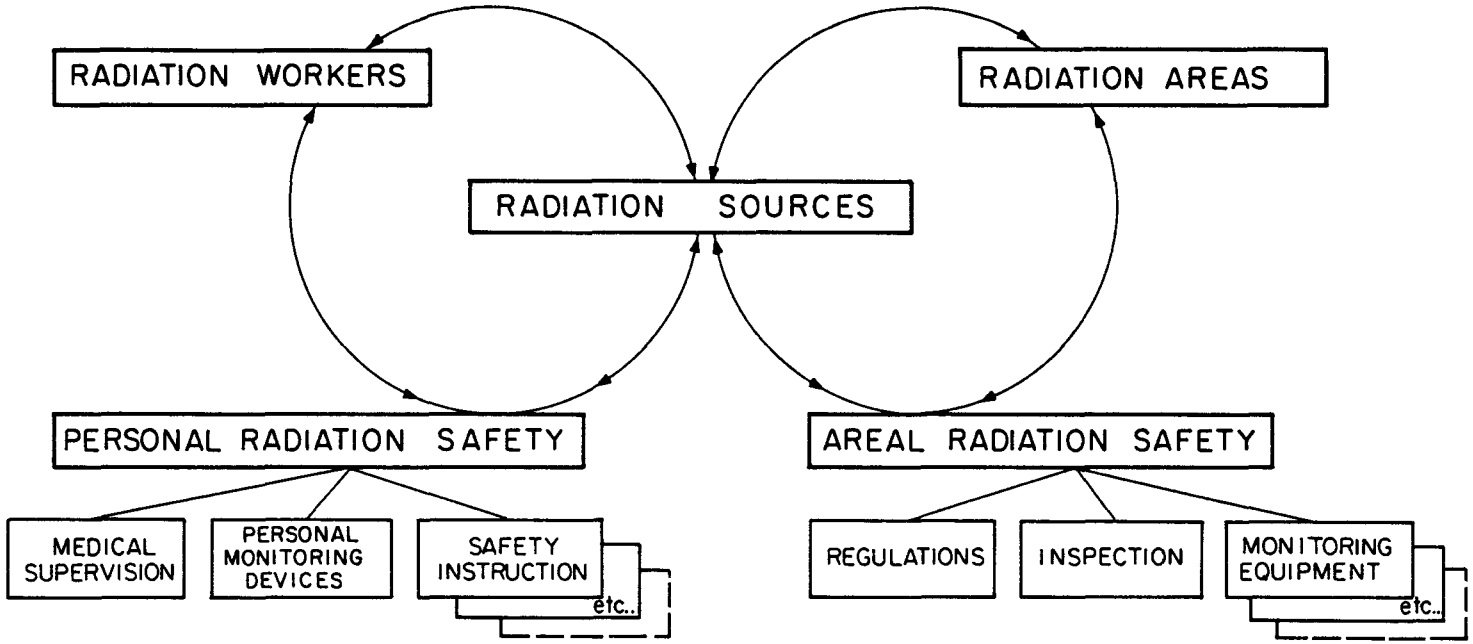

Fig 1 : Schematic Block Diagram of the Successive Information Flow.

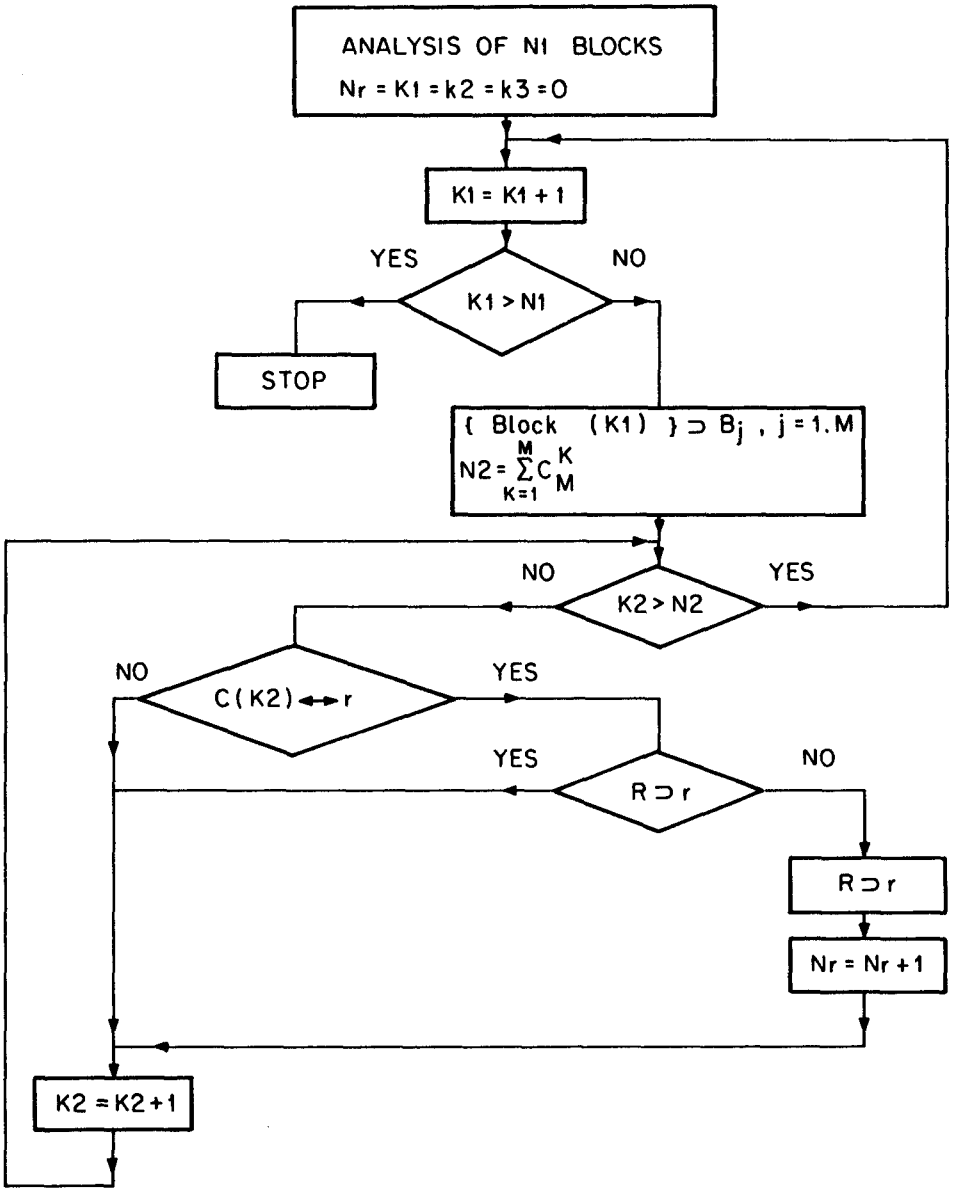

Fig 2 : Flow Chart for Classification of Radiation Sources.

Nr. K1. K2 – COUNTERS
N1 – #OF BLOCKS FOR ANALYSIS
Bj's – MEANS INCLUDED IN THE K1'th BLOCK
N2 – # OF COMBINATIONS OF Bj's
C(K2) – THE K2'th COMBINATION
R – THE SET OF DIFFERENT RADIATION SOURCES
r – RADIATION SOURCE

confirm with the regulations of ICRP.

The computerized integrated radiation safety system described is especially suitable for universities and research institutes where a large variety of open and sealed radioactive sources and radiation producing equipment is used. It interlinks all the steps and provisions to be taken and the necessary regulations to be issued for most practical types of radiation experiments. The flexibility of the system makes it possible to fit an almost individual solution for each case, adjusting a suitable procedure, optimal from both safety and economical aspects.

SOME CONSIDERATIONS ON COSTS INTENDED FOR THE COST-BENEFIT ANALYSIS IN THE OPTIMIZATION OF RADIATION PROTECTION FOR WORKERS

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CONSIDERATION ON THE CONCEPT OF 'COSTS'

To carry out Cost-Benefit Analysis for optimization of radiation protection (ICRP77, ICRP83), first of all, it is necessary to clear the concept of 'Cost'. (Our concepts on costs concerning the optimization of radiation protection are following.)

From the view point of decision basis, one should clearly distinguish between protectional planning for the public and that for workers. The decision for the former must be founded on the social justification of the practice involving social costs so-called NEGATIVE EXTERNALITY in welfare economics. The decision of one enterprise for the latter must be affected by the practice predictable among the other industries of interest. The following is limited only to the protectional planning for workers.

Costs of Radiation Protection are in general considered definite monetary value. However, Costs of Radiation Detriment are so obscure that they have not been recognized as definite quantitative value as well as monetary one. 'Costs of Radiation Detriment' are ICRP's newly coined word and conceptual quantities intending for use in protectional design and planning. It is better to have the sense for the suggestive object in ICRP's concept of the 'Cost-Benefit Analysis in the Optimization of Radiation Protection' rather than straying off into the problem with a sense of values: "Is the conversion of Radiation Detriment into monetary value right or wrong from the ethical point of view ?", because we could not leave 'Costs' out of our cousideration in any protectional decision-making processes whether we recognized those or not.

TWO INCOMPATIBLE OBJECTIVE FUNCTIONS IN PROTECTIONAL DECISION-MAKING PROCESSES

Introduction of radiation protection to an industry must increase costs. This would lead to selling the capital's stocks at a profit. From the managerial standpoint of the industry, 'costs' are likely to be cut off to maximize profits. Seeking profit is a fundamental objective function to the managerial executives.

On the other hand, protectional objective function is also necessary. Costs of Radiological Detriment are defined to justify the investment in protectional measures from the economic point of view. Cost-Benefit Analysis using costs of radiological detriment has emphasized the 'Protectional objective function' which demands an investment of the management in protectional facilities and/or instruments to the extent of 'alaRA' (this expression will be given a little later). The relationship between the two objective functions is schematically given in Fig. 1. The decision-maker should resolve himself to the arbitration between the objective functions in the processes of the optimization of radiation protection.

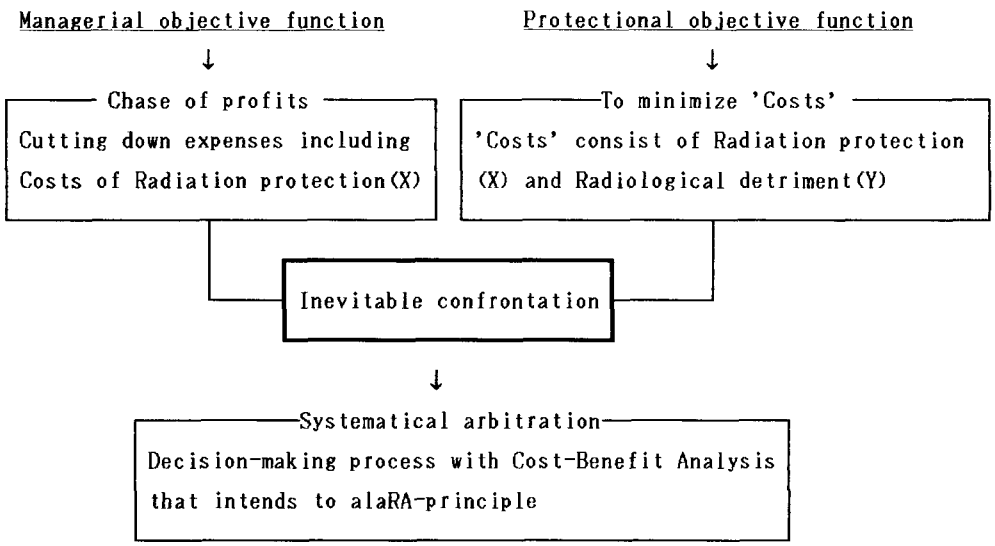

Fig. 1 The Relationship between the two objective functions in decision-making processes intending to the Optimization of Radiation Protection

SOME APPLICATIONS OF COST-BENEFIT ANALYSIS TO GENERAL CASES OF DECISION

Decision-making processes are devided into some steps, namely, plan at the administrative steps, design and operation at ordinary works, negotiation between labor and management, and so on. The decision-making process intending for the Optimization of Radiation Protection is only a part of operational decisions. While, Costs of Radiological Detriment are classified into two types: one is Cost of the objective health detriment with stochastic nature and the other is that of the psychological detriment. These two types of Cost are fulfilled their function separately according to circumstances of decision-making processes.

As a tentative, the authors suggest that the cost of protection must be balanced with that of the objective health detriment with stochastic nature (based on so-called α-value) through the labor and management joint consultation. The other type of cost resulting from the psychological detriment (represented by the β-term) must be discussed in the negotiations on wage expenses. As psy--chological stresses resulted from radiation exposure would depend mainly on personal character of each worker, it would vary diversely and could not be treated in the same way as the stochastic health detriment.

Investigations on the optimum protectional measures and costs have to be distinguished from managerial negotiations on wage expenses. For worker, decision-making procedures for the optimization should be composed of two stages. First, the Cost-Benefit Analysis is carried out simply using the cost of the objective health detriment (as 'alaRA' process). Second, the negotiations including the other non-health ones between the management and the labor union are followed (as 'ALAra' process).

MARKET SYSTEM CRITERIA

Decision-making processes of each enterprise are easily influenced by surroundings, especially by the trends of market. Market price is a criterion to keep the balance between supply and demand on operational decisions. So, Cost-Benefit Analysis cannot be taken into the practice independently of the market system criteria. Unfavorable influences of market system criteria with decision-making processes intending to the present Optimization are noted as follows.

Cost-Benefit Analysis loses its rationality, if the market system is made in keen competition, because decisions of the labor just in above case must stand on the same managerial basis as that of the management to keep the market share through the joint consultation. To avoid unfavorable dumping the theoretical framework on market operations would be elaborated to guarantee reasonable competitions. Further studies are necessary on the relations between decision-makings for the Optimization and the market system criteria.

CONCLUSION

(1) Managerial decision-makings are naturally governed by the profit-chasing objective functions that are apt to reduce the 'Cost'. To convince the workers of the bases of the decision, it is proper to demonstrate the utility of protection cost as saving the cost of the objective health detriment.

(2) For worker, decision-making procedures for the optimization should be composed of two stages. First, the Cost-Benefit Analysis is carried out simply using the cost of the objective health detriments (as 'alaRA' process). Second, the negotiations between the management and the labor union are followed to consider the other non-health ones (as 'ALAra' process).

(3) The market share for the enterprise utilizing radiation being declined excessively, there must be no room for the present optimization because of financial difficulties. The optimum radiation protection and ALARA could be realized under the market system well functioned.

ACKNOWLEDGEMENT

The authors would like to express their sincere thanks to Professor Yasuo Yoshizawa and Assoc. Professor Tomoko Kusama for their invaluable suggestions and discussions throughout the present work.

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THE USE OF MICROPROCESSORS AT TRIUMF IN THE CONTROL OF RADIATION SAFETY INTERLOCK SYSTEMS

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ABSTRACT

At TRIUMF the cyclotron vault, all primary beam lines, and each experimental area has a dedicated control unit to manage the safety interlock control of the area lockup sequence, beam blocker drive and area access. Typically each area has 24 devices which are monitored to control 16 outputs. These control units (Area Safety Units) were first implemented through the use of relay logic. The relay logic was reliable but difficult to modify to incorporate changes to the areas. In 1979 it was decided to use microprocessors in the form of single board computers to control the Area Safety Units. This paper will discuss the details of the hardware and software used as well as the advantages of microprocessor control.

INTRODUCTION

The TRIUMF cyclotron is capable of accelerating H^- ions to an energy greater than 500 MeV at currents exceeding 100 microamperes. This results in radiation fields greater than 1 Gray per hour in certain areas which dictates that personnel must be excluded from these areas during operation. To accomplish this there is an extensive safety interlock system comprised of the Central Safety System (CSS)[1] for global site radiation safety and machine protect, and dedicated Area Safety Units for control of the exclusion areas.

At present there are 12 such personnel exclusion areas. The ASU controls the area lockup sequence and may also provide control over local devices such as beam blockers or radiation emission devices such as RF or DC separators. Each ASU operates independently and makes decisions based on the status of local devices and signals received from the CSS. The results are then sent to the CSS or used to control local devices. Signals are sent from the CSS to the ASU to enable the operation of certain local devices.

TYPICAL LOCKUP AREA

Each area's safety system consists of an exclusion area (walls, fences, etc.), doors, a lockup chain, alarms (horns and beacons) and an ASU. In addition to these devices there must be at least 2 or 3 devices to prevent beam from being delivered to the areas. The lockup chain ensures that a physical search of the area is completed before beam can be delivered to the area. Depending on the physical layout of the area, the lockup sequence may require one or two people to complete.

A lockup chain consists of a series of watchman stations which are connected in such a manner that they have to be armed in a specific sequence. The watchman stations have a push-button to "arm" and an emergency trip push-button or "Panic Button" which is hardwired directly to a fast trip relay circuit. The watchman stations are physically positioned so that the entire area must be checked to complete the sequence.

The door lock mechanism is such that the gate or door must be closed and locked before the key can be removed. Where there is more than one door a transfer lock is included in the lockup routine. The transfer lock mechanism has a tumbler for each door in the area and includes a control key. All the keys from the doors must be inserted and turned in the transfer lock before the control key can be removed and the control key has

to be inserted and turned before any of the other keys can be removed. Either the control key, or if there is only one door, the door key must be inserted and turned to the capture position in the key release unit as one of the conditions to define the area as "secured". Microswitches signal the ASU logic when the key is captured in the release unit and the release of the keys is controlled by the ASU. This method of keying ensures that all doors must be closed and locked during the operation of an area.

MICROPROCESSOR CONTROLLED AREA SAFETY UNITS

When the decision was made to change from relay logic to microprocessors for the control of the ASUs there were a few conditions that had to be met.

1 The program had to be stored in EPROM

2 The system had to be able to self start on a powerup

3 Must use high true logic(+24 VDC for a true or "on" condition).

4 The Central Safety System employs Boolean algebra using standard logic notation ("*" is the AND function, "+" is the "OR" function,etc.) to determine which outputs to set to a true condition. It was desirable to retain the Boolean algebra capabilities in any new design.

The first microprocessor controlled ASUs used a single board computer (SBC) manufactured by Cromemco. The board was Z80 based, had 4 I/O ports, 1 serial and 3 parallel, and included sockets for four 2716 EPROMs. Two of the sockets were populated with EPROMs which contained Control Basic and a monitor program. The Control Basic is an interpreter basic which handles boolean algebra utilizing standard logic notation. Industry standard opto-isolated I/O modules and mounting racks were interfaced to the processor board via a fan-out board and associated cabling. After the program was debugged it was burned into EPROMs which were then installed in the sockets on the SBC. The system would autostart on a reset or powerup.

THE STD BUS

A major component on the single board computer was obsoleted by the integrated circuit manufacturer which rendered the boards impractical for any future designs. It was therefore necessary to select a new system to install in future designs using the same criteria as in the original selection with the additions that:

1 The equipment must be multi-sourced

2 The new equipment must maintain as much compatibility as possible with existing equipment.

For these reasons the STD BUS was chosen. I/O boards are manufactured for the STD BUS which connect directly to the existing module mounting racks through a 50 conductor ribbon cable. These boards are industry standard and are produced by a number of different companies. A wide range of microprocessor boards is also readily available including one which has an EPROM based basic that supports all the instructions included in the Cromemco version with some enhancements. By selecting the STD BUS any future upgrades can be implemented merely by replacing boards in the STD BUS card cage

without any changes to the field wiring. To upgrade the existing microprocessor controlled ASUs to the STD-BUS the single board computer and fan out board are removed and replaced by a card cage, associated boards and a power supply.

AREA SAFETY UNIT HARDWARE (Details)

Input Output Section

External input signal levels are at +24 VDC for a true or safe condition and all devices driven by the ASU must receive +24 VDC or 115 VAC to operate. The +24 VDC and 115 VAC power is supplied by an uninterruptable power supply. Any interruption or break in the cabling to or from the ASU will disable the appropriate devices and fail safe. Plug in modules as per Nema ICS 2-230.02 are used to convert these levels to +5 logic levels for the STD rack. The modules incorporate opto-isolators to provide 2500 VAC isolation between the field wiring and the logic levels for both input and output modules. Racks are available to house 4, 8, 16, or 24 modules and have a separate LED indicator and fuse, between module and external circuitry, for each module. The I/O module mounting racks are connected to the General Purpose Interface Card (I/O board) in the STD BUS via edge connectors and a 50 conductor ribbon cable. Each I/O board can interface to 24 or 48 inputs and/or outputs and require 4 or 8 consecutive port addresses which are set by onboard plug in jumpers. All components in the I/O section are multi-sourced and are readily available.

Microprocessor

A number of processor boards are manufactured for the STD BUS incorporating all of the major microprocessor chips. The board selected for the first STD BUS installations was a Z80 based board with 2 RS-232 serial ports, 4 byte wide memory sockets and a basic interpreter on EPROM. Onboard jumpers are used to configure the sockets for size and type of memory RAM, EPROM or EEPROM. In the development system, EPROM based BASIC firmware occupies one socket, two sockets contain RAM and the forth socket contains an EEPROM. In the operating system the EEPROM is replaced by an EPROM.

ASU SOFTWARE

An ASU program consists of five main areas:

1 initialization; all variables are set to zero

2 variable assignments; the input ports are read and the value is assigned to an array variable. Array variables conserve the limited number of integer variables supported by BWS-Basic. BWS-Basic supports 52 integer variables referenced as A-Z and A0-Z0. Each bit in these arrays is extracted and declared as a integer variable.

3 This section contains the Boolean algebra equations which determine the status of the outputs. The first two lines of this section constitute a watchdog timer routine which toggles the first output module once every cycle through the program

4 In this portion, input latch and timer functions are executed. Input latches are necessary to hold a value for a momentary contact device such as a push-button or door microswitch. The length of time the lockup alarm sounds and the time allocated to complete the lockup sequence are examples of timer functions performed by the ASU. The timers do not use the timer functions incorporated in BWS-Basic as the software overhead required by this function would result in unacceptably long program cycle times. Rather, array variables are set to a numeric value and these are decremented each cycle to a zero value at which point the timed task halts.

5 Here the output variable bits are packed into 8 bit bytes and sent to the output ports. The final line of the program is a statement to return program execution to the start of section #2.

In the change over to the STD BUS BWS-BASIC is employed to implement the logic. The advantage of Basic is that it is readily understood and although the syntax varies from one "BASIC" to another the underlying structure remains the same. When upgrading one the earlier SBC controlled ASUs to the STD BUS the syntax of some of the routines had to be rewritten but the Boolean equations did not change. This meant that once the initialization, input, variable assignment, and output routines had been written the equations could be imported from the SBC version.

The program for the STD-BUS was written on an IBM PC/XT clone using a memory resident program incorporating a full screen editor. The program was written and saved on the PC and then transfered to the STD-BUS using a terminal emulator program that provides ASCII file transfer and capture.

Once the program is transferred to the STD BUS it is tested on a simulator which consists of all of the hardware of an operational ASU but employs switches to simulate inputs and LEDs to display the status of the outputs. During the debugging process changes to the program for diagnostic purposes are written to the STD-BUS CPU RAM while changes to the program are done on the PC and transferred to the STD-BUS. This is to ensure that there is a record of program changes.

The next phase is implemented once the program appears to have all the bugs removed and involves moving the program from RAM to a semi-permanent memory either a battery backed RAM or an EEPROM. This is necessary to check that the program will perform properly on a power up or reset. The program is then thoroughly checked using the simulator.

The development system is now ready for long term testing which involves setting up the simulator to mimic the operating modes of the lockup area it will be installed in. It is run in each mode for a number of days to try and induce a failure. If the program passes this test it is transferred to an EPROM which is then installed on the CPU board. The long term tests are again run and when these are passed the CPU board is ready to install in an ASU. The first STD BUS controlled ASU was installed in the Spring of 1987. There is a TRIUMF Safety Group document available with further details on both the hardware and software for this system.

The major advantages of the microprocessor installations are that any changes to the operating modes of an experimental area generally only necessitate modifications to the software. Previously, any modifications to the relay logic often required major rewiring. There was no simulator available for the relay logic ASUs. This meant that the major testing had to be done after the changes were implemented. Both the modification and testing involved considerable time. This made for very inflexible systems. With the use of microprocessors all the changes and major testing could be done prior to installing the CPU board in the ASU. The use of the microprocessor has resulted in more flexible and reliable safety interlocks for experimental areas.

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GUIDE TO DESIGN A MONITORING PROGRAM ACCORDING TO ICRP 35

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ABSTRACT

Part I

The ICRP, in its publication 35, gives the General Principles of Monitoring for Radiation Protection of Workers. It is a very nice publication, very concise but difficult for anyone that want to design a monitoring program for his specific situation and also to satisfy national regulatories bodies. For this reason, the present paper intend to provide a guide to select different types of monitoring to be introduced in a particular situation satisfiyng the ICRP-35.Our guide, in this first part was subdivides into the following topics:

1 - The aim of the monitoring program
2 - Supplementary benefits of the monitoring
3 - The different types of workplace monitoring and their selection for a specific case
4 - The different types of individual monitoring and their selection for a specific case.

As in ICRP we consider also the routine, operational and special kinds of monitoring. In each paragraph of our paper we mention, in parenthesis, the corresponding paragraph of ICRP-35 that it can furnish more details about the subject.

Part II

If a monitoring program of the workplace for external radiation should be introduced in specific work situation, the present paper gives a guide to implement a monitoring program according ICRP-35 on a simple manner. Our design was subdivided into the following topics:

1 - Introduction
2 - Choice of the monitoring points
3 - Equipment used
4 - Description of the model used to estimate the result in terms of the ICRP recommendations
5 - Interpretation of the result
6 - Quality assurance
7 - Reassessment of monitoring program
8 - Record keeping of the abnormal situations

In each paragraph of our paper we mention, in parenthesis, the corresponding paragraph of ICRP-35 that it can furnish more details about the subject.

REPORT OF THE INTERNATIONAL RADIATION PROTECTION ASSOCIATION LONG-RANGE PLANNING COMMITTEE

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An ad hoc long-range planning committee was formed by the "new" Executive Council meeting during the VIth International Congress of the International Radiation Protection Association in Berlin. Professor Nishiwaki, Dr. T. Schlesinger, and Charles Meinhold (Chairman) were appointed to the Committee at that time. They were instructed to review and evaluate current programs and activities of the IRPA and to consider additional programs and activities which should be given careful consideration by the Executive Council. At the Executive Council Meeting at Ispra in 1985, Dr. John Johnson, who had chaired the Associated Societies Panel Committee, was appointed to the Committee.

A letter containing a list of potential topics was prepared for distribution to the Associated Societies with the intention of stimulating their thoughts and ideas and requesting comments and suggestions. This letter was printed in the IRPA Newsletter (Vol. 7, No. 1) and was sent to all Associated Society Presidents in 1986. During the Salzburg meeting in December 1986, the Committee reviewed the responses from the Associated Societies and although it noted that the response from the Associated Societies was disappointing in number, there was a general consensus of many of the points raised in the letter.

During discussions at the Tokyo meeting of the Executive Council, it was decided that the final committee report contain a review of the responses together with a recommendation for further actions by the Executive Council on each of the topics presented in the original letter. The Associated Societies which provided comments and suggestions are:

1. Australian Radiation Protection Society
2. Association Belge de Radioprotection
3. British Radiation Protection Association
4. Canadian Radiation Protection Association
5. Fachverband für Strahlenschutz
6. Societe Francaise de Radioprotection
7. Health Physics Society
8. Associazone Italiano de Protezione Contro le Radiazioni
9. Radiation Research & Protection Society of Ireland
10. Japan Health Physics Society
11. Nordic Society for Radiation Protection
12. South African Association of Physicists in Medicine and Biology.

In this presentation, each of the main topics from the original letter to the Associated Society Presidents will be presented, followed by a review of the Associated Society responses, and summarized by a recommendation to the Executive Council for further action.

MEETING FREQUENCIES

The IRPA Congress meets every three to four years - this seems adequate for the scientific exchange but may be inadequate for the exchange between the Associated Societies.

Regional congresses might be made a more formal part of the IRPA program, i.e., the Executive Council might be required to meet at such congresses and the presidents or other representatives of the Associated Societies might be invited to report on their activities and to discuss the IRPA needs as part of a formal Associated Societies forum chaired by the IRPA President. Each Associated Society would be asked to designate a representative and a well-defined agenda of up to four topics would be circulated prior to the meeting with a specific society asked to provide the leadership for each topic.

Associated Society Comments

The present frequency of IRPA congresses is fully accepted. There is general acceptance of increasing the role of the IRPA Executive Council in regional congresses in order to promote interaction between IRPA and the Associated Societies and between the Associated Societies themselves.

Many of the Associated Societies mentioned that financial consideration might inhibit their attendance at regional congresses.

Recommendations

1. The Regional Congress Committee be requested to explore the development of Associated Society forums as a requirement for the Regional Congress program.
2. The Executive Council should hold its meetings during the Regional Congresses.

INIRC

As the International Non-Ionizing Radiation Committee (INIRC) becomes the internationally recognized body for making recommendations on non-ionizing radiation, its relationship to IRPA should be more permanent, more rigorous, and more distant. Perhaps we should look at the relationship between ICRP and the International Society of Radiology as a model, i.e., the Committee would report to the IRPA Congress but in essence run its own affairs. There is a suggestion that

perhaps such independence should take place when INIRC has more clearly become the internationally recognized authiority in this area.

Associated Society Comments

There was general agreement that eventually INIRC should be similar to the ICRP in its relationship with IRPA although the societies want to ensure the availability of feedback on recommendations.

Recommendations

The Executive Council should establish an ad hoc committee to develop a new relationship with INIRC based to some extent on that between ICRP and the International Society for Radiology. The target date should be no sooner than the time of the VIII International Congress of IRPA in 1992.

CERTIFICATION

This is one of our more important topics, particularly if there is a desire that we be perceived as a professional rather than a scientific organization. There is a feeling that certification ought to be the function of the individual societies since, if it is not, it must, almost of necessity, be driven to the lowest common denominator. For example, let us assume that Society A wants to identify the top 1% of its radiation protection personnel since they will become inspectorate cadre, whereas Society B wants to identify the top 50% of its radiation protection personnel since they want all their supervising health physicists to meet an existing standard. Although Society A has legal meaning to its certification program, Society B does it only for professional edification. Under these two extremes, a common certification program would be almost impossible. It would seem, however, that the IRPA ought to be involved in helping the Associated Societies develop their own certification program through workshops, training sessions, and via information exchange.

Perhaps the IRPA should also assist in the development of minimum requirements for professional health physicists. In addition to stimulating individual professional development, the Associated Societies or competent authorities could use this material as a basis for their certification.

Associated Society Comments

General agreement that IRPA should only coordinate workshops and the development of data bases although some suggested that the IRPA meet "minimum" standards. Some societies stressed the importance of IRPA being primarily a "scientific" organization.

Recommendation

The International Congress Program Committee should be tasked with programming workshops on certification methods and requirements. Alternatively, an interested Associated Society could be asked to assemble pertinent information on legal qualifications, examinations, etc. for distribution to the Associated Societies.

COMMUNICATIONS

In the IRPA Constitution we find, "Its primary purpose is to provide a medium whereby those engaged in radiation protection activities in all countries may communicate more readily with each other and through this process advance radiation protection in many parts of the world." Clearly the Congress provides much of its required communication. The suggestion of an Associated Societies Forum given above would improve communications both between the Associated Societies but also between the Executive Council and the Associated Societies.

Perhaps another important "communication" service the IRPA could perform is to be the distributor and collator of the comments received on documents such as ICRP and ICRU draft reports. Much of the ICRP's inability to allow comments on their drafts has to do with their inability to handle this paperwork. This is a function which the IRPA might consider. There has also been a suggestion that the IRPA establish a library of regulations in radiation protection for access by the IRPA members. If this proves too difficult, perhaps the publication of an annual bibliography might be sufficient.

Associated Society Comments

There is a widespread desire to have IRPA act as a clearinghouse for ICRP and other international draft reports. There was a general acceptance of the Associated Society panel focusing on communication between both the IRPA Executive Council and the Associated Societies and between the Associated Societies themselves.

Recommendations

The President of IRPA should communicate with the Chairman of the ICRP to develop the liaison suggested here. The IRPA Executive Council should, at the time of each international congress, appoint one Associated Society as the focal point for synthesizing comments and suggestions on ICRP matters for transmittal to the chairman of ICRP's Committee 4. This method of providing IRPA's input to other organizations should be considered.

SCIENTIFIC AND PROFESSIONAL DEVELOPMENT OF THE MEMBERS

Perhaps we should consider formalized tutorial sessions at both the IRPA Congress and at the regional meetings. These might take the form of early morning or late evening sessions or even weekend or full-week programs before or after such meetings. Conversely, perhaps this is best left to the Associated Societies - language difficulties alone might make this option more effective.

Associated Society Comments

Wide range of Associated Societies reactions from formal tutorial sessions at the international congress to an ICRP-developed training module to simply provide data bases and workshops on technique for use in national meetings. General note that we don't want to detract from the general scientific exchange.

Recommendations

The Executive Council should request that the next International Congress Program Committee develop a workshop on this topic in conjunction with the International Congress meeting in 1992.

PUBLIC INFORMATION

This is another area with some dangerous pitfalls in terms of the differences between countries. As with certification, the development of workshops and dissemination of Associated Socieities materials might be our function here.

Associated Society Comments

General acceptance of workshops and data bases by IRPA. Implementation should be left to the Associated Societies.

Recommendations

The Executive Council should request the next International Congress Program Committee to develop a workshop on this topic in conjunction with the International Congress meeting in 1992.

FULL-TIME SECRETARIAT

Much of the above could not be accomplished without present administrative structure. This may well be the topic that we should begin with, since much of the additional communication discussed above could only be possible with a

full-time paid secretariat. This, however, would be very hard on the Associated Societies who can't afford this luxury.

Associated Society Comments

Nearly all agree it is desirable but too expensive. Some suggested that the Executive Council be selected from among the full-time secretaries in the Associated Societies.

Recommendations

The IRPA Executive Council should form an ad hoc committee to review this matter in detail.

FINANCES AND FINANCIAL HARDSHIPS

Perhaps some specifically identified IRPA programs should be supported by grants from either the Associated Societies or by outside organizations. Perhaps we need to develop special arrangements for dues, meeting registration, journal subscriptions, etc., for members from societies with severely limited financial resources. Perhaps the more affluent societies could assist the less affluent societies by grants in aid.

Associated Society Comments

Many Associated Societies felt the more affluent societies could assist the less affluent societies although there was no support for an alternate dues schedule. The Fachverband made a suggestion for IRPA 7 which demonstrates in an excellent manner what can be done.

Recommendations

The IRPA Treasurer should communicate with the less affluent societies to determine ways that affluent societies could help. He should then communicate these needs to the more affluent societies.

IRPA RESPONSE TO SCIENTIFIC ISSUES

An additional topic has been suggested by the Societe Francaise de Radioprotection. They suggest that IRPA could play an important role in providing an international consensus on specific scientific issues. This suggestion has been favorably received by several other Associated Societies.

Recommendations

The Executive Council should form an ad hoc committee to develop guidance and procedures for dealing with IRPA's

response to important scientific issues including addressing concerns to the international organizations where appropriate.

FURTHER SCIENTIFIC ACTIVITIES

At the Tokyo meeting of the Executive Council, President Carter suggested that we might select scientific areas in addition to non-ionizing radiation for IRPA activities.

Recommendation

The Executive Council should include this topic in the charge to the ad hoc committee recommended in response to the suggestion of the Societe Francaise de Radioprotection given above.

In summary then, it should be apparent that the work of the ad hoc committee on Long-Range Planning is complete, the IRPA effort is only beginning. The Committee was encouraged by the thoughtful and helpful suggestions made by the Associated Societies during this process. More important, however, is the concept that the Associated Societies become aware of their responsibilities to actively participate in developing the course of IRPA's role.

INVESTIGATION OF AERIAL DISPERSION OF RADIOACTIVE DUST FROM AN OPEN-PIT URANIUM MINE

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ABSTRACT

Detailed investigation of aerial dispersion of radioactive dust from the biggest uranium mining and milling operations in Australia has been carried out. Spatial distributions of the long-lived uranium series radionuclides and their origin (mining and milling operations versus natural radiation background) have been studied.

Air concentration, horizontal flux and dry and wet deposition of the radionuclides were investigated along 45 km transect, in the direction of the prevailing monsoonal winds in the region.

INTRODUCTION

Ranger Uranium Mines operates an open-pit mine and treatment plant at a remote, subequatorial part of the Northern Territory. Characteristic for the region are two distinct seasons, dry and wet, with steady monsoonal winds. Ranger deposits are located c. 200 km east of Darwin, in the lowlands of the Alligator Rivers region.

About 6 mln tonnes of rock is mined annually from the only operating orebody no.1, resulting in an annual output of c. 3000 tonnes of uranium oxide (U_3O_8).

About 1500 people live permanently near Ranger, among them c. 1200 in Jabiru township, distant 8 km WNW of Ranger, and c. 300 in Jabiru East, 3 km NW of Ranger (Fig. 1).

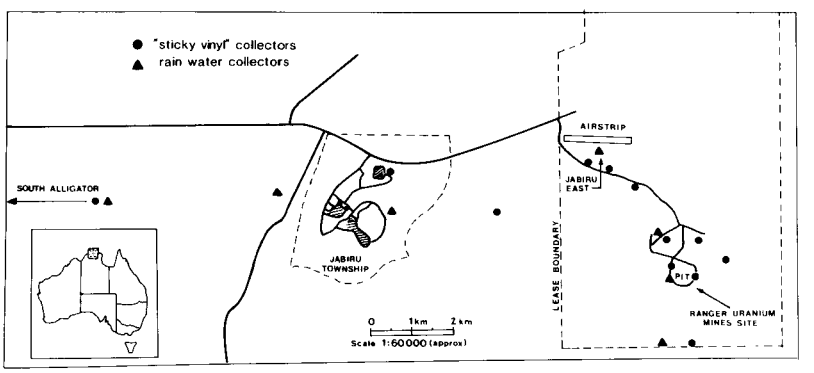


Fig. 1

The principal aim of the study was to investigate the spatial distribution of the longlived radionuclides of the ^{238}U series and their origin i.e. mining operations versus natural radiation background, along the path of the monsoonal winds in the region. Incidentally, the Ranger site and the townships are located along that path, with the prevailing dry season (April-October) winds blowing towards the townships.

The investigations were conducted from mid 1984 till the beginning of 1987. They included passive and active dry deposition collection, and

were complemented by wet dust collection during the wet seasons of 1984/1985 and 1986/1987. The wind frequence for E+SE winds are on average 75 %, 20% and 50% in dry season, wet season and all seasons respectively.

MATERIALS AND METHODS

Passive "sticky vinyl" dust collectors were used as principal dust monitoring devices during the dry seasons. The collectors were made of a clear self-adhesive vinyl ("Contact" brand, Nylex Corp. Ltd, Victoria). Each collector consisted of a sheet of the vinyl streched on a plywood frame, with areas between 0.35 to 1 m^2. Horizontal frames were at 1 m height and the vertical frames were 1.5 m above the ground at their centre. The monitoring sites extended along up to 45 km long transect from the Ranger site, parallel to the path of the prevailing winds in the region (Fig. 1). The collectors were changed about once per month.

Periodically, in some of the above sites, simultaneous active dust sampling was conducted by means of high volume air filter sampling. In most sites, wet deposition of radionuclides in rainfalls was also carried out by means of 30 liter sealed plastic containers equipped with 0.20 m diameter collecting funnels.

RESULTS AND DISCUSSION

The investigation on horizontal transport of radionuclides in dust from the pit, was carried out during part of the dry seasons 1984-1986. Figure 2 displays the average load on vertically oriented "sticky vinyls", all facing the pit along a transect from approximate SE to NW direction. The observed decrease of activity load, J, with distance from the pit, x, in NW direction, is represented by expressions of the form;

$$J = k \cdot x^a$$

where J is expressed in $Bq \cdot m^{-2} \cdot d^{-1}$ and x in km, and k and a are constants for each radionuclide. The line of best fit using this expression is shown for each radionuclide in Figure 2, and the values for k and a are given in Table 1. The data for x=45 km are considered to represent natural background levels and were therefore omitted in the fitting exercise.

TABLE 1.

Vinyl orientation	Radionuclide	a	k ($Bq \cdot m^{-2} \cdot d^{-1}$)	r
Vertical	$^{238,234}U$	-3.18	0.234	0.97
"	^{232}Th	-1.51	0.0015	0.85
"	^{230}Th	-3.10	0.236	0.96
"	^{226}Ra	-2.92	0.367	0.99
"	^{210}Pb	-2.40	0.243	0.97
"	^{210}Po	-2.44	0.079	0.87
"	Average, except ^{232}Th	-3.0	0.25	0.94
Horizontal	$^{238,234}U$	-2.67	1.93	0.98

a and k are constants described in the text, r is the correlation coefficient.

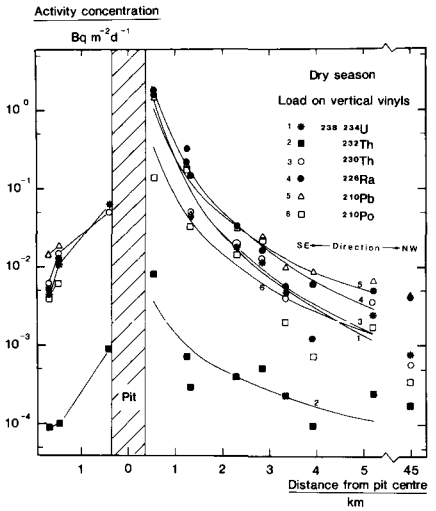


Fig.2

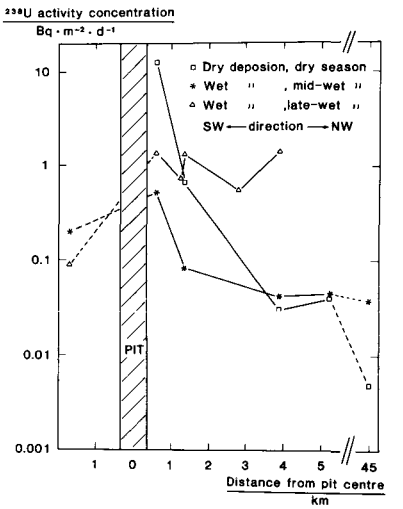


Fig.3

The data for the uranium series radionuclides, except ^{210}Pb and ^{210}Po, are well described by an inverse cubic dependence of activity load versus distance. The less rapid decrease of load for ^{210}Pb and ^{210}Po reflects a faster approach to the relatively higher natural background for these nuclides. For ^{210}Po is the load as well generally much lower, which is unexpected. We believe the reason might be evaporation of ^{210}Po from the vinyls. For ^{232}Th, where we have found that activity concentrations in soil are of the same order of magnitude in the pit as in the surroundings, is the activity load expectedly much lower and a faster approach to natural background levels is found.

Along with vertically oriented vinyls were horizontally oriented vinyls exposed at 5 joint sampling locations for the same time periods. The latter vinyls will act as dry deposition/ fall-out collectors, but will not distinguish between different source directions. Figure 3 shows the observed dry deposition activity concentrations for ^{238}U as a function of distance from the pit in the NW direction. A fit of the data, using the same expression as for vertical vinyls, gives a somewhat slower decrease rate (Table 1), which is expected because of contributions to dry deposition from sources with directions other than the pit area, i.e. natural background sources.

By gravimetric measurements on the mass load per unit area of horizontal vinyls, were the specific activities on the vinyls determined (Table 2). It is interesting to note the closeness between surface soil activity concentration in the pit, which corresponds to about 0.18 % uranium, and the dry deposition concentration close to the pit, although the former is only a grab sample and does not represent an average concentration in the pit surface soil. In the surroundings, more than a few km's away from the pit, will the dry deposition concentrations clearly exceed the surface soil concentrations, giving a activity contribution of uranium series radionuclides to the top soil, surfaces like plants and water of the order of 5-100 $Bq \cdot m^{-2} \cdot y^{-1}$.

The wet deposition of radionuclides was studied during the wet seasons by means of rain collectors, at sampling locations identical to the dry season study. The radionuclide load per unit area in rainfall

was determined and Figure 3 shows the results for ^{238}U wet deposition rate during "mid-wet" season and "late-wet" season. The radionuclide load is overestimated because dry deposition was not prevented. The results for the "late-wet" period indicates an almost distance independence on load in the NW direction. This contrasts sharply with dry deposition data (Fig. 3), but if we assume a wash-out of the total air column and that transverse dispersion at distances comparable with the source diameter is not significant, and that the plume height is less than the cloud height, one could expect an almost constant radionuclide load with distance from the source.

TABLE 2.

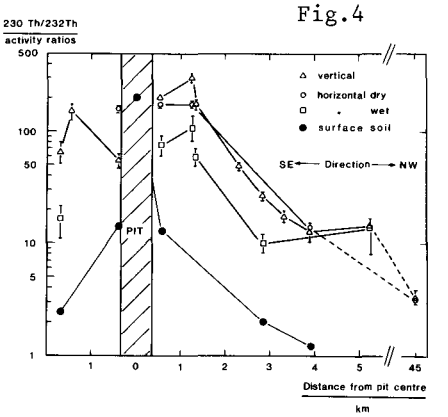
Specific activities ($Bq \cdot kg^{-1}$) of radionuclides in dust from dry deposition and surface soil.

Distance from pit centre (km)	Horizontal vinyl $^{238,234}U$	^{232}Th	^{230}Th	Surface soil $^{238,234}U$	^{232}Th	^{230}Th
0	-	-	-	23000	115	23250
0.55	20250	130	22800	2120	235	3090
1.325	7605	45	7920	425	-	-
2.95	-	-	-	115	75	150
3.9	220	18	260	65	65	80
5.2	240	-	-	25	-	-
45	-	-	-	20	-	-

During the "mid-wet" period is the wind no longer pronounced in the E-SE direction. This would reduce the activity load in the NW direction. The results (Fig. 3) confirms that and also shows that for locations like Jabiru-East and Jabiru is the dry and wet deposition of radionuclides of the same strength.

Figure 4 shows the average $^{230}Th/^{232}Th$ activity ratios obtained for vertical and horizontal vinyls and surface soil. The ratios drop fast for the surface soil when leaving the pit area compared to the wet and dry load. This makes clear that the pit area is acting as a source of radioactive dust along the transect. If we use a $^{232}Th/^{232}Th$ activity ratio of 200 for the pit and between 1 and 2 for the surroundings we find that for Jabiru-East and Jabiru will the aerial dispersion from the pit area give a 4-9 times increase in dry and wet activity load , compared to natural background levels.

Fig.4



By comparing the load of radionuclides on vertical vinyls exposed simultaneously in 4 major wind directions at 4 sites along the transect with radionuclide concentrations in the air by air filter sampling, the load on vertical vinyls was transformed to "generated" air concentrations. For the locations Jabiru-East and Jabiru the obtained air concentrations would give rise to a committed effective dose equivalent of the order of 0.05-0.20 mSv for one year exposure.

RADIOACTIVE CONTAMINATION OF STEEL

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In 1986, we reported seven instances of accidental radioactive contamination of steel manufactured or imported into the United States as a result of smelting of sealed sources (Lu 86). Since then, two additional cases of accidental smelting of radioactive sources have been reported, both occurring in 1987. One is similar to earlier events in that 740 to 930 MBq of ^{137}Cs, probably from one or more gauges, was accidentally melted in a steel mill. The event was discovered when a truckload of flue dust from the mill caused a radiation monitor at the mill's weigh station to alarm when the truck passed. Contamination was confined to flue dust. The second event was unique in two respects, the radioisotope and the smelted metal. In this case, a radium source which came perhaps from a gauge or from a nuclear medicine program was mixed with aluminum metal scrap. Dross (slag) from the smelter was then transferred to a salvage yard where radiation monitoring of the incoming dross resulted in detection of the radioactivity. The contamination was confined to the dross. In both cases the contaminated smelting by-products - flue dust and dross - were collected for appropriate disposal. No cost estimates for these incidents are presently available.

In 1986, the U.S. Nuclear Regulatory Commission published a hazardous scrap warning poster and distributed it to U.S. ferrous metal scrap dealers and mills (NRC 86). The poster is intended for display at scrapyards and mills to inform workers of the possibility of radioactive material in scrap, how to identify warning labels, and who to contact for help. Following the aluminum contamination incident, additional distribution was made in 1987 to the non-ferrous metal industry. The U.S. Occupational Safety and Health Administration and the National Association of Recycling Industries assisted us by providing mailing lists.

In 1987, with the help of Professor John F. Elliott of the Massachusetts Institute of Technology, we made available a table showing the likely primary pathways of radioisotopes in steelmaking furnaces (NU 87). The isotopes selected are those frequently used as sources in industry or medicine (Table 1).

The well publicized incidents of steel plants becoming contaminated by melted sealed radioactive sources have provided

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*This article was prepared by employees of the U.S. Nuclear Regulatory Commission (NRC). The NRC has neither approved nor disapproved its technical content.

Table 1

Likely Primary Pathways of Selected Elements in Steelmaking Furnaces*

Element	Likely Pathway**
Co	Steel
Cs	Vaporizes***
Ra	Slag
Am	Slag
Ir	Steel
Pb	Vaporizes***
Po	Vaporizes***
Sr	Slag
Pu	Slag
U	Slag
Y	Slag
Th	Slag

*Typical temperatures of steel making furnaces: Basic Open Hearth, 2750 to 2950°F (1482 to 1610°C); Basic Oxygen (O_2 lance), 2750 to 3020°F (1482 to 1660°C); Electric Arc, 2750 to 3050°F (1482 to 1677°C).

**No consideration given to chemical form. In cases where slag or vaporization is the primary pathway, there may be low residual levels of the element in the metal bath. Persons using this table should consider it as guidance only and not as a definitive predictive statement of the ultimate disposition of radioisotopes in a specific case. Confirmatory measurements are recommended in all cases.

***Liable to be collected in pollution control equipment.

a powerful incentive to the steel industry to monitor incoming scrap for radioactivity. Installation of radiation monitors (usually costing a few thousand dollars) appears to be a wise investment even given some uncertainty of the detectors always finding such sources.

Scrap steel arrives in steel mills or scrap yards in trucks, railroad cars, or barges and since these routes of entry are controlled, the detectors are usually mounted at entry gates or at weigh scales. Typically, 5 x 5 cm NaI (Tl) detectors are mounted above or at the sides of the vehicles. Alarm points are frequently set at about 2X-3X background, which is high enough to avoid "false alarms" caused by fluctuations in background

counting rates. Total reliance should not be placed on such systems to detect radioactive sources in scrap. Studies have shown that a large source in a well-shielded container in the center of a railroad gondola car or truck filled with scrap can go by undetected by such systems (La 86). These detection systems are, however, reliable, provide a degree of protection for scrap handlers that was not available before and have proven effective in identifying radioactive sources in scrap metal. Thus, the authors believe that use of the poster combined with a radiation detection system helps prevent unknowingly receiving sources in scrap shipments.

We do not have figures on the numbers of steel plants and scrap yards in the U.S. that have installed monitors but we can guess the numbers are increasing because of the increasing numbers of reports of another source of radioactive contamination of scrap-naturally occurring radioactive materials - NORM.

In the U.S., steel scrap contaminated with NORM comes from places as far apart as Alaska and Florida and has originated from oil and phosphate industries, from a kaolin clay plant, and from water softening equipment. In most cases the NORM was detected by stationary monitors and contact gamma radiation levels (for cases where we have data) ranged from 0.5 μSv/hr to 0.1 mSv/hr at contact.

An unusual case involved a cattle guard fabricated directly from recycled steel oil drilling pipe. Maximum radiation levels were 15 μSv/hr contact.

We are aware of only one case where such scrap, once found, was sent to a commercial low-level waste disposal site. In the other cases, the contaminated scrap was either off-loaded and stored on-site at the steel plant or yard or returned to the plant that generated the scrap. Obviously, these are neither permanent nor satisfactory solutions.

In the U.S. disposition of the scrap is made difficult by the absence of regulatory standards for handling NORM. Except for radium in uranium and thorium mill tailings, NORM is not covered by the U.S. Atomic Energy Act.

Whether the radioactivity in scrap metal is from sealed sources or from NORM the burden of this problem is presently being carried by the scrap dealers and mills. Radiation monitors cost several thousands of dollars and up. Trip and demurrage charges for rail cars transporting scrap can also run into the thousands of dollars. Special handling or hauling of identified radioactive scrap carries labor costs. Shouldn't the responsibility for preventing, controlling and monitoring for radioactivity in metal scrap be at the generator's end?

We have no conclusions per se. This report is an interim one on a problem that is in search of a solution. To reach a

solution will require the cooperation of source generators, the metal industries, trade associations, governmental and voluntary standard setting agencies and the radiation protection community.

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TRANSURANIC WASTE TRANSPORTATION ISSUES IN THE UNITED STATES

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New Mexico Environmental Evaluation Group

The United States Department of Energy (DOE) expects to begin disposal of defense transuranic wastes at the Waste Isolation Pilot Plant (WIPP) in Southeastern New Mexico before the end of 1988. Approximately 25,000 truck shipments involving 35 million vehicle-kilometers will be required to transport about 175,000 m^3 of contact-handled transuranic waste. Up to 5,000 shipments of remote-handled transuranic waste (RH-TRU) will also be shipped to WIPP in shielded casks. This paper addresses only the shipment of CH-TRU wastes. The locations of waste generating sites and the WIPP site are shown in Figure 1.

The DOE designed a shipping container called TRUPACT-I for the purpose of shipping CH-TRU waste. TRUPACT-I was rectangular in shape with overall dimensions of 7.6 m long, 2.4m wide, and 2.7m high. It was designed to carry 36 0.208m^3 drums or a similar volume of boxes with a payload of 8300 kg. The design included two features that would have prevented certification by the U.S. Nuclear Regulatory Commission (NRC) unless a variance were allowed: (1) the package had only single containment; and (2) the waste cavity was continuously vented through high efficiency particulate (HEPA) filters. Although the DOE is not required to obtain NRC certification, their own certification process is intended to be equivalent to the NRC process.

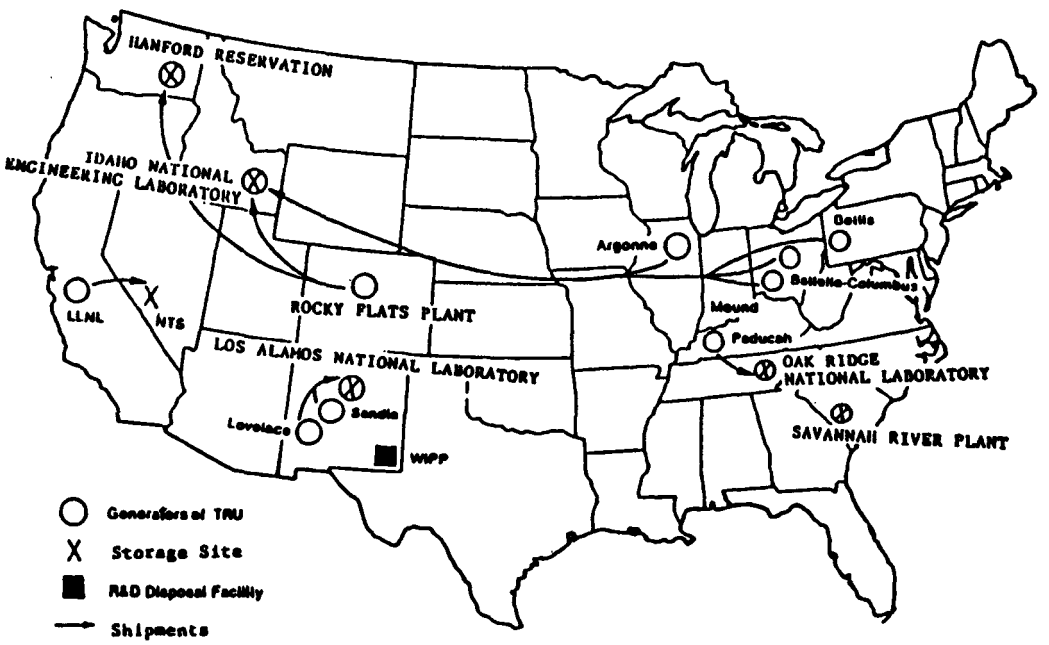

Fig. 1 Points of Origin of TRU wastes.

The NRC regulation on double containment requires that any shipments of plutonium containing over 20 curies must be shipped as a solid and must be placed in a separate inner container that restricts the loss of plutonium to the interior of the outer container to less than A_2 (2 mCi) in one week when subjected to Hypothetical Accident Conditions. Certain forms of plutonium (such as reactor fuel elements and metals) are exempt from this regulation. Other forms of plutonium may be exempt from the double containment requirement if it can be shown they are "sufficiently non-respirable". The waste being shipped to WIPP contains a variety of contaminated material forms: sludges, metal and glassware, combustible material, soil, and filters. Some of these waste forms could probably be shown to be "sufficiently non-respirable" but others would not be. The DOE considered, but rejected, a recommendation by an American National Standards Institute review committee to qualify specific non-waste forms for this exemption.

NRC regulations specifically prohibit continuous venting of packages during transport and that policy does not permit intermittent venting, even though IAEA regulations do allow this for type B(M) packages. DOE modified the original design of the TRUPACT in 1981 to include venting because of a concern that cyclical pressure changes in the storage cavity due to temperature and altitude could ultimately result in fatigue failure.

At the time the venting decision was made, the hydrogen gas generating properties of some of the CH-TRU wastes were recognized but poorly understood and there were no plans to control hydrogen concentration by recombiners or getters in either the individual waste containers or the TRUPACT cavity. It has been estimated that some shipments of waste could generate flammable or explosive concentrations of hydrogen gas within a few weeks in a sealed TRUPACT.

CONCERNS ABOUT TRUPACT-I DESIGN

The DOE position was that because of the relatively low hazard of CH-TRU wastes and because of a desire to maximize payload and minimize the number of shipments to WIPP (and thereby decrease the number of non-radiological injuries and fatalities from transportation accidents) the design met the intent of the regulations and optimized health and safety.

The New Mexico Environmental Evaluation Group (EEG), which has the responsibility of providing a scientific evaluation of the WIPP Project for the State of New Mexico, objected to the design. Reasons for the EEG objection were: (1) a design that fully met the NRC regulations would be less likely to release radioactive material in the event of a severe accident; (2) the inhalation toxicity of CH-TRU wastes shipments was not trivial; and (3) venting was not the preferred or best mechanism for controlling the hydrogen gas generation problems.

The TRUPACT-I design met all aspects of the full-scale Hypothetical Accident Conditions on the first attempt except that the filters and the door seal eventually failed following the thermal test. A redesigned door passed a subsequent thermal test. However, the drums inside the TRUPACT lost 1.26×10^{-3} of their total content to the TRUPACT storage cavity during the hypothetical accident tests (4700 A_2 for the maximum proposed load). U. S. truck accident statistics lead to a projection of as many as 12 accidents more severe than the Hypothetical Accident Conditions and an EEG prediction of 2 to 12 accidents where radioactive releases would occur. EEG concluded that a double contained design where the release to the outer cavity following Hypothetical accident conditions would be less than 1 A_2 would be much less likely to ever have a release accident.

The primary source of radiation exposure to the public in a severe accident is expected to be via the inhalation pathway and the inhalation toxicity of TRU waste that may be transported in a TRUPACT (>8800 alpha curies) could be up to 3.7 times that present in a spent fuel assembly (10-year decay time). Since the actinides in spent fuel assemblies are in a less mobile form than are most CH-TRU wastes it was concluded that the hazards from truck shipments of CH-TRU wastes were equal to or greater than truck shipments of spent fuel and needed to be treated with the same level of care.

EEG has been concerned about the gas generation problem in the TRUPACT since 1983 and has urged DOE to study gas generation and diffusion rates from various waste forms and develop programs for controlling the problem. The EEG analysis has concluded that venting of drums and the TRUPACT is not the most reliable way of controlling hydrogen concentrations and has the disadvantage of provided a weak point in the design that could fail in the event of a severe accident. It is believed that more effective hydrogen control would be provided by the use of either hydrogen recombiners or the use of propargyl ethers as hydrogen getters in drums and/or the TRUPACT.

RESOLUTION OF THE PROBLEM

The DOE finally agreed in May 1986 to design a doubly contained and non vented TRUPACT-II. The original TRUPACT-II design was also rectangular in shape and was expected to have a payload of about 5500 kg. However, DOE also made a decision to require NRC certification of the TRUPACT in June 1987 and subsequent evaluations raised several questions about whether the rectangular design could be certified by NRC without lengthy delays. The concerns were over the use of a rectangular shaped pressure vessel and some materials used in the design.

Following the decision to abandon a rectangular TRUPACT-II design, the DOE requested private manufacturers to propose a design that met specified criteria without requiring a certain shape. All proposals received used a right circular cylinder design and other details which are similar to those that have been certified by NRC for the other waste forms. The design

accepted would include 3 separate cylinderical TRUPACT-II units on a truck trailer. This design would actually carry more drums (42), have a larger payload (9500 kg) and be more economical in costs than the TRUPACT-I. The design must still be proven by analysis, full scale testing, and NRC certification.

CONCLUSIONS AND DISCUSSION

The current design of TRUPACT-II, if it can be certified by NRC, will be in all respects (number of drums carried, payload, and cost) superior to TRUPACT-I while fully meeting the safety related requirements for double containment and non venting. This suggests that the current NRC regulations are reasonable because they can be met by an appropriate design.

This saga of TRUPACT-I does raise two broader philosophical questions concerning transportation: (1) Are some present package regulations unnecessarily restrictive in that they require additional conservative assumptions that may increase transportation costs while increasing radiological safety only marginally? (2) Is it appropriate to use a criteria that optimizes both radiological and non-radiological health and safety benefits in developing transportation regulations?

In response to the first questions, the authors believe that most packaging regulations have adequate to excessive degrees of conservatism. However, it is extremely difficult to quantify the behavior of packages under severe accident conditions. Also, since releases have very negative public relations aspects, it is beneficial to the nuclear industry to attempt to eliminate <u>any</u> radioactive material releases from transportation accidents. Our bias would be toward an ALARA design. It was argued that TRUPACT-I was ALARA and that the regulations were not reasonably achievable. However, the promise of the TRUPACT-II design suggests the real problem with TRUPACT-I was the conceptual design, not the regulations.

The argument of maximizing payload in order to minimize non-radiological injuries and deaths was made by DOE in defense of the TRUPACT-I design. This argument of "trading off" radiological and non-radiological benefits has a theoretical appeal but has never been explicitly used as a basis for radiation protection standards and regulations. Furthermore, the concept if carried to its logical conclusion would drastically effect everything we do in the radiation protection area because radiological safety practices are already higher than those of most other non-radiation industries. The WIPP Project itself is estimated to lead to more deaths during construction, operation, and transportation than would be expected to occur from radiation problems if a do-nothing alternative had been chosen.

RADIOLOGICAL ASSESSMENT OF MINERAL SANDMINING IN AUSTRALIA

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INTRODUCTION

Australia is a major producer and exporter of minerals derived from sandmining operations. The most significant minerals are ilmenite, rutile, zircon and monazite. All of these contain small quantities of thorium-series and uranium-series radionuclides, with monazite, being the most active by far, containing 6-7% thorium-232.

The minerals occur naturally as small grains in beach sand. Mine sites on the West coast lie on pre-historic beach sands, several kilometres from the present coastline, and mining is typically by mobile earthmoving equipment such as scraper or bucketwheel. On the East coast, mine sites are near the coastline or on offshore islands and some of the mining is by underwater dredge. A primary concentration process at the mine site removes the debris by screening and much of the silica sand by wet gravity separation. There is little of radiological significance to this point in the process, as the concentration of radionuclides in the ore as mined is very low.

The primary concentrate, which typically may contain about 1% monazite, is transported to a secondary, or dry, separation plant, which is often remote from the mine site. It is during the secondary separation process that radiation exposures can become significant, principally from inhalation of radioactive dust and irradiation by gamma rays.

MINERAL SEPARATION

Separation of the primary concentrate into the various mineral products exploits differences in their densities and electrical and magnetic properties (Fig. 1). Differences in electrical conductivity permit some minerals to be separated in high, non-uniform electric fields; differences in magnetic susceptibility allow others to be separated in high magnetic fields. Differences in density allow both wet and dry gravity separation. All of these processes, apart from wet sedimentation, require a very dry process material for efficient separation. The feed to the separation equipment is in a spatially dilute form to allow each grain to feel the effect of the applied electric or magnetic field, or for efficient separation using gravity. As the efficiency of any one stage is rather poor, many stages may be required, and incompletely separated streams are recirculated for repeated passes, to reach an acceptable purity of mineral product.

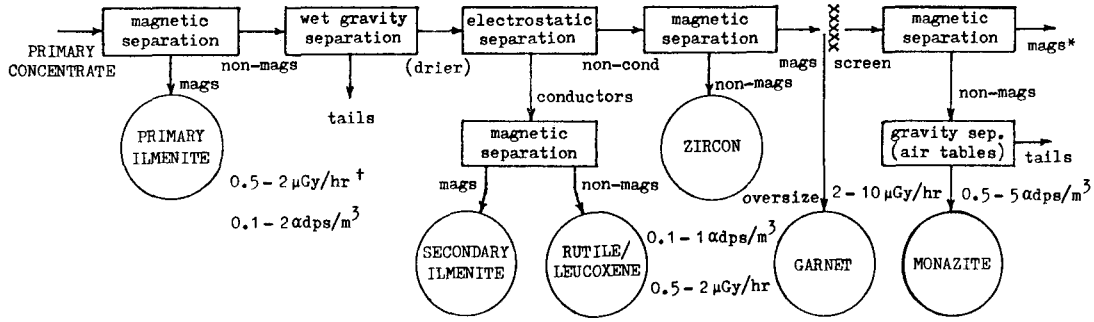
A consequence of the machine handling of hundreds of tonnes per day of mineral ores in the manner described above is that copious dusts are raised containing radioactive minerals which may be inhaled by plant workers. An additional exposure pathway arises from external radiation, especially near the monazite separation stages and in the monazite product store. A very approximate indication is given in Figure 1 of typical absorbed dose rates and gross alpha activity concentrations in air. Conditions vary from plant to plant, but where all or most of the separation processes are housed in the one building, activity concentrations can be similar throughout - a feature that is facilitated by the open interior design of many plants.

RADIOLOGICAL ASSESSMENT

The Australian Radiation Laboratory has conducted on-site measurements of absorbed dose rates, dust concentrations in air, particle size distributions, and radon and thoron and daughter concentrations at several of the major sandmining operations in Australia. A detailed report is in preparation, but some general observations are reported here, based on a representative subset of the data. The major pathways for occupational exposure have been identified as inhalation of dust containing thorium-series and uranium-series radionuclides, and external radiation, with only small contributions from radon, thoron and their daughter products.

Radon and thoron and daughters

Radon daughter concentrations are typically around 1 mWL or less at most locations, with thoron daughter concentrations in the range 1-10 mWL. Inside a monazite store containing bulk (unbagged) monazite the thoron daughter concentration near the breathing zone (1.8 m from the ground) was measured at 15 mWL. This is nearly two orders of magnitude smaller than the Derived Air Concentration (DAC) for thoron daughters of 1200 mWL [1].



*at some sites a xenotime stream may proceed from here

†the figures given are indicative only

Figure 1. Schematic illustration of a secondary separation process.

Radon concentrations up to 200 Bq m^{-3} were measured, with thoron concentrations in the latter stages of the secondary separation plant reaching 1200 Bq m^{-3}. In the store referred to above, thoron concentrations up to 5000 Bq m^{-3} were measured (one fiftieth of the DAC [1]). Ventilation is generally good, and equilibrium ratios (F-factors) are extremely low. Low thoron and thoron-daughter concentrations were not unexpected, as the short half-life of thoron does not permit significant diffusion of thoron from the mineral matrix. The thoron emanation coefficient has been estimated at about 1% or less [2].

External radiation

Absorbed dose rates increase through the separation process, being smallest, around 1 μGy hr^{-1}, in the initial stages (ilmenite section) and rising to around 10 μGy hr^{-1} or more near monazite extraction equipment and monazite bagging or binning operations. In areas where monazite product is stored, absorbed dose rates may reach 100 μGy hr^{-1} or more, depending on the exposure geometry - several times the pro-rata exposure rate permitted for a 2000-hour working year. Clearly, occupation of storage areas requires control. Access to storage areas is normally limited to storage and retrieval operations.

Employees working in secondary separation plants wear personal dosemeters. Most employees receive less than 5 mSv per year from external radiation, but some may receive up to 20 mSv in a year [3]. While the latter figure is below the current occupational limit of 50 mSv in a year, it will be necessary to review the working environment of those most exposed in the light of the recent trend towards controlling time-averaged, long-term exposures to less than 15 mSv per year [4],[5].

Radioactive dusts

Workplace dusts have been sampled at many locations in several dry separation plants. Gross alpha activity concentrations in air range from 0.02 to 50 alpha decays per second per cubic metre (αdps m^{-3}). The activity concentration values form an approximately lognormal distribution. The few measurements at the extreme upper end of the range almost certainly include material that is not normally considered as 'airborne dust' or 'inhalable' dust. A quantity of large particle size material is thrown from the separation equipment during processing. Although this rapidly reaches a vertical terminal velocity, some of it, depending on the location of the sampler, can be drawn into the sampler and retained on the filter paper. The presence of this large particle material confounds both activity concentration measurements and particle sizing studies, and consequently 'area sampling' results should be interpreted with caution. Whenever possible, personal sampling should be used to assess occupational exposure to workplace dusts.

Particle size distributions for airborne dusts have been measured using two different cascade impactors. Sizing measurements were made with a baffle placed some distance above the impactor to prevent large particle material falling vertically into the inlet. Consequently, the particle size distribution above about 20 μm is not known for samples in air. Activity Median Aerodynamic Diameters (AMADs) range from 2 to 12 μm, and an overall average value of about 6 μm, with rather large uncertainty, seems appropriate for the purposes of dose estimation. Similar values have been obtained independently [6].

There appears to be a significant difference in mean alpha activity concentrations in air between operations on the West coast (~1 αdps m^{-3}) and on the East coast (~0.1 αdps m^{-3}). The largest personal sampler concentration measured in a limited survey of workers on site was 3 αdps m^{-3}, during one shift at a West coast dry separation plant.

Taking an AMAD of 6 μm and the recommendations of the International Commission on Radiological Protection concerning calculation of DACs [7], the DAC for monazite dust based on an annual limit of 50 mSv is about 0.9 αdps m^{-3}. Many of the concentrations measured exceeded this value, especially at the West coast sites. Inhalation of radioactivity in dust during mineral sands processing is clearly a very significant exposure pathway. Fortunately, occupancy of the high activity sections of the plant is low and, for extremely dusty procedures, dust masks are worn. However, a routine monitoring program including personal dust sampling is strongly recommended in secondary separation plants, especially if adherence to a 15 mSv annual average is to be demonstrated.

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ACCUMULATION OF RADIONUCLIDES IN SEWAGE SLUDGE

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ABSTRACT

Monitoring of radioactivity in municipal waste water and sewage sludge has been common practise ever since the beginning of using radioisotopes on a broad scale in the early sixties. Accumulation of radionuclides in sludge may lead to radiation exposure of population by two major routes: incineration and use in agriculture.
While incineration is important for volatile nuclides, e.g. I 131, agricultural use is of interest for longlived radionuclides, particularly. In a research project accumulation of radionuclides in sewage sludge and elimination in waste water was investigated by simulation sewage treatment in an experimental facility on laboratory scale. Accumulation in sludge and elimination in waste water were measured for 30 radionuclides, predominantly gamma emitters. Additionally measurements were performed on samples of a municipal sewage plant where uranium effeluents from fuel fabrication were released to. According to the results of the study, elements and radionuclides may be grouped into three categories according to their accumulation behavior:

1) low accumulation factors (1000 l/kg dry weight) and elimination rates (10 %): Na, Rb, Cs, Sr, As, Sb, I
2) medium accumulation factors (3000 - 6000 l/kg dry weight) and elimination rates (50 %): Tl, Se, U, Nb, Mn, Tc, Co
3) large accumulation factors (6000 - 10000 l/kg dry weight) and elimination rates (90 %): Be, Sn, Pb, Ag, Au, Zn, Cd, Hg, Sc, Y, Ce, Gd, Cr, Fe.

After the reactor accident of Chernobyl the findings of the laboratory study were ascertained for fission products by a comprehensive monitoring program in two sewage plants and appended by an investigation of sludge incineration. It was shown that with the exception of I 131 all radionuclides were retained in ashes and filters to a large extent.

TYPICAL REMOVAL EFFICIENCIES FOR RADIOACTIVE IODINE/IODIDE FOR CHARCOAL AND SILVER ZEOLITE CARTRIDGES UTILIZED IN RADIATION PROTECTION APPLICATIONS IN NUCLEAR ENERGY INSTALLATIONS

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ABSTRACT

The removal efficiencies of radioactive methyliodide for charcoal and silver zeolite cartridges utilized worldwide in nuclear installations to determine concentration levels of radioactive iodine/iodide in work areas, plant containment atmospheres and gas storage vessels are determined as a function of flowrate, particle size, and sample duration.

A comparison of silver zeolite adsorbents and activated charcoal adsorbents (with Triethylenediamine impregnation) is made. A comparison of the pressure drop across the filter as a function of adsorbent mesh size and flowrate is made to assist in evaluating the compatibility of the adsorbent mesh size with air sampler operating specifications.

The study has been conducted utilizing the standard test method ASTM D3803, 1979 to achieve a relationship of these data to performance requirements mandated by the USNRC for nuclear grade adsorbents utilized in nuclear power plant air cleaning systems.

Equations for removal efficiencies for Methyl Iodide (CH_3I) as a function of flowrate are determined utilizing a computer program and graphically illustrated to facilitate the identification of trends and efficiency differences due to mesh size, type of adsorbent, sample duration and flowrate.

RADIOACTIVE CONTAMINATION IN THE BOLOGNA SEWAGE SYSTEM DUE TO NUCLEAR MEDICINE EXAMINATIONS

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1 - INTRODUCTION

Liquid wastes produced by the city of Bologna and surrounding areas are collected by a sewage system, leading to a depuration plant. At present, the capacity of the sewage system is about 230.000 m^3 per day, half of which is treated by the depurator.

The incoming flow is fairly constant throughout the year, except for August when, owing to factory shut-down for holidays, there is a decrease of about 60.000 m^3 per day.

The treatment of liquid waste in the depurator is in four phases:

a) primary decantation;
b) active oxidation;
c) active decantation;
d) disinfection.

After a treatment lasting 12-13 hours, "clarified" liquids are discharged into the Navile canal. This leads to the Reno river and then to the Adriatic sea.

Muds produced during decantation are further treated and reduced to ashes (within 24-48 hours) that are then stored.

Since the sewage system also collects liquid waste from two Nuclear Medicine Departements (Malpighi and Maggiore Hospitals), we decided to measure radioactivity in the liquids both at entrance and at exit from the depurator, and in the muds and ashes produced by treatment.

2 - MATERIALS AND METHODS

We gathered samples of the liquids at entrance and at exit from the depurator and of the ashes and muds.

For gamma ray spectrometry, 2 dm^3 of untreated samples were counted in Marinelli's beaker by a high-purity germanium coaxial detector (2 keV resolution at 1330 keV, relative efficency 30%) or a 3" x 3" NaI (Tl) crystal (6.5 % resolution at 662 keV). Minimum detectable activity was about 0.5 Bq/dm^3.

We did not regularly check the activity of beta-emitter radionuclides, since they are used only for "in vitro" analysis and liquid waste is not released into the sewage system.

In order to plot the destiny of a radionuclide released into the sewers, we carried out a test using a known activity of ^{99m}Tc and collected samples of the liquid at entrance to the depurator every 15 minutes. The test was done on a day when there was no waste release from the Nuclear Medicine Depts.

3 - RESULTS

The above-mentioned test, allows us to evaluate the transit time in the sewers. From the moment of release (e.g. at the hospital) and arrival at the depurator, a 3 hour delay was observed. Knowing the capacity of liquid coming into the depurator at measurement time, we observed that only a fraction $K_o = 0.30$ of the released activity (A_r) reaches the depurator.

Thus, if A_m is the activity measured at the depurator:

$$A_m = K_o \times A_r \quad .$$

In order to confirm these observations, we collected a series of samples over 24 hours.

The total weekly incoming activity, for each radionuclide, can be calculated as:

$$A_t = \sum_j \left(\sum_i A_i \times Q_i \right)_j$$

where A_i are the observed active concentrations in $Bq \cdot h/dm^3$ (fig.1) and Q_i are the capacities at the same time in dm^3/h (fig 2).

The activities, for the radionuclides found in the incoming liquids, are reported in Tab. 1 where they are compared with weekly administered activities (A_a).

Since it appears reasonable to assume that a fraction (f) of about 0.5 of the injected activity is excreted by the patients in the first few hours after administration, we can evaluate the fraction K_1,

$$K_1 = 1/f \times (A_t/A_a)$$

for ^{99m}Tc we obtained: $K_1 = 0.266$
and for 131 I: $K_1 = 0.302$

These values approximate satisfactorily with K_o ($K_o = 0.30$).

A second drop in liquid radioactivity concentrations took place during waste treatment in the depurator. This decrease is basically due to transit time of the liquid in the depurator (12-13 hours) and to mud production processes. We then introduced a second factor K_2:

$$K_2 = A_e/A_t$$

where A_e is the activity at exit to depurator released into environment. We obtained:

^{99m}Tc	$K_2 = 0.002$
131 I	$K_2 = 0.065$

The physical half life of ^{99m}Tc accounts for the difference in the two values.

Annual activities evaluated in muds and ashes are:

	Muds	Ashes
^{99m}Tc	1702 MBq	7585 MBq
131 I	3330 MBq	1665 MBq

4 - CONCLUSIONS

The model we used is based on a few simple factors that can be easily measured. The activity released into the environment through the liquid waste after treatment can be expressed by:

$$A_e = A_a \times f \times K_1 \times K_2$$

in which K_1 and K_2 are the "transfer factors" for passage in the sewage and depurator system. These factors, also include decay correction that takes into account the time needed for the two steps.

By this method, the release of radioactive meterials into the environment by the two Nuclear Medicine Depts. in our city (see tab.2 and tab.3) can be predicted and risks for the population assessed.

FIG. 1 Tc-99m LIQUID AT ENTRANCE

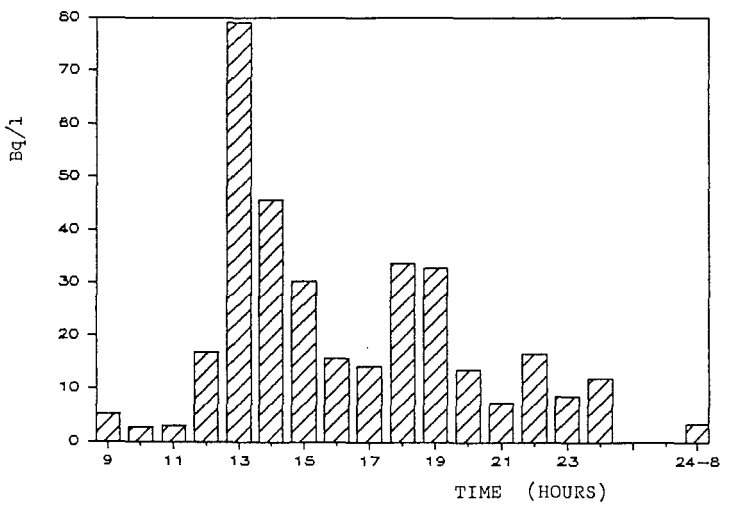
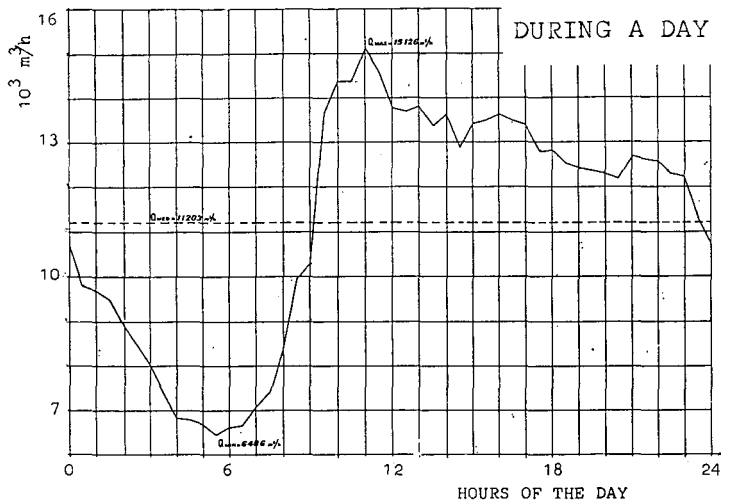


FIG.2 INPUT CAPACITY AT THE DEPURATOR DURING A DAY



TAB. 1 WEEKLY RADIOACTIVITY REPORT

		Tc-99m	I-131	Ga-67	Se-75	Tl-201	Xe-133	I-125
ADMINIS.	MBq	153439	3441	2812	11.84	2812	370	3.7
ACTIVITY	mCi	4147	93	76	0.32	76	10	0.1
LIQUID AT	MBq	20387	518	407	1.85	407	55.5	0.37
ENTRANCE	mCi	551	14	11	0.05	11	1.5	0.01

TAB. 2 ACTIVITY PER YEAR

		Tc-99m	I-131	Ga-67	Se-75	Tl-201	Xe-133	I-125
ADMINIS.	MBq	7671950	173049	140600	555	140600	18500	185
ACTIVITY	mCi	207350	4677	3800	15	3800	500	5
LIQUID AT	MBq	1020349	25271	21090	88.8	21090	2775	25.9
ENTRANCE	mCi	27577	683	570	2.4	570	75	0.7
LIQUID AT	MBq	2035	1628	1369	3.7	1369	1813	1.85
EXIT	mCi	55	44	37	0.1	37	49	0.05

TAB. 3 LIQUID RADIOACTIVITY LEVEL AT ENTRANCE

		Tc-99m	I-131	Ga-67	Se-75	Tl-201
MAX. MEASURED	Bq/l	92.5	21.8	1.2	0.5	11.1
CONCENTRATION	uCi/ml	2.5E-06	5.9E-07	3.3E-08	1.3E-08	3.0E-07
AVERAGE	Bq/l	10.7	0.3	0.2	.0	0.2
CONCENTRATION	uCi/ml	2.9E-07	7.4E-09	5.8E-09	2.6E-11	5.8E-09
MAC	Bq/l	74000.0	12.2	1221.0	3700.0	3700.0
(WATER)	uCi/ml	2.0E-03	3.3E-07	3.3E-05	1.0E-04	1.0E-04
AVG.CONC./MAC		1E-04	2E-02	2E-04	3E-07	6E-05

TAB. 3a LIQUID RADIOACTIVITY LEVEL AT EXIT

		Tc-99m	I-131	Ga-67	Se-75	Tl-201
MAX. MEASURED	Bq/l	1.6	1.1	0.9		
CONCENTRATION	uCi/ml	4.2E-08	3.1E-08	2.3E-08		
AVEREGE	Bq/l	2.1E-02	1.7E-02	1.4E-02		
CONCENTRATION	uCi/ml	5.8E-10	4.7E-10	3.9E-10		
MAC IN	Bq/l	74000.0	12.2	1221.0	3700.0	3700.0
WATER	uCi/ml	2.0E-03	3.3E-07	3.3E-05	1.0E-04	1.0E-04
AVG.CONC./MAC		3E-07	1E-03	1E-05		

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THE CONTROL OF EMISSIONS FROM NUCLEAR POWER REACTORS IN CANADA

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INTRODUCTION

Nuclear power reactors in Canada are of the CANDU pressurized heavy water design. These are located in the Provinces of Ontario, Quebec, and New Brunswick. Most of the nuclear generating capacity is in the Province of Ontario which currently has 16 commissioned reactors with a total capacity of 11,500 MW(e). There are four reactors under construction with an additional capacity of 3400 MW(e). Nuclear power currently accounts for approximately 50% of the electrical power generation of Ontario. Regulation of the reactors is a Federal Government responsibility administered by the Atomic Energy Control Board (AECB) which licenses the reactors and sets occupational and public dose limits.

DERIVED RELEASE LIMITS

Derived limits for the release of radionuclides are based on a methodology endorsed by the AECB and documented in the Canadian Standards Association Standard N288.1. Release limits are calculated such that no member of the public receives a dose in excess of the dose limit set for the public by the AECB. This dose limit is currently set at 5 mSv per year (effective dose equivalent). The methodology incorporates the environmental transfer model shown in Figure 1.

In this model the quantity of radioactive material in any compartment is given by X(i) and transfer from any compartment i to compartment j is characterized by a transfer parameter P(ij) such that the amount in compartment j arising from transfer from compartment i under steady state conditions is P(ij)X(i). Thus the quantity represented by any compartment j is given by:

$$X(j) = \sum_{i} P(ij)X(i)$$

where the summation is over all compartments transferring into compartment j.

The standard presents equations for calculating the transfer parameters P(ij) on a site specific basis, and in addition, gives default values to be used where no site specific data are available. The default values have been chosen based on data in the scientific literature and are believed to be conservative. Derived release limits are calculated separately for releases to air and surface water. They are given by the following equation:

$$DRL = \frac{\text{Annual Dose Limit}}{X(9)/X(0)}$$

where X(0) is the release rate to either atmosphere or surface water.

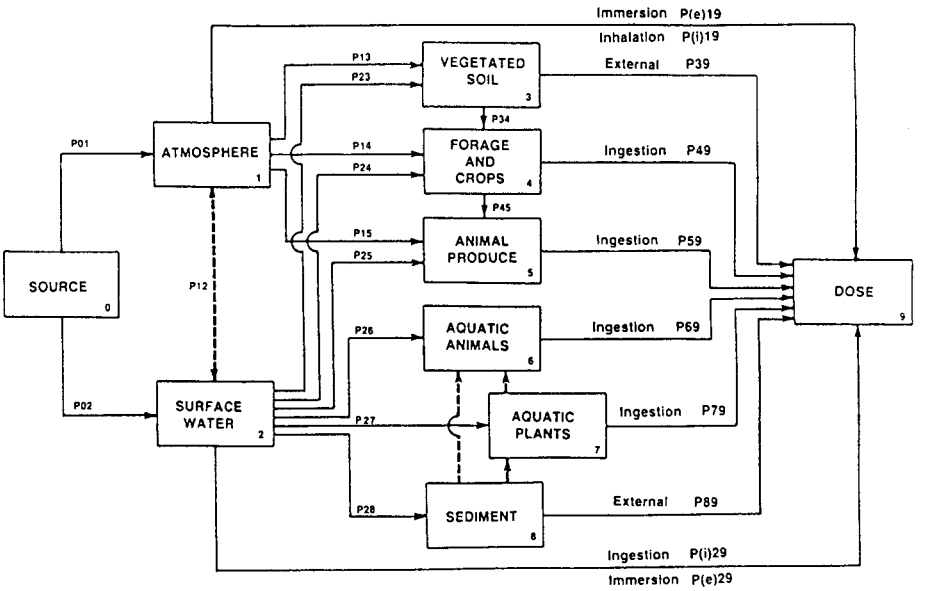

Figure 1: Environmental Transfer Model

Derived release limits are calculated for each radionuclide which contributes significantly to the source term. Limits for releases to air and surface water are calculated independently. Account must therefore be taken of multiple exposures to a critical group from all radionuclides and release sources (air and water). The dose limitation requirements will be met under the following additional condition.

$$\sum_i \sum_j \sum_k \frac{R(ijk)}{DRL(ijk)} < 1$$

where R(ijk) = actual release of radionuclide i, from effluent source j (air or water), and source facility k (takes account of a multiple facility site);

and DRL(ijk) = derived release limit for radionuclide i, effluent source j, and facility k.

These dose limits are applied to a "critical group" which represents those individuals who are expected to receive the highest dose from emissions from the facility in question. Release limits are therefore "site specific".

OPERATIONAL TARGETS

In recognition of the principle that doses should be kept as low as reasonably achievable, design criteria and operational targets are set at a small fraction (currently 1%) of the derived release limits. In practice, actual releases for most radionuclides are well below even these targets.

EFFLUENT MONITORING

Monitoring of all significant release pathways for radionuclides is conducted to ensure compliance with the DRLs and station operational targets. Monitoring may be continuous, as for releases to atmosphere, or on a batch basis, for controlled pumpouts of liquid holding tanks. The major emissions from CANDU pressurized heavy water reactors are noble gases and tritium. Carbon-14 has been shown to be significant only for Ontario Hydro's Pickering Generating Station.

ENVIRONMENTAL MONITORING

Environmental monitoring programs may be needed to achieve any of the following objectives:

(1) To validate the environmental pathway model or any transfer parameter used in the model.
(2) To provide a check on the effluent monitoring design and operating systems by comparing doses estimated from effluent monitoring with those from an environmental monitoring program.
(3) To show compliance with licensing limits where effluent monitoring is not possible or feasible.
(4) To provide direct measurements, which may be more convincing to members of the public than effluent monitoring, which would help allay public concerns and foster public confidence in the control of the licensed activity.
(5) To contribute to the preparedness for off-site emergency situations where the case so warrants.

Another standard (Canadian Standards Association Document N288.4)(in preparation) presents guidelines and criteria for determining if an environmental monitoring program is necessary in relation to a specific facility.

The proposed guidelines would require an environmental monitoring program if the sum of the committed effective dose equivalents to a typical member of a critical group from all

radionuclides and pathways from one year of operation is estimated to exceed 50 uSv.

An environmental monitoring program would not be necessary, under most conditions, if the sum of the committed effective dose equivalents to a typical member of a critical group from all radionuclides and pathways from one year of operation is estimated to be less than 5 uSv per year.

An additional criterion would call for an environmental monitoring program where the release potential from the facility is such that, in the event of an accident, the sum of the effective and committed effective dose equivalents to a typical member of a critical group might exceed 5000 uSv.

Experience from operating CANDU stations has shown that the dose to a typical member of a critical group is between one and five uSv per year. An environmental monitoring program would therefore be required under the third criterion. The programs in place concentrate on measurements of external dose from noble gases, airborne levels of tritium, and levels of tritium and carbon-14 in foodstuffs. Using Ontario Hydro's Pickering Nuclear Station as an example, the total dose to a member of a critical group during 1986 was estimated to be 36 uSv of which 20% is from tritium, 40% is from noble gases, and 40% is from carbon-14.

CONCLUSIONS

Emission and environmental monitoring data show that the dose to a critical group resulting from the operation of nuclear power plants in Canada is only a small fraction of that due to the natural background. This provides a confirmation that the release limits and targets which have been set continue to provide a satisfactory degree of protection to the public.

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DUST PARTICLE SIZE DETERMINATIONS AT AN OPEN CUT URANIUM MINE AND MILLING OPERATION

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ABSTRACT

To determine the activity median aerodynamic diameter (AMAD) and the mass median aerodynamic diameter (MMAD) of uranium ore dust and product dust. Numerous measurements were made at different stages of the mining and milling operations. Determinations of AMAD and MMAD were made in the following areas: Mine, Crushing (Primary and Secondary), Grinding, General Services areas, and the Uranium packing areas. Once AMAD's are established derived limits for dust in the air can be calculated. Inhalation of radioactive dust is the most significant pathway by which workers receive a radiation exposure.

To collect the required samples cascade impactors were used in conjunction with high volume air samplers. Samples were collected for varying periods depending on the ambient dust concentrations and work related considerations. Two methods were used to determine the activity deposited on the various cascade impactor stages, one method assessed the total long-lived alpha activity, and the other the total amount of uranium present. Results indicate that the calculated AMAD's can vary greatly depending on where the samples are collected from. It is invisaged that maybe two or three AMAD's may have to be used to derive dust concentration limits for different locations in the mine and mill.

DESIGN OF RADIOCHEMICAL LABORATORIES FOR USE IN PHARMACEUTICAL RESEARCH

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During the development of new drugs extensive use is made of labelled compounds to study their absorption, distribution and metabolism. At the Wellcome Research Laboratories, these novel compounds are synthesised in a radiochemical facility that has been incorporated into a modern purpose built medicinal chemistry building. The facility has an area of approximately $10m^2$ and consists of a large laboratory with controlled access and a separate office suite. It has been designed to safely handle up to 9.25 GBq of ^{14}C labelled compound per synthesis and to comply with the UK Ionising Radiations Regulations 1985. Five syntheses can be carried out at any one time.

The laboratory has been designed as a containment area in which the air pressure is always below ambient. Access is through an airlock entry in which there are changing and decontamination facilities. There is a separate entry to the office area which is partitioned off from the laboratory with glazed panels so that continuous observation of the activities in the laboratory is possible. Work surfaces, fume cupboards and unit furniture are of materials capable of resisting the solvent based cleaners used for decontamination. There are 2 'walk-in' safety cabinets for tall apparatus and three fume cupboards. The laboratory floor is bunded 4" below the office and adjacent area to contain any liquid spills. It is covered with a flexible 2mm heavy contact polyvinyl sheet with welded seams, which is impermeable to aqueous solutions but not all organic solvents. The walls are spray painted and finished with polyurethane lacquer to give a wipe clean finish. All joints are sealed with a smooth finish silicone mastic to avoid crevices where contaminants could accumulate.

SYSTEM OF ENTRY AND EXIT

The laboratory is entered through an airlock which has electromagnetic interlocks fitted to the doors to prevent both being open at once as the laboratory is maintained at slight negative pressure to prevent contamination of adjacent areas. Within this airlock there are electronically controlled washing facilities and hand drier so that hands may be cleaned without touching taps or towels. Changing facilities and a radiation monitor are located within this area. A step over bench is placed in the middle of the lobby. It is constructed of melamine covered chipboard and incorporates a shoe rack. In case of a failure of the interlock system or other emergency, breakglass alarm points are present which activate a local alarm/flashing light and relay the alarm state to the Building Management System.

THE WORKBENCHES

The laboratory furniture comprises cantilever framed supports with solid epoxy resin work surfaces, dished to contain any spillage. Flexible metal underbench units are suspended below to facilitate easy floor cleaning. The cupboards and drawers are painted steel for easy cleaning and in extreme situations the paint can be stripped off with the contaminant and cabinets repainted. Similarly the resin tops are impermeable to most chemicals but in exceptional cases of contamination the surfaces can be sanded down and repolished.

THE WASTE DRAINAGE SYSTEM

All sinks discharge into a borosilicate glass drainage system. For ease of maintenance the drains from each half of the laboratory pass to vertical voids on either side of the laboratory. The drainage system has been routed so that it enters the main drain of the building at the lowest practicable point. This maximises the dilution factor for radioactive residues and minimises the risk of reflux into other areas should a blockage occur in the system.

THE SAFETY CABINETS

Each fume cupboard is constructed with a double skin. The outer skin is zinc coated steel finished with an epoxy powder coating. The inner skin is a stainless steel shell produced in one piece with no seams and all curves are smoothed to prevent radioactive materials adhering to sharp edges or corners. There is a glazed top panel with anti-glare lighting giving excellent illumination of the work zone. The work top is heavy gauge stainless steel with reinforcing bars on the underside so that it supports the shielding and equipment with minimum deflection.

The sash is 6mm laminated safety glass with an epoxy coated steel frame. It is counter balanced by weights running in guides and suspended on stainless steel cables over nylon pulleys with a fail-safe device in the event of cable failure. The sash is fitted with an adjustable stop limiting the opening to the optimum height of 500mm, with a warning light on the control panel which indicates when this height has been exceeded. When closed the sash can be locked so that radioactive material can be kept secure in the fume cupboard rather than taken to store every night.

Uniform airflow characteristics at the working area are achieved by streamlining the entrance faces of the cupboard. The design incorporates an automatic by-pass to eliminate high face velocities at lower sash openings. The average face velocity is 0.75m/s at 500mm opening height.

Each fume cupboard is fitted with flow sensors and electrically operated airtight dampers. If the sensors indicate that the extraction has failed the dampers will automatically close to prevent reverse airflow through the

fume cupboard. These dampers will also close in the event of power failure so that radioactivity is contained within the cupboards and the exhaust systems. Each fume cupboard is separately ducted to roof level and thence, through filters and silencer, to the atmosphere.

All the services are controlled from the panels mounted at the side of each cabinet. The services available are electricity, water, steam and most routine gases such as nitrogen, hydrogen, oxygen and helium. Vacuum is generated for each cupboard and is independent of the general vacuum system for the building.

STORAGE OF CHEMICALS

Located between the fume cupboards are vertical steel pull out cupboards ventilated by the fume cupboard extraction system. These vertical cupboards are provided with racking and trays for storage of containers of various sizes and are mounted on pantograph drawer gears. Ventilated lockable flammable solvent storage cupboards are positioned under the fume cupboard work tops.

THE ENVIRONMENTAL CONTROL AND MONITORING SYSTEM

For safety reasons, the air supplied to the laboratories in the medicinal chemistry building cannot be recirculated. To limit the substantial costs of preheating and circulating the air, the environment is controlled by the Building Management System computer and separate air-handling units are dedicated to each laboratory. The air-handling plant is located on each floor to facilitate the distribution of ducts and to minimise the fire risk. The computer is programmed to continuously monitor the air-conditioning demands. Thus, in the radiochemical laboratory, as the number of fume cupboards in use increases or decreases, the computer balances the input air with the rate of extraction. The flow of air in the laboratory is from the ceiling plenum through the perforated steel ceiling tiles and out through the fume cupboards. Although the cupboards can be controlled manually, the computer has overall control which ensures that at all times at least two fume cupboards are operating in order to maintain the negative room pressure.

A series of sensors within the system monitor the performance of the plant and status reports are produced by the Building Management System. An immediate warning is given when plant failure causes loss of negative pressure, failure of fume cupboard extract or failure of the labelled compound cold stores. Further alarm systems monitor the gas supplies to the fume cupboards, the integrity of the fire escape door and the airlock.

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We thank Mr M Dennis and Mr A Stanley (Project Engineers) and Mr D Greenslade (Radiochemist) for their contributions.

DOSE ASSESSMENT IN RADIOACTIVE MATERIAL TRANSPORT

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As of 1961, AIEA has issued regulations for the safe transport of various modifications, principally owing to the increase in number of packages transported and to the different chemical-phisycal characteristics of the radioisotopes used in the course of the twenty years following.

This paper deals whith air, road and rail transport under normal conditions. Conseguently, the assessment of possible ionizing radiation doses received by transport personnel and by the public must be performed especially on the basis of the transport index and radiation level of the packages. The most recent pubblication of AIEA regulations on the subject dates back to 1985. As with the preceding 1973 publication, it indicates three categories of packages with reference to the afore mentioned transport index and radiation level, measured in sievert (Sv):

Category I	- White:	radiation level: $\leq$ 0.005 mSv/h
Category II	- Yellow:	radiation level: 0.005 $<$ D $\leq$ 0.5 mSv/h-transport index: 1
Category III	- Yellow:	radiation level: 0.5 $<$ D $\leq$ 2 mSv/h
		radiation level: $\leq$ 10 mSv/h for "full load"
		transport index: 10

For the first time, the 1985 publication makes esplicit and complete reference to the norms of radiation protection issued by AIEA itself in collaboration with OIT, Who and the NEA of OCDE (Basic Safety Standards for Radiation Protection, 1982, Ed. Safety Series No. 9). In these norms, AIEA establishes for trasport workers a dose level of 5 mSv per year. The aforementioned radiation protection norms shall serve as reference for the administrative and organizational aspects with regards to remaining within these dose levels. These norms must, of course, take into account the particular conditions under wich they are applied. As far as nuclear installation are concerned, or the use of radioisotopes or order ionizing radiation sources, the problem of assessing and cheking the doses received can be performed using parameter which can be established and maintained at least within certan limits. The case of transport is completely different because the only objectively known fact, as far as we are concerned, is the previously mentioned irradiatione of packages. It is a particulary difficult task, however, to shift from the latter to evaluating possible doses because this involves a large number of parameters which cannot always be known and properly evaluated (i.e. doses received at work stations and time spent at them; number of packages per

irradiation category; distances from packages belonging to different categories; division of initial loads over different routes via different transport means, etc.). There is presently a nationwide drive towards special regulation setting load limits and the distance of workers from the loads so as to limit the absorption of possible doses (Cfr. Italy, Regolamento per il Trasporto di Merci Pericolose per Ferrovia - tr.n. Regulation for Transport of Dangerous Goods via Rail). Other methods are utilized experimentally to evaluate the doses absorbed. We find that all these methods, however, are based on the irradiation of packages and not on methods, however, are based on the irradiation of pakcages and not on dose measurements.

In the behalf that the assessment of doses from packages irradiation is somewhat difficult and consequently involves a high degree af approximation, we have elaborated a form which we propose with this paper. our proposed solution practically overturns the problem, and thus may appear oversimplified. We in any case welcome any semplification if it can prove useful to improving or, at least, facilitating the evaluation of the possible dose absorbed by transporters of radiactive material. This proposal choses to ignore the external packages irradiation, of course accepting those established by AIEA, in favour of directly measuring the doses in question. Form no. 1, which refers to drivers and escorts, records the real or possible dose measured at the onset of each transport in the worker's work station; of the various values recorded, we chose the one which guarantees the greatest precautions, which is to say the highest one measured. This value, multiplied by the time duration of the trip, gives the probably absorbed dose value per trip. Hence, it is easy for the supervisor to calculate the annual dose or to take the proper measures if he foresees that a worker might surpass the maximum dose admitted (5 mSv/y). An assessment performed in this manner doubtlessly safeguards workers if accompanied by very strict orders, such as not sleeping in the cabin during possible stops.

The problem for loading and unloading is more complex, given the nature of the operations themselves. However, for these tasks too, there is a form (no. 2). It must record the dose value measured in the trolley driving station. This measurements must be performed by loading the packages taken as examples inthe worst condition with respect to package irradiation, for each trolley used and not for each transport. The dose values measured shall be multiplied by the time duration of the operations, and thus we obtain the values of the presumed dose absorption by the workers involved in these operations. These values are so precautionary as to make additional doses owing to manual operation negligible. We must bear in mind, in this regard, that the percentage of III-Yellow packages (dose up to 2 mSv/h transported is generally rather modest. In 1984, 1985 and 1986, in Italy this percentage averaged approxipately 7-8% of the total number of packages transported containing radioactive material. This estimate refers to a significant sample (about 50% of total packages). With regards to the public, it is especially important to know the main traffic flow routes and those which transit through inhabitated areas or areas frequented by the public. Here it is sufficient to make evaluations which take into account the measurements taken autside the means of transport and of the time the public is exposed to them at various distances.

Air transport merits special consideration. Here passengers (the public) could absorb unreasonable and rather high doses if they are frequently present in aircraft transporting packages containg radioactive material. These situations require a careful examination of the main traffic flow routes. Aircraft personnel can be considered as on a par with the workers concerned by form no. 1,and thus use the same form.

The considerations made above, especially as concerns forms no. 1 and 2, may lead to modifications in packages irradiatio, to be proposed to AIEA, or to special equipment on the means of transport.

It is significant that the 1985 AIEA publication established for transport workers a dose level of 5 mSv per year, for members of the public a dose level of no more than 1 mSv per year, whith explicit reference to the aforementioned "Basic Safety Standards". This makes possible to perform assessments using the aforementioned forms, effecting any adjustment deemed proper or necessary for each individual carrier, on the transports and loads in question.

Form. no. 1 (*)

Worker name .....
Year .....

Measured dose (mSv/h) (**)	Date (+)	Trip Duration (Hours)	Total Dose (mSv/trip) (++)
.....			

Total Annual Dose ('').....

(*) For transport workers, drivers or escort, or aircraft staff.
(**) Refers to the dose measured (see text) in the worker's work station, real or possible, for each individual transport. The dose must be measured at transport onser, choosing the highest value measured.
(+) The date must be that of departure.
(++) Obtained by multiplying the value of the measured dose by the time duration of the trip.
('') Must not be higher than 5 mSv.

Form. no. 2 (*)

Worker name .....
Year .....
Dose in Sv/h measured in the trolley driving station having a supposed load arranged in the conditions to produce the highest dose value, on the basis of packages transported by the firm or board which employs the worker (see text).....

Date	Time employed in Hours for each movement	Dose in Sv calculated for each movement (**)
.....		

Total Annual Dose (+) .....

(*) For workers involved in loading and unloading operations
(**) Obtained by multiplying the value of the dose measured by the time employed for each movement.
(+) It must not be higher than 5 mSv.

RECONSTUCTING FALLOUT EXPOSURES TO THE U.S. POPULATION FROM WEAPONS TESTING IN NEVADA DURING THE 1950'S

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ABSTRACT

For the past several years, the EML has been participating in several major studies designed to reconstruct the radiation exposure received by persons living downwind from the Nevada Test Site (NTS) as a result of weapons tests carried out during the 1950's. This reconstruction has utilized a number of novel techniques designed to infer fallout based on retrospective measurements, as well as re-evaluations of monitoring data from the period of testing. Fallout in states near the NTS has been inferred from measurements of Cs-137 and Pu isotopes in contemporary soil samples from undisturbed sites. A re-evaluation of monitoring data taken using gummed-film at about 100 sites throughout the U.S.A. has provided reasonable estimates of fallout for sites in the central and eastern U.S. where total depositions were too low to be determined by the retrospective soil analysis method. Analysis of sediment layers from lakes and reservoirs downwind from the NTS have provided an independent time history of fallout and plutonium isotopic mixture for some areas. Contemporary gamma-spectrometric data obtained during aerial surveys for uranium ore have been used to estimate topsoil Cs-137 activity in order to screen for potential NTS fallout hotspots.

Cumulative NTS fallout deposition and resulting external exposures for the entire U.S.A. have been calculated and the particular shots contributing most of the fallout at a given site have been identified. These depositions are being utilized by a number of groups to estimate population doses utilizing state of the art pathway models. The reconstruction methods used in these studies have application to assessing exposure from other types of man-made fission product contamination, including fallout from a reactor accident such as occurred at Chernobyl in 1986.

BIKINI ATOLL IONIZING RADIATION SURVEY--MAY 1985-MAY 1986

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INTRODUCTION

Between 1946 and 1958, the United States conducted 23 nuclear tests at the Bikini Atoll in the Marshall Islands. The single largest detonation was the "Bravo" test, which resulted in extensive radioactive contamination of a number of islands and prevented the timely resettlement of the native population. Since 1958, many studies have been conducted to assess cleanup options and the internal and external radiation doses the Bikinians would likely receive, should they resettle the islands.

Although the external dose rates from β and γ radiation have been previously determined by aerial and ground measurement techniques, technical constraints limited the assessment of external β dose rates from the Cs-137 and Sr-90/Y-90 contamination on the islands. Now, because of the recent development of very thin thermoluminescent dosimeters (TLDs), these external β dose rates can be measured.

THE SURVEY

The purpose of this survey was to (1) determine the β dose rate at 7 mg/cm^2 and the deep dose rate (γ) on the two habitable islands in the atoll, Bikini and Eneu; (2) compare the dose rates at heights of 1 cm, 50 cm, and 100 cm; and (3) determine the effect of various ground covers on the β dose rate.

We conducted this survey in two 6-month phases, and based results on data from 800 Panasonic-802 dosimeters. Each dosimeter contains 4 elements (E1, E2, E3, and E4); E1 and E2 are $Li_2B_4O_7$:Cu and E3 and E4 are $CaSO_4$:Tm (hereafter referred to as Li or Ca). These dosimeters are normally exposed in their holders, which contain absorbers, but in this experiment, half of the dosimeters were exposed out of their holders so that both the Li and Ca chips measured the β radiation. We sealed each dosimeter in a Saran bag that was lined with aluminized mylar: the Saran (2 mg/cm^2) provided moisture protection and the reflective aluminized mylar (1 mg/cm^2) minimized heat buildup. Except where the bags were exposed to intense reflected light or got buried in the ground, this packaging configuration was successful. Bagged Panasonic dosimeters were placed between aluminum supports that were stapled to wood blocks. A layer of plastic tape was placed over the top and sides of the dosimeters to protect them from direct rain and sunlight.

We distributed the dosimeters among 102 monitoring sites, each consisting of 1 out-of-holder (O) and 1 in-holder (I) dosimeter at heights of 1 cm, 50 cm, and 100 cm; 11 β spectrometer arrays, used to assess the maximum and average energy of the β radiation; and 6 fade study stations, used to assess environmentally induced fading of the Panasonic dosimeters. Each β spectrometer array

consisted of five out-of-holder dosimeters at heights of 1, 50, and 100 cm. At each level, one dosimeter was left bare, while the other four were covered with aluminum absorbers so that total absorber thicknesses were 14, 21, 48, 84, and 233 mg/cm^2, respectively. Each fade study site consisted of two out-of-holder dosimeters, sandwiched between thick aluminum absorbers and mounted in a holder equidistant from a 10 μCi Cs-137 source. Since any β response was eliminated by the aluminum absorbers, and both Li and Ca respond linearly to the 662 keV photons from Cs-137, any fading of the Li relative to the Ca would be evident by comparing the measured doses at the end of the experiment. Three fade study locations were selected, representing the full range of thermal environments: one in a house protected from direct rain and sunlight, one in a breezy, semi-shaded area, and one in the middle of the island with intense sunlight and little breeze.

We selected monitoring sites based on the Marshallese life style, giving emphasis to areas where people would likely spend the most time. To assess the effect of ground cover, many of the areas monitored included two adjacent sites, one cleared of plants and debris, and one left uncleared. We also evaluated the effect of coral gravel as a ground cover by placing a 1-m-radius pad of it in two highly contaminated areas on Bikini and putting a monitoring station in the center of each. Nearby, we established stations over cleared and uncleared soil.

Another area specifically evaluated was the Excavation Plot, an experimental garden established in the most contaminated area of Bikini. All plants and the top 40 cm of soil had been removed from this 2-acre plot, where different crops were then grown. The Control Plot, equal in size and adjacent to the Excavation Plot, was also stripped of plants and used as an experimental garden, but the topsoil was left essentially undisturbed. A 90-foot-wide Buffer Zone, left in its natural condition, separated the Excavation and Control Plots.

DATA ANALYSIS:

The dosimeters were shipped to Bikini in a lead box and the control dosimeters indicated the transportation dose was statistically indistinguishable from background. Therefore, we made no specific correction for transportation dose.

The cosmic ray background of 3.3 μR/hr had been previously assessed by various experimenters and data from Eneu corroborate this value. We subtracted a background of 3.3 μR/hr (29 mrem/yr) from the dose rates reported in this survey.

Li data from the six fade study sites were within two standard deviations of the respective Ca data, so we did not apply any fade correction to the Li data.

Because Ca significantly overresponds to low-energy photons, we compared data from E3(I) (Ca) and E2(I) (Li), both of which are covered with absorbers that attenuate only 4% of 30 keV photons. We found the dose on E3(I) exceeded that on E2(I) by more than 3

standard deviations only 3.9% of the time, indicating that low-energy photons made an insignificant contribution to the total radiation dose. Therefore, we did not make a correction for Ca overresponse.

We also compared E1(O) and E2(O) (Li) to E3(O) and E4(O) (Ca) and found that in 85% of the cases, the average of the dose on the Li chips fell within three standard deviations of the average on the Ca chips. In 7% of the cases, the Li chips read higher than the Ca, and in 8% of the cases, the Li chips read lower. After extensive evaluation, we found that Li TLDs were adversely affected by moisture, heat, and light, and had limited accuracy at low doses. Ca TLDs did not have these limitations, and since there was no low-energy Ca overresponse to contend with, we used only the Ca data (E3 and E4) to calculate dose rates.

Analysis of the β spectrometer arrays indicated that the average energy of the β spectrum was somewhere between that of Y-90 and Tl-204. At 7 mg/cm^2, the efficiency of the Ca TLDs to Y-90 was 85%, and to Tl-204 was 72%. We chose a calibration factor of 79% and then determined the β dose rates at 7 mg/cm^2 by averaging the E3(O) and E4(O) data, subtracting the corresponding E4(I) data, and dividing by 0.79.

We used E4(I) data to assess the exposure rate in air from penetrating γ radiation, and Kerr's conversion factor for the testes (0.75 rads in tissue/R in air, at 662 keV) to convert to dose in tissue.

Published conversion factors for photon dose to the skin range from 0.685 to 0.78 rads in tissue/R in air. We chose to use 0.75 because it was conservative and the same conversion factor used to calculate dose in the testes. Shallow dose rates were then calculated by adding the photon dose rate to the skin to the respective β dose rate.

After measuring the precision and accuracy of the dosimeters, we calculated a total experimental error of ±15% on the raw data, and propagated the errors to report the 95% confidence interval of the reported dose rates. We assumed that the background value of 3.3 μR/hr and the conversion of 0.75 rad in tissue/R in air were constants.

DISCUSSION OF RESULTS

The mean deep dose rate on Eneu was approximately 18 mrem/yr, and the mean β dose rate varied from 23 mrem/yr 1 cm off the ground to 6 mrem/yr 100 cm off the ground. The highest β dose rate measured on the island was 90 mrem/yr at 1 cm, and 42 mrem/yr at 100 cm. The highest measured deep dose rate was 88 mrem/yr, but there were only three areas on the island where the measured deep dose rate exceeded 30 mrem/yr. Natural ground cover had no measurable effect on the dose rates.

Bikini's radiation profile was more complicated than Eneu's. Since there were many unique areas to be evaluated, data from Bikini Island was divided into subgroups. In general, the highest

β dose rate measured in each subgroup was 1.5-2.5 times the mean, and the highest deep dose rate was 1.5-2 times the mean. Exceptions to this generalization existed where the dose rates varied little between sites.

IN HOUSES: The mean deep dose rate in the houses was about 37 mrem/yr. In a house with concrete made with reef aggregate, no β radiation was detected, but in the houses made from island aggregate, β radiation averaged 116 mrem/yr at 1 cm, and 46 mrem/yr at 1 m.

AROUND HOUSES: The mean deep dose rate around houses was about 110 mrem/yr. The mean β dose rate ranged from 301 mrem/yr at 1 cm to 165 mrem/year at 100 cm.

GENERAL AREAS: The mean deep dose rate was about 200 mrem/yr, and the mean β dose rate ranged from 550 mrem/yr at 1 cm to 192 mrem/yr at 100 cm. These values are higher than the true island average, though, since a disproportionate number of sites were selected in highly contaminated areas.

EXCAVATION EXPERIMENT: Both the β and deep dose rates varied greatly in the Buffer Zone and the Control Plot, probably as a result of soil disturbances that occurred during excavation and planting. In these areas, deep doses ranged from about 100 mrem/yr to 536 mrem/yr, and β dose rates ranged from 173 to 1886 mrem/yr at 1 cm, and from 100 to 500 mrem/yr at 1 m.

The dose rates in the Excavation Plot were consistently low: the mean deep dose rate was 47 mrem/yr at 1 m, and the mean β dose rate was 88 mrem/yr at 1 cm and 54 mrem/yr at 100 cm. Removing the top 40 cm of soil reduced the β dose rate between 80% and 94% at 1 cm, and between 72% and 87% at 1 m.

VARIATION OF DOSE RATE WITH HEIGHT: In heavily contaminated areas, the mean β dose rate at 1 cm was about 2.5 times the respective mean deep dose rate; at 50 cm, it was 1.5 times the mean deep dose rate, and at 100 cm, it about equaled the mean deep dose rate. In lightly contaminated areas the β dose rates more closely paralleled the deep dose rates at all heights.

EFFECT OF GROUND COVER ON β DOSE RATES: A comparison of data from cleared and uncleared areas showed that ground cover did not consistently enhance or degrade the β dose rates. Presumably, the large variations in the β dose rate that existed within small geographical areas overwhelmed the small botanical differences we were trying to measure.

Coral gravel reduced the 1 cm β dose rate from 1,015 to 110 mrem/yr (89%) in one area, and from 346 to 79 mrem/yr (77%) in the other.

A full report on this survey, including raw data and references, can be obtained from the author by requesting report UCRL-553798.

RADIOACTIVE CONTAMINATION AT MARALINGA.

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Australian Radiation Laboratory.

Atomic weapons tests were conducted in Australia at three locations between 1952 and 1963. There were twelve trials where nuclear explosions took place (major trials), and many other trials where devices and components associated with the development of atomic weapons were burnt or exploded with conventional explosives (minor trials).

Radioactive residues from the major trials have not proved to be of radiological consequence in the long term, and it has been estimated that remaining activation and fission products will cease to be of any concern beyond about 2030 (Lokan, 1985).

Between 1960 and 1963, however, a series of safety trials involving plutonium were carried out at the Taranaki site at Maralinga, where twelve firings were conducted in three series of trials from firing pads within a few hundred metres of each other, leading to the local dispersal of some 22 kilogram of plutonium-239. Shallow burial pits were constructed close to each firing pad, and much of the material was ultimately buried in these pits. The remainder of the plutonium was carried in wind-borne plumes, and was generally deposited down range to the north of the firing pads. The distribution of plutonium on the ground, for the first several kilometres, was reconstructed at the end of the series (Turner, 1964) from measurements taken at the time, and indicated the presence of four plumes, made up of overlays of individual firings, extending several kilometres from the firing pads.

Between 1984 and 1986 the Australian Radiation Laboratory conducted extensive field investigations to map the extent and distribution of these plutonium plumes. The distribution of americium-241, which is a decay product of plutonium-241 present in the original material, was determined from its 60 keV gamma ray using portable thin-crystal sodium iodide scintillation detectors. In addition soil samples were collected to a depth of 2.5 cm from some 400 sites within 1 kilometre of the firing sites, and a further 350 samples from sites between 1 and 100 kilometres. These were analysed with the aid of high-resolution gamma ray spectroscopy, to determine the concentration of americium-241 in Bq/kg in surface soil. The results of these analyses are presented in Figure 1. Independent studies of the ratio of americium-241 to plutonium-239 (Tracy, 1987) for Taranaki soil samples indicate a ratio of about 1 : 8.4, and imply surface concentrations along the plume axes, which are an order of magnitude greater than was originally inferred.

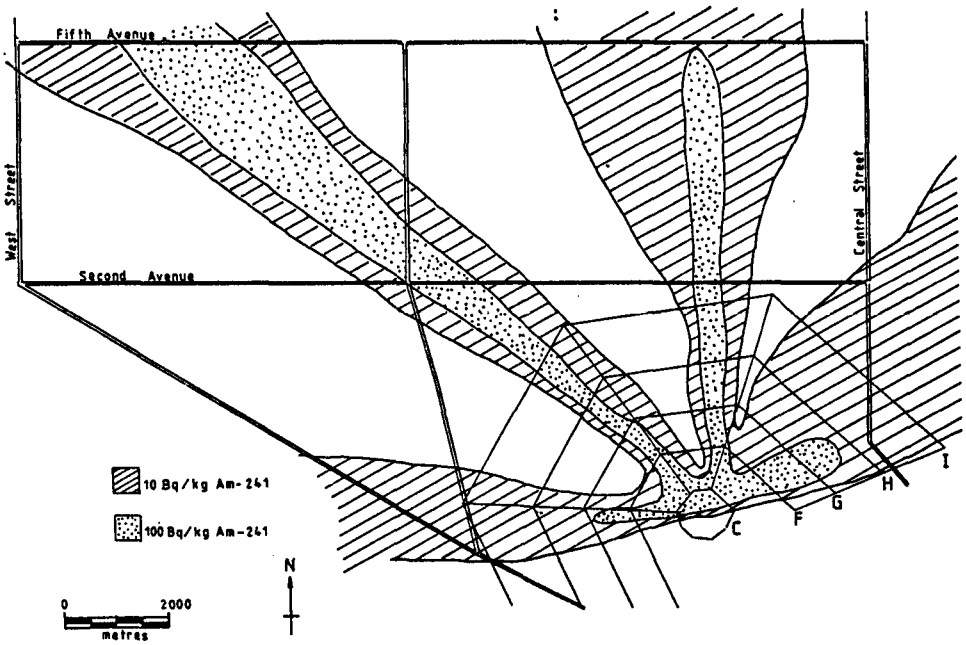

Figure 1: Am-241 contamination levels based on soil samples measured in 1984-1986.

The main feature of the plumes is that they are quite narrow, and contrary to an expectation that surface contamination would gradually be dispersed in this sandy semi-arid environment, they appear to have undergone little observable change in the past twentyfive years. In addition, the evidence to date suggests that there has been little downward migration of plutonium, as depth profiles have indicated that most (90 per cent) of the activity still lies in the top centimetre of soil.

In order to obtain further detail about the character of the dispersed plutonium, eight sites within the four identified plumes were selected for further study. Composite soil samples from each site, which covered an area of 100 x 100 m, were analysed to determine the average concentration and uniformity of the soil contamination. Mean concentrations in soil, expressed in terms of americium-241, at distances of 1.5 and 5 km from the firing pads are presented in Table 1.

Table 1.

Mean Concentrations in Plumes (Bq/g of Americium-241)

Distance from Firing Pad (km)	West	North West	North	North East
1.5	0.35	0.52	1.07	0.75
5	0.01	0.71	0.17	0.10

The values in Table 1 represent the averages of about fifty soil samples drawn from each site, and individual samples commonly showed sample to sample variations in excess of an order of magnitude. For the the site in the north plume at a distance of 1.5km from the firing pads concentrations varied from 0.04 to 4.2 Bq/g of americium-241. Some of the more active samples from this and other sites were successively subdivided, and it was shown that individual particles of diameters between 100 and 200 micron accounted for most of the activity in these cases.

The distribution of activity as a function of particle size, which is of great importance to the future assessment of potential inhalation hazard, was investigated at each site by separating the soil into different size fractions with the aid of a set of graded sieves. Average mass and activity distributions are presented in Table 2, which indicate that in general the "fine fraction" (< 45 micron) , which represents only three per cent of the mass, contains about thirty per cent of the activity.

Table 2.

Distribution of Mass and Activity in Taranaki Soil.

Size Range (micron)	Percentage of Mass	Percentage of Activity
1000 - 710	2	0.3
710 - 500	12	3
500 - 250	37	12
250 - 125	24	11
125 - 75	14	32
75 - 45	8	13
< 45	3	29

The "fine fraction" was investigated further for samples from the 1.5 km site on the North West plume, using a micro particle classifier to obtain mass and activity distributions as a function of size. These results, which are preliminary at this stage, and may not be generally representative, are presented in Table 3. The size ranges given are "optical" diameters, rather than aerodynamic diameters, as the classifier was calibrated by comparison with visual assessment of sizes under a microscope. It should be noted that the size ranges are approximate only, as the cut-off sizes for the particle classifier are density dependent, and density variations in the soil have not been taken into account.

Table 3.

Activity and Mass Distributions for Soil below 45 Micron.

Size Range (micron)	Percentage Mass	Percentage Activity	Specific Activity (Bq/g Am-241)
< 3	2	1	1.6
3 - 4	2	1	1.5
4 - 6	11	7	1.8
6 - 12	17	15	2.5
12 - 17	18	15	2.4
17 - 25	19	14	2.2
25 - 45	31	47 *	4.4

Overall americium-241 concentration : 2.9 Bq/g

* May be due to a single high activity particle.

Table 3 demonstrates nevertheless that for this site at least, the activity concentration does not vary sharply with particle size within the respirable range - say up to 6 micron in diameter, corresponding to an aerodynamic diameter of about 10 micron - but stays relatively constant at 1.5 to 1.8 Bq/g of americium-241 or 12 - 15 Bq/g of plutonium-239.

This work is part of a broader study of the plutonium environment at Taranaki, which is addressing the inhalation pathway, and requires much more information about the physicial and chemical nature of the plutonium, dust loadings for both normal and atypical conditions and potential occupancy before any realistic dose assessment can be undertaken.

It is a pleasure to acknowledge the technical assistance of Myra Wilks in the preparation and sizing of the soil samples.

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FIELD STUDIES TO DETERMINE ACCEPTABLE LEVELS OF CONTAMINATION AT FORMER UK NUCLEAR TESTING SITES, MARALINGA AND EMU IN AUSTRALIA

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INTRODUCTION

The Maralinga and Emu regions of South Australia were used between 1953 and 1961 for the UK nuclear weapon development program. Two types of trials were conducted - the major trials involved the detonation of fission weapons and the minor trials dealt with weapon design and operational safety.

The minor trials led to substantial environmental degradation having uncertain consequences since they involved the widespread dispersion of plutonium, beryllium and natural uranium. It is not possible from historic records and contemporary measurements to account for all the material involved in the minor trials. In the case of plutonium, for example, 22 kg was initially dispersed: somewhat less than 2 kg is identifiable as fragments or smaller particulate on or near the surface to a distance of ~ 4 km from ground zero or along well-defined plume deposition zones; some is buried in a series of concrete-capped disposal pits; and the remainder is dispersed, presumably at low deposition concentrations, over a large area.

In 1986, as a result of the findings of the Royal Commission on Nuclear Testing in Australia[2], the UK and Australian Governments agreed to set up a Technical Assessment Group (TAG) with one American, two Australian and two British members to review the Maralinga-Emu situation. TAG was to advise on a series of clean-up options and their associated costs and examine land-use options ranging from unrestricted use by the traditional Aboriginal owners to options involving various degrees of administrative and physical control.

In its interim report, presented in May 1986, TAG observed that the existing data base was inadequate and suggested a series of field and laboratory studies that would partly correct this situation. Six of these studies were concerned with redefining the existing levels of contamination and establishing acceptable levels of contamination for a range of land-use options. This paper discusses the rationale, organisational support, scope and experimental protocol adopted for each of the six studies.

EXTENT OF SURFACE CONTAMINATION

Previously staff of ARL[1] had contoured the levels of ^{239}Pu contamination (by measuring the associated ^{241}Am radioactivity) using hand-held monitors and radiochemical determinations of the ^{239}Pu:^{241}Am ratios. In the main this work described the situation out to open contours equivalent, for example, to about 2 Bq g^{-1} of ^{239}Pu at a distance of 600 m from ground zero at the Taranaki site. ARL also identified several plumes of Pu contamination originating from Taranaki; the north-west plume, was followed to a distance of 32 km where the contamination level was 20 m Bq g^{-1} of ^{239}Pu[1].

TAG proposed an aerial survey followed by a vehicular survey in order to more fully map the contamination levels. The contract for the former was let to EG&G through the US Department of Energy. The Royal Air Force supplied the two Wessex helicopters. In all, 223 on-line survey flying hours were completed for a survey distance of 39 000 km over an area of about 1 500 km^2. The advanced data reduction programs used by EG&G achieved detection limits that were better than expected and significantly reduced the scope needed for the land survey which is to follow.

LAND USE

It is anticipated that the most restrictive land use will revolve around those Aboriginal groups which have claims on the Maralinga and Emu lands. Some rural Australian Aboriginals have, over the last twenty years, reverted to a more traditional life-style. This is due largely to changing social attitudes amongst Aboriginals and non-Aboriginals and has been made possible by political change and the relaxation of legislative control. The change is evident in the formation of 'outstations' where the traditional social relationships have been re-established. These groups derive some of their food supply from 'bush-tucker', particularly for those food items that were traditional favourites (e.g. kangaroo, some reptiles and grubs).

The anthropological study will concentrate on quantifying the life-style of an out-station community living at Oak Valley (100 km north-west of Maralinga) to determine the nature and importance of the various exposure routes. Key elements of this study include dietary composition, methods for food preparation, practices that enhance the intake of soil and fire-ash through inhalation and ingestion, exploitation of the various land systems and the likely occupancy of the Maralinga and Emu lands when these are returned to them.

EXPOSURE ROUTES

INHALATION

Conventional wisdom suggests that inhalation of ^{239}Pu-^{241}Am is the most restrictive exposure route. The studies deal with five aspects of the problem.

Between 1957 and 1966, dust hazes, local dust storms and regional dust storms were recorded. The yearly frequency and mean associated wind speed (m s^{-1}) were - dust haze : 1.6, 11.6; local dust : 1.6, 12.2; dust storms : 1.2, 12.7[3].

To record the air concentration of Pu/Am that will arise from such events, eight battery-operated/solar-charged continuous air samplers have been installed. These low volume samplers are cut-in and cut-out by a built-in anemometer set 7 and 6 m s^{-1} respectively.

To emulate dust-raising activities, a series of human activities (shovelling, digging etc) will be carried out up-wind of a high volume air sampler. A dust-raising unit (large fan, wind chambers, high volume air sampler) will also be run at several of the locations to acquire information on particle size and specific activity of wind-suspended contamination.

Personal air samplers are used to monitor exposure of Aboriginals to dust. The experiments are of one of three classes. Certain activities, e.g. digging for grubs and rabbits, earth-oven preparation, children's games, riding the back of a pick-up truck along bush tracks, will be simulated working in contaminated land. Other activities e.g. sleeping on the ground, ceremonial performances, being down-wind from large 'comfort' brush fires will be monitored with fixed samplers in uncontaminated areas. The third category will involve the use of Aboriginal volunteers wearing the samplers in uncontaminated areas while undertaking activities judged by the anthropologists as likely to lead to the highest dust loadings.

The samplers for the second and third categories will record only the dust loading (respirable dust in mg h^{-1} of the physical activity). The results will be converted to radioactivity inhaled, based on the ratio of particle size distribution in the uncontaminated areas to the specific activity/particle size distribution in the contaminated areas.

INGESTION

Sampling is concentrated on those bush food items found to be important by the anthropologists. Observations suggest that substantial amounts of soil and fire-ash are ingested with the food, be it of bush tucker or supermarket origin. The quantity of contaminated soil and ash consumed will be determined by preparing the food in the Aboriginal way.

WOUND CONTAMINATION

As a result of their life-style and the harshness of the environment, Aboriginals receive more cuts and abrasions than would non-Aboriginals. In the Oak Valley community, ceremonial cutting is no longer practised, but this could change over time. The potential for significant wound contamination is enhanced by the traditional treatment which can involve packing the wound with mud.

The importance of wound contamination depends markedly on two factors - the specific activity of the contamination and the metabolic fate of such discrete sources which find their way into cuts. The first is being addressed in the land survey and the second by the bioavailability studies.

BIOAVAILABILITY

The basic position is that where the ICRP has recommended values for the dosimetric parameters, they become 'default' values and the purpose of the experimental bioavailability studies is to ensure, where possible, that the default values are sufficiently conservative. The significance of this policy is detailed below for each of the exposure routes.

Inhalation

At least initially both the ICRP[5] and NVO[4] lung dosimetry models will be used. Among the more important parameters in these models are: the percentage translocated from the upper respiratory tract to the gut, the distribution of the contamination between the three classes of lung solubility (D, W,Y) and the percentages translocated to the liver, kidney, bone, thoracic and abdominal lymph nodes. The experimental program is aimed at all but the first of these parameters.

In vitro solubility in lung fluid and a few lung retention measurements on material of intermediate specific activity will provide the bridge between data obtained using high specific activity in the animal work and those for material drawn from areas having an acceptable level of contamination.

Ingestion

The primary dosimetric factor is that for the gut transfer (f_1). The ICRP recommends[6] that 'on taking account of the many factors that have been shown to influence the absorption of plutonium compounds in animal species (maturity, dietary deficiencies, fasting, chemical form), a value of 10^{-3} for f_1 is considered cautious and unlikely to be exceeded by a significant amount in any critical group of adults'. Further, it is stated that this value 'gives a sufficient margin of safety for radiation protection purposes in all situations where the intake cannot be described precisely'. From the discussion of limits for occupational workers 'precisely' can be interpreted as where 'the chemical and physical state of the ingested material can be confidently established'.

In a quasi-technical sense, there could be a range of views on the adequacy of an experimental f_1 value for soil/ash relative to, say, Pu citrate. If we assume that these are in the same ratio of animal results on Pu oxide/Pu citrate would this constitute a 'precise' description of the soil/ash contamination in terms of the ICRP recommendations?

An overriding problem with respect to f_1 values for Aboriginals living traditionally is the importance of consumption habits (fasting) and nutrient status. The anthropologists are attempting a partial quantification of these factors.

Wound Contamination

The dosimetric strategy to be followed with respect to wound contamination is to equate the annual risk coefficient for mortality to that equivalent to 1 mSv y^{-1}. The probabilities covered are: a wound being inflicted, the wound becoming contaminated, translocation of contamination from the wound site, organ distribution of the translocated contamination, and the risk coefficient for the wound site and other exposed organs.

CONCLUDING REMARKS

The field studies outlined above are derived from ideas contributed by TAG members and the individual study leaders. They take account of several unique features:

- the lack of even phenomenological descriptions for the events that caused the Pu/Am contamination;
- an arid environment;
- land uses that include Aboriginals following a fairly traditional life-style;
- uncertainties on how the Aboriginal culture and land-use will change over time;
- the more than usual amount of uncertainty that surrounds the dosimetric factors and how the recommended values may change over time.

The quality of our predictions can only be judged when we have got most of the field data. At the present rate of progress this is expected to be in 1989.

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TRANSFER OF IODINE-131 FROM DEPOSITION-TO-MILK : ESTIMATION OF PASTURE INTAKE

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For most individuals, milk consumption is the most important environmental pathway for the transfer of Iodine-131 from fallout to humans (Eisenbud and Wrenn 1963, Garner and Russell 1966). In assessments of radiological transport of I-131 from fallout deposition to cow's milk, knowledge of the fraction of the dairy cow's diet that is due to fresh pasture is essential because it is the only portion of the feed that may be contaminated to a substantial extent. For studies involving past fallout events covering large geographic areas, such as the current effort by the National Cancer Institute to assess the exposure to I-131 received by the American people during the Nevada Test Site atmospheric weapons tests conducted during the 1950's, it is necessary to derive this estimate of pasture consumption from past records. These estimates will be utilized in a model described in a companion paper presented in this symposium (Bouville et al. 1988).

Information on current pasture feeding practices cannot be used for the 1950's because the trend toward larger farms and greater daily food intake requirements by high milk producing cows has greatly diminished the use of pasture feeding on dairy farms (McCullough 1981). Today, in many cases, pasture feeding has been replaced by drylot feeding (Ward and Whicker 1987), which utilizes little or no pasture grazing.

In the United States, the only nationwide standardized information source for data on the diet fed to dairy cows is provided by the Dairy Herd Improvement Association (DHIA). Farmers participating in this program report the weight of the cows, the milk production, the fat content in the milk, the amount of concentrates, succulent and dry forages fed to the cows, and the number of days the cows were put on pasture to graze. The DHIA summarizes the data in yearly herd averages and calculates both the estimated net energy in the diet from the different feeds and the ratio of the amount of energy fed to the amount required by the cows. The yearly herd averages, starting in 1953 for most states, have been stored in computer form.

From these data, the average daily total net energy intake by cows can be calculated and the fraction of the energy intake that is from pasture can be then converted to mass of pasture consumed per day. In order to be consistent with data values derived by the DHIA during the 1950's and 1960's, the methodology presented uses the same assumptions and relationships that were used at that time (Wadell 1986 pers. comm.). Current methods employ slightly different assumptions. Therefore, the results calculated using today's methods (NRC 1978; McCullough 1981; Etgen and Reaves 1978) would be slightly different from those presented here. The steps of the calculation are shown in equations 1-6, below.

The net energy intake, NE (Mcal), needed by lactating cows is the sum of the net energies needed for maintenance of the body, NE_m, and milk production, NE_p:

$$NE = NE_m + NE_p. \quad (1)$$

The net energy needed by lactating cows for maintenance of the body is a function of the average body weight, BWT, of the cow:

$$NE_m = [0.012 \times BWT] + 0.83, \quad (2)$$

where NE_m is expressed in Mcal and BWT in kg. The net energy needed to produce milk, Mcal, depends on the percentage of fat in the milk, FATPC, and on the daily milk yield, MY, in kg. It is calculated by:

$$NE_p = [(0.09 \times FATPC) + 0.35] \times MY. \quad (3)$$

The daily milk yield is determined by dividing the reported total yearly milk production by the number of days in the year.

The actual amount of energy fed to the cow is usually greater than the calculated net energy intake requirement, in order to account for the additional energy expended during the normal activity of the animal. The feed index, FI, defined by the DHIA as the ratio of energy that is actually fed to the cow to the calculated requirement, ranges from 1.05 to 1.15 in the Northeast and is used to estimate the total net energy fed, NE_{fed} (Mcal), to the cow:

$$NE_{fed} = NE \times FI. \quad (4)$$

The yearly aver$e of the daily pasture intake, PI_y, in kg (dry matter), is estimated as:

$$PI_y = [NE_{fed} \times NEPAST] \times CF, \quad (5)$$

where NEPAST is an estimate of the fraction of the cow's net energy intake that is from pasture in a given year, and CF is a conversion factor from Mcal to kg (dry matter). For the purpose of this study, an average value of 1.47 Mcal kg^{-1} has been adopted for CF on the basis of the relevant data for the major pasture types (NRC 1978).

The annual averages of pasture intake, for each of the northeastern states, were calculated using the method described above with the computerized data available from 1953 to 1964. The results showed a significant increase in pasture intake after 1958. It is unlikely that the pasture portion of the diet increased during the late 1950's because the use of pasture feeding was declining during that time (Ward and Whicker 1987). However, it has been speculated that this increase could be due to changes in data collection or tabulation that may have occurred when the DHIA records were computerized in 1959 (Wadell 1986 pers.comm.). Since it is unknown if the estimates reported before or after 1958 are more representative, the average and standard deviation of the complete data set are being used.

Using the 26,800 records available for the state of Pennsylvania, from 1953 to 1964, as an example, it was found that the average DHIA cow weighed 683 kg and produced about 15 kg of 4.0% butterfat milk per day. Virtually all of the farmers reporting to the DHIA put their cows out to pasture during the year. The total daily net energy intake averaged over the whole year, NE_{fed}, was calculated as 19.1 Mcal d^{-1}; the value of NEPAST, the net energy from pasture, was 14.9%. Converting the net energy intake to mass of intake, it was found that the daily total dry matter intake was 13.4 kg d^{-1}, including the daily pasture intake averaged over a year, PI_y, of about 2 kg d^{-1}. The total dry matter intake, 13.4 kg d^{-1} falls within the 10 to 20 kg d^{-1} range of

reported values for dry matter intake of dairy cows (Morrison 1961; Koranda 1965; NRC 1978; CES 1979; Ward and Whicker 1987; Leaver 1985).

The pasture intake usually occurred only during a defined pasture season each year so the length of the pasture season must be used to estimate the average daily intake during the time of grazing. The pasture season length was estimated, in each state, from the reported DHIA values, USDA Extension Dairy Specialists advice, and other sources. The average daily pasture intake during the pasture season, PI_s, in kg, is calculated by:

$$PI_s = PI_y \times (DY/PD), \quad (6)$$

where DY is the number of days in a year and PD is the number of pasture days.

Daily intake of pasture is not constant during the pasture season. Information on the monthly variations of feeding practices in each state was also obtained from the USDA Extension Dairy Specialists. The preliminary estimates of daily pasture intake by month for Pennsylvania are shown in Fig. 1. As is illustrated, the cows were out on pasture by the beginning of May through most of October, for an estimated 176 days. The peak pasture season occurs during the spring and early summer, and pasture intake drops during the heat of the summer as a result of the decrease in pasture quality. During the peak of the pasture season in Pennsylvania, the pasture intake is estimated to be about 50% of the cow's total dry matter intake or about 7 kg d^{-1}.

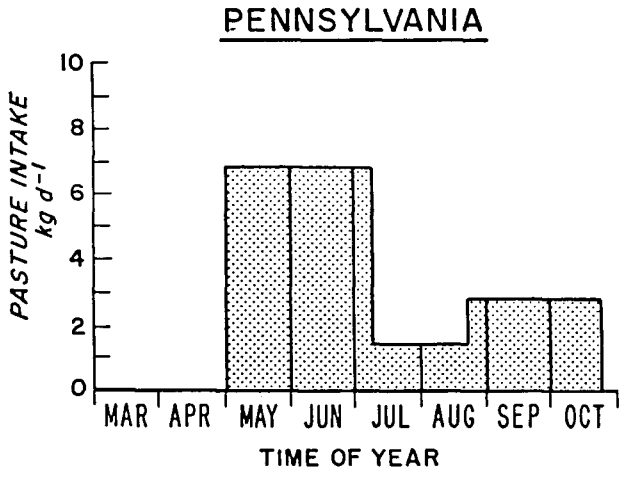

Figure 1. Preliminary estimates of pasture intake of dairy cows in Pennsylvania, USA during the 1950's and early 1960's.

Unfortunately, the percentage of net energy from pasture is no longer reported by the DHIA, so a direct comparison cannot be made between past and present pasture practices. A decrease in pasture utilization is however apparent from the number of farmers using dry lot feeding and the shorter pasture season reported by the farmers that put their cows on pasture. Virtually all of the farmers during the 1950's and 1960's reported some pasture feeding, but only about 75 percent of the 6100 records collected in Pennsylvania during 1985 reported days on pasture for their herds. For these farmers, the pasture season averaged 107 days a year, which is about two months shorter than the average value reported for the 1950's.

Total dry matter intake also varied between past and present practices. For example, in Pennsylvania during 1985, the average DHIA cow was smaller (544 kg vs 683 kg) and produced more milk per day (23.4 kg vs 15 kg), with a lower butterfat content (3.7% vs 4.0%), than a cow 2 to 3 decades earlier. Using current methods of calculation (NRC 1978) and the 1985 average values presented above, an estimate of the total daily dry matter intake of 15.8 kg d^{-1} is calculated. This shows that even though pasture intake may be providing a smaller part of the total diet of dairy cows today, the total daily intake has increased.

Acknowledgements - The authors would like to thank: H.L.Beck, Environmental Measurements Laboratory; J.Till, Radiation Assessments Corporation; G.Ward, F.W.Whicker, Colorado State University; L.Wadell, Cornell University; G.Wiggans, C.Ernst, Animal Improvement Programs Laboratory, Maryland; K.Butcher, G.Griffin, North Carolina State University.

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ASSESSMENT OF IODINE-131 TRANSFER TO COW'S MILK AND TO MAN RESULTING FROM THE NEVADA WEAPONS TESTS OF THE 1950'S

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Public law 97-414 directs the United States Secretary of Health and Human Services to "conduct scientific research and prepare analyses necessary to develop valid and credible assessments of the exposure to I-131 that the American people received from the Nevada atmospheric bomb tests". The National Cancer Institute (NCI) was requested to respond to this mandate. In so doing, a task group, established to assist the NCI in this effort, suggested that it might be possible to estimate, for each of the most important tests, the I-131 exposures from fallout for representative individuals in and for the populations of each county of the contiguous U.S.

The main pathway to man from fallout I-131 is, for most individuals, via the grass-cow-milk chain. In order to assess I-131 exposures for persons in each county of the contiguous U.S. the following estimates must be made:

* the activities of I-131 deposited on soil and pasture grass,
* the resulting I-131 concentrations in cows' milk, and,
* the quantity of I-131 ingested by man.

The most significant atmospheric weapons tests with respect to fallout occurred in the 1950's, during which time most of the monitoring of environmental radioactivity consisted of gross beta or gamma measurements. Assessments of exposures due to I-131 can be inferred from the original measurements of gross beta or gamma activity, from current or past measurements of radionuclides other than I-131, or from mathematical models.

ESTIMATION OF DEPOSITION DENSITIES

Re-analysis of historical monitoring data and meteorological modeling are two complementary methods which we use to estimate the I-131 that was deposited on the ground across the U.S. following each test. For both approaches, the assumption is made that the I-131 was in particulate form.

Review and re-analysis of historical monitoring data

Monitoring of long-range fallout deposition in the U.S. in the 1950's was carried out primarily by the Environmental Measurements Laboratory (EML) in cooperation with the U.S. Weather Service (Beck 1984; Beck et al. 1988; Harley et al. 1960). The EML deposition network across the U.S. evolved gradually from 10 locations with trays of water in 1951 to 93 locations with gummed-paper collectors in 1952, and finally to about 100 locations at which gummed-film collectors were used. Usually two films were exposed during a 24-h period beginning at 1230 GMT. The geographical coverage of the gummed-film network was relatively homogeneous over the contiguous U.S. The samples collected were ashed and counted for total beta activity.

Beck (1984) reviewed and re-analyzed the available gummed-film data that could be found in the HASL/EML archives, together with other, less extensive fallout data, in order to derive depositions of Cs-137, I-131, and I-133. The resulting data set includes daily depositions of I-131 at about 100 locations in the U.S. during most of the atmospheric testing period. Those I-131 depositions are associated with information on the amount of precipitation measured during the same 24-h periods. Some of the gummed-film results have been reported (Beck 1984, 1988). The deposition densities in counties without monitoring stations are estimated with interpolation techniques that take into account the daily amounts of precipitation.

Meteorological modeling

The radioactive cloud that is formed after an atmospheric detonation near the ground surface usually extends from the ground surface to the highest layers of the troposphere and occasionally reaches into the stratosphere. The dispersion of the radioactive cloud has been analyzed for each important atmospheric test using routine maps, provided twice a day by weather services, depicting airflow at constant pressure levels. These maps were used to construct 6-h trajectories of air parcels moving across the U.S. at several altitudes. The amounts of I-131 contained in the various segments of the radioactive cloud are assumed to remain confined between the altitudes of adjacent trajectories unless they are partially removed by wet precipitation processes. The calculation of the fraction of I-131 that is removed from the cloud segment by precipitation is carried out with an empirical method based on the use of relationships between the column content of I-131 in the overhead cloud and the I-131 deposition estimated from gummed-film data.

This approach is used primarily to validate the results obtained by interpolating the monitoring data. A detailed description of the meteorological modeling can be found in Hoecker and Machta (1988).

ESTIMATION OF THE I-131 CONCENTRATION IN FRESH COW'S MILK

The time-integrated concentration in milk (IC) corresponding to a unit deposition density on the ground (DG) on a given day and in a given county is estimated as:

$$IC = DG \cdot F/Y \cdot T_{eff} \cdot PI \cdot F_m \qquad (1)$$

in which F/Y is the mass interception coefficient [m^2 kg^{-1}(dry weight)], T_{eff} is the effective half-time of retention by the vegetation (days), PI is the pasture intake [kg(dry weight) d^{-1}], and F_m is the intake-to-milk transfer coefficient (d L^{-1}). These four parameters are discussed in turn.

Interception coefficient

The fraction of I-131 radioactivity intercepted by vegetation (F), or the interception coefficient, is estimated to vary as a function of the standing crop biomass (Y) according to the relationship proposed by Chamberlain (1970):

$$F = 1 - e^{-\alpha Y} \qquad (2)$$

where the foliar interception constant (α) represents the influence of all other, less important, parameters.

On the basis of numerous experiments with small-sized aerosols, the numerical value of α, for dry deposition conditions and long-range fallout, is taken to be equal to 2.8 m^2 kg^{-1}(dry weight); for short- and intermediate-range fallout, account is taken of the variation of α with particle size and therefore with distance from the detonation site. Simon (1987) estimates that the variation of α [m^2 kg^{-1}(dry weight)] as a function of distance D (km) can be expressed as:

$$\alpha(D) = 7.01\ 10^{-4}\ D^{1.1268}. \qquad (3)$$

In the eastern part of the country, most of the fallout occurred via wet deposition processes. There is little information on the influence of the amount or intensity of precipitation on the value of α. On the basis of a literature review by Voilleque (1986), the variation of α as a function of the rainfall amount P (mm), for $P > 5$ mm, is estimated by:

$$\alpha = [E + S/P]\ Y \qquad (4)$$

where E = 1.3 m^2 kg^{-1}(dry weight) and S = 16 mm kg^{-1}(dry weight) m^{-2}.

Effective half-time of retention

After I-131 is deposited on vegetation, environmental removal processes (T_w) combine with radioactive decay (T_r) to reduce the quantity of initial contamination on the vegetation surface . An effective half-time (T_{eff}) can be calculated as:

$$T_{eff} = (T_w \cdot T_r) / (T_w + T_r) \qquad (5)$$

where T_r is the radioactive half-life of I-131. Given the short radioactive half-life of I-131, the effective half-life T_{eff} is not very sensitive to large variations of the environmental half-time T_w. If the value of T_w is assumed to be 14 d, T_{eff} is found to be equal to about 5 d.

Pasture intake by dairy cows

The way in which the pasture intake by dairy cows is estimated is described in detail in another paper in this Congress (Dreicer et al. 1988). The daily pasture intake, averaged over the year, is calculated as:

$$PI = CF \cdot NE \cdot FI \cdot NEPAST \qquad (6)$$

where PI is the pasture intake, expressed in kg d^{-1}(dry weight),
NE is the net energy requirement of the cow (Mcal),
FI is the feed index, defined as the ratio of the net energy fed to the cow and of its net energy requirement,
NEPAST is the fraction of the net energy fed to the cow that is derived from pasture, and
CF is the conversion factor from energy to mass, taken to be 1.47 Mcal kg^{-1}(dry weight).

All of the elements necessary to estimate yearly averages of the daily pasture intake are available from dairy records for a large number of herds and for each year of the atmospheric testing period. For the purpose of this study, state averages for the entire testing period have been used in the determination of the daily pasture intake, averaged over the year.

Given the short half-life of I-131, it is necessary to determine the temporal variation of the pasture intake throughout the grazing season. Estimates of the beginning and the end of the pasture season and of the fraction of the total diet obtained from pasture each month were acquired for each of the 48 states considered, from U.S. Department of Agriculture Extension Specialists and other sources. Examples of results are given in Dreicer et al. (1988).

Intake-to-milk transfer coefficient

The intake-to-milk transfer coefficient F_m (d L^{-1}) is the time-integrated concentration in milk per unit activity intake of I-131. Reported values of F_m for iodine secreted into milk range from 2 10^{-3} to 4 10^{-2} d L^{-1} (Hoffman 1979). A value of 8.1 10^{-3} d L^{-1} has been adopted in this study; it is assumed to be independent of any influencing parameter.

ESTIMATION OF I-131 INTAKE BY MAN

The final step of the assessment consists of estimating how much milk was produced and where it was consumed. To this end, information was gathered on the milk production in each county, on the milk distribution pattern in each state, on the delay between production and consumption of milk, and on the consumption of milk as a function of factors such as race, age, and sex.

Time-integrated I-131 concentration in consumed milk

The amount of milk available for human consumption is estimated for each county and apportioned among the following four categories for which different delay times between milking and consumption are assumed:

* The milk consumed on farm (delay: 1 d)
* The milk sold for fluid use:
    (a) produced in the same county (delay: 2 d)
    (b) produced in the same region (delay: 3 d)
    (c) produced in another region (delay: 4 d)

In order to model the local distribution of this milk, each state is subdivided into groups of counties (called "regions") determined either by the available marketing data, by major population areas, or by state topography. The existence of a surplus or deficit of milk in each county is determined from the milk consumption needs of the population of the county at that time. The surpluses and deficits of milk are first balanced within the established regions by distributing the excess milk from the surplus counties of the region to the deficit counties of that region. If there still is a surplus (or deficit) after this step, the milk is assumed to be shipped to (or from) another region. The flow of milk is determined on the basis of the available marketing statistics, and the advice of experts.

The calculation of the I-131 concentrations in consumed milk take into account the possible mixing of milk from various origins as well as of the radioactive decay which occurs between milking and consumption.

Activity intake by man

The collective intake of I-131 is represented by the total activity of I-131 contained in the four categories of milk discussed above. Individual intakes of I-131 are much more difficult to assess, as they imply that the category of milk consumed by that individual is identified, and that the milk consumption rate of that individual, which depends upon many factors such as sex, age, race and degree of urbanization, is known. For that reason, only typical individual intakes are estimated for each county of the contiguous United States and for each atmospheric nuclear weapons test which resulted in significant fallout.

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RADIOACTIVITE NATURELLE ET ARTIFICIELLE DU MOLLUSQUE *Tridacna maxima* EN POLYNESIE FRANCAISE. INTERET DOSIMETRIQUE

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I - INTRODUCTION

En 1987, des mesures de ^{210}Pb et ^{210}Po ont été effectuées sur le bénitier (*Tridacna maxima*) prélevé sur différents sites de Polynésie, dans le cadre de la surveillance biologique de l'environnement, que la France exerce depuis 1963. Les mesures portent essentiellement sur deux fractions du mollusque: la chair (consommée) et les viscères. Par ailleurs, en deux lieux précis: PAPARA (district de TAHITI) et TAKAPOTO (atoll des TUAMOTU) trente bénitiers ont été récoltés, chaque bénitier donnant lieu à deux mesures du seul ^{210}Po, une sur la chair une autre sur les viscères.

A titre de comparaison nous montrons l'évolution depuis 1966 de la teneur en ^{60}Co de la chair du même mollusque en deux lieux: TAHITI (marché de PAPEETE) et TUREIA atoll situé à 80 km au nord de MURUROA. Le ^{60}Co est l'un des radioéléments artificiels que le bénitier concentre le plus efficacement.

II - METHODE

2.1. - Mesure du ^{210}Pb.

Elle s'effectue par spectrométrie γ, sur détecteur GeHP, des échantillons lyophilisés. Selon la taille des bénitiers, l'échantillon représente de 20 à 200 individus. La limite de détection est voisine de 5 Bq/kg Frais.

2.2. - Mesure du ^{210}Po.

Elle s'effectue par spectrométrie α après minéralisation acide des échantillons lyophilisés et électrodéposition sur disque en argent. Le ^{208}Po est utilisé comme traceur. La limite de détection est de l'ordre de 0,04 Bq/kg Frais.

2.3. - Mesure du ^{60}Co.

Le ^{60}Co est mesuré par spectrométrie γ, sur détecteur NaI(Tl) des échantillons incinérés.

III - RESULTATS

3.1. - Les tableaux I, II et III rassemblent l'essentiel des résultats acquis. En ce qui concerne le tableau I où la comparaison entre les teneurs en plomb et en polonium peut être menée, les valeurs en ^{210}Po ont été ramenées au jour du prélèvement en tenant compte d'une part de la décroissance radioactive du ^{210}Po et de la descendance du ^{210}Pb d'autre part. Par contre les tableaux II et III rassemblent des valeurs non corrigées mais homogènes entre elles car chaque série de mesures fut effectuée en une quinzaine de jours.

3.2. - La figure 1 montre l'évolution des moyennes annuelles des teneurs en ^{60}Co de la chair de bénitier à TAHITI et TUREIA.

IV - DISCUSSION

4.1. - Quelques constatations.

4.1.1.- Que ce soit dans la chair ou dans les viscères, le

plomb et le polonium ne sont pas en équilibre: le ^{210}Po est presque toujours en excès.

4.1.2.- Que ce soit pour le plomb ou pour le polonium, les viscères présentent presque toujours les teneurs les plus fortes.

4.1.3.- Il n'y a pas de corrélation entre la taille des bénitiers (donc leur âge) et les concentrations en plomb et polonium (activités spécifiques). Par contre - et en conséquence - les corrélations sont nettes entre la taille et les teneurs globales (activités spécifiques x poids frais).

4.1.4.- Bien que la variabilité des teneurs soit grande pour un même site (tableaux II et III) le paramètre essentiel semble être le lieu de prélèvement: plus d'un facteur 10 entre les échantillons de RANGIROA et ceux de TAKAPOTO. C'est du moins ce que laissent paraître les résultats au premier abord, car en fait cette variation spatiale peut être aussi due à des cycles temporels de grande ampleur.

4.1.5.- Pour ce qui concerne l'évolution de la teneur en ^{60}Co, on note une décroissance régulière à partir d'un maximum situé dans les deux années suivant le début des expérimentations en 1966. La variabilité des résultats peut provenir de réinjections de radioactivité artificielle avant 1974, fin des essais aériens, et de fluctuations d'échantillonnage après.

4.2. - Intérêt dosimétrique.

La consommation de bénitiers varie en fonction du lieu géographique, et à l'issue d'enquêtes alimentaires, les organismes chargés de la surveillance radiologique de la Polynésie Française (2) ont adopté les valeurs suivantes, qui correspondent au maximum d'une fourchette pouvant varier de 1 à 10:

PAPEETE	3,29 kg/an
TUREIA (TUAMOTU)	36,50 kg/an

En considérant ces valeurs et les activités moyennes en ^{210}Pb (12 Bq/kg) et en ^{210}Po (41 Bq/kg) figurant dans le tableau I, et la somme des moyennes annuelles présentées sur la figure 1, les évaluations suivantes sont obtenues:

LIEU	**DOSE EFFICACE**			
	^{210}Pb (µSv/an)	^{210}Po (µSv/an)	Total (µSv/an)	^{60}Co µSv cumulés 66-86
PAPEETE	54	59	113	0,2
TUAMOTU (TUREIA)	596	652	1248	19

Inférieures à la dose maximale admissible pour le public -5000 µSv/an- ces doses peuvent se comparer aux 180 µSv/an dus au ^{40}K.

Dans le cas particulier des TUAMOTU les doses atteintes sont du même ordre de grandeur (1000 µSv/an) que celles rapportées par BOUVILLE (1) concernant certaines zones arctiques et subarctiques.

La participation de ^{60}Co à la dose efficace délivrée aux populations à travers la consommation de bénitier depuis le début des essais nucléaires en Polynésie Française, est négligeable devant la dose efficace due à la radioactivité naturelle.

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Lieu	Taille	Teneur en Pb-210 (Bq/kgF)		Teneur en Po-210 (Bq/kgF)	
		chair PbC	viscères PbV	chair PoC	viscères PoV
ANAA	Gros=3	4,8±4,7	240±6	20±2	252±16
MOOREA	Gros=3	8,7±4,5	286±5	44±3	417±26
MOOREA	Moyen=2	12±4	186±5	42±3	424±26
MOOREA	Petit=1	16±5	184±6	81±7	466±29
RANGIROA	Gros=3	13±4	458±5	106±7	493±31
RANGIROA	Gros=3	27±4	382±5	25±2	696±45
TAKAPOTO	Gros=3	<4,9	54±4	4,5±02	67±5
TAKAPOTO	Moyen=2	<4	35±4	5,6±03	107±8
TAKAPOTO	Petit=1	<6	26±4	7,1±04	93±6
TEAHUPOO	Gros=3	11±5	174±6	45±3	255±17
TEAHUPOO	Moyen=2	14±5	112±4	62±4	294±18
TEAHUPOO	Petit=1	16±5	72±4	82±5	319±16
TUBUAI	?	6,2±4,1	75±7	45±2	89±7
VAIRAO	Gros=3	<5,9	142±4	25±2	233±15
VAIRAO	Moyen=2	5,8±4,1	126±5	23±1	234±15
VAIRAO	Petit=1	7,1±4,7	106±5	33±2	293±18
Moyenne		12	166	41	296
Ecart type		6	123	29	172
Effectif		12	16	16	16

TABLEAU I

Teneurs en Pb-210 et Po-210 des bénitiers en différents sites.

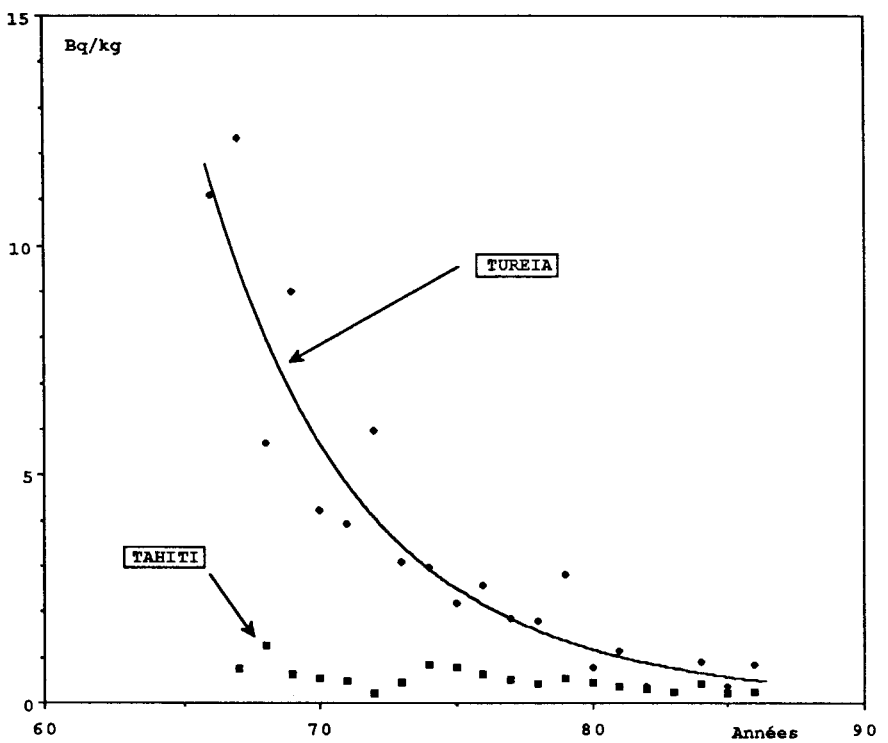

Figure 1: Co-60 dans les bénitiers .
(un point = moyenne de plusieurs valeurs annuelles)

Taille (mm)	CHAIR		VISCERES	
	Poids Frais (g)	Teneur Po-210 (Bq/kgF)	Poids Frais (g)	Teneur Po-210 (Bq/kgF)
69	6	2,1 ± 0,3	5,2	50 ± 6
69	5,8	4,2 ± 0,5	5,9	77 ± 6
69	6,6	4,3 ± 0,4	7,2	76 ± 5
69	6,1	4,3 ± 0,5	7,1	54 ± 5
69	4,7	3,6 ± 0,4	5,1	56 ± 6
70	4,2	4,3 ± 0,5	5,2	54 ± 5
70	5,2	2,3 ± 0,3	5,6	71 ± 6
70	5,6	1,9 ± 0,3	5,4	66 ± 5
70	8,7	2,1 ± 0,3	7,5	37 ± 6
70	7	2,7 ± 0,3	5,6	83 ± 10
87	9,3	2,8 ± 0,3	10,7	72 ± 7
88	12,5	4,9 ± 0,4	10,2	89 ± 7
89	8,4	2,5 ± 0,3	11	87 ± 6
90	9,8	4,9 ± 0,5	10,7	64 ± 6
90	11,4	2,2 ± 0,2	12,6	68 ± 5
90	11	1,5 ± 0,2	11,9	71 ± 6
92	9,5	7,9 ± 0,6	16,8	60 ± 5
93	15,2	5,4 ± 0,4	10,6	109 ± 7
94	11,8	2,3 ± 0,3	8,7	95 ± 7
95	13,1	4,8 ± 0,4	20	63 ± 4
106	17,8	106 ± 0,2	17,5	67 ± 5
107	15,9	2,1 ± 0,2	14,3	86 ± 7
108	13,4	1,9 ± 0,2	18,5	53 ± 4
108	16	3,1 ± 0,3	21,8	94 ± 7
109	15,7	3,0 ± 0,2	25,1	41 ± 3
109	11,2	2,7 ± 0,3	21	69 ± 5
110	14,9	3,3 ± ,03	23,8	89 ± 7
110	17,7	2,7 ± 0,2	22,3	69 ± 5
110	16,5	4,3 ± 0,3	16,4	61 ± 5
111	18	2,5 ± 0,2	19,4	67 ± 4
MOYENNE		3,3		71
ECART TYPE		1,4		17
EFFECTIF		30		30

TABLEAU III

Caractéristiques et teneurs en Po-210 des bénitiers de Takapoto.

Taille (mm)	CHAIR		VISCERES	
	Poids Frais (g)	Teneur Po-210 (Bq/kgF)	Poids Frais (g)	Teneur Po-210 (Bq/kgF)
66	6,4	35 ± 2	3,5	541 ± 36
67	7,2	59 ± 4	5,3	213 ± 15
70	5,9	74 ± 5	7,1	509 ± 33
70	6,7	31 ± 2	3,9	149 ± 12
70	7,1	46 ± 3	5,6	269 ± 19
70	6,1	39 ± 3	7,5	178 ± 15
70	6	30 ± 2	5,9	131 ± 11
70	6,5	39 ± 3	5,8	287 ± 24
70	5,6	54 ± 4	9,7	445 ± 29
70	5,6	39 ± 3	6,3	230 ± 22
85	11,8	16 ± 1	15,6	256 ± 16
90	11,2	19 ± 1	11,3	361 ± 23
91	11,9	127 ± 8	13,6	271 ± 17
91	12,3	88 ± 5	13,4	275 ± 18
92	15,5	45 ± 3	11,5	220 ± 14
95	12,6	45 ± 3	9,5	452 ± 29
95	8,9	27 ± 2	18	296 ± 20
95	10,3	50 ± 3	10,3	263 ± 17
96	10,6	39 ± 2	9	385 ± 25
96	10,7	48 ± 3	12,5	237 ± 17
108	29,9	34 ± 2	28,1	343 ± 21
112	30,6	31 ± 2	24,6	288 ± 18
118	26,7	72 ± 4	19,2	256 ± 16
119	43,4	94 ± 6	29,9	175 ± 11
120	29,2	87 ± 5	33,3	301 ± 19
120	23,6	95 ± 6	26,9	164 ± 10
121	25,8	33 ± 2	28,8	353 ± 23
127	30,9	75 ± 5	29	355 ± 22
130	36,2	42 ± 3	24,5	369 ± 23
131	24,8	28 ± 2	48,4	286 ± 19
MOYENNE		51		295
ECART TYPE		26		102
EFFECTIF		30		30

TABLEAU II

Caractéristiques et teneurs en Po-210 des bénétiers de Papara.

INTERPRETATION OF INTERNAL DOSE CALCULATIONS FOR DOSE RECORD KEEPING

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INTRODUCTION

Persons working with radionuclides are exposed to external and internal sources of radiation. The fundamental concept of adding the dose equivalent from both types of source, devised in ICRP26[1] is embodied in the UK regulations produced by the Health & Safety Executive in regulations[2] and a code of practice[3]. The measurement of external radiation is well established[4] through the use of film or thermoluminescent dosemeters. Internal dose assessment presents many more problems.

The techniques of whole (or partial) body monitoring are inadequate for most actinides, eg ^{239}Pu, at the annual level of intake (ALI)[5] and detection in urine and faeces at ≤ 0.1 ALI is only possible in the first few days after intake: sampling of the aerosol in the breathing zone is therefore important. Rapid variation in the concentration of aerosols and radioactive gases can be observed over distances of less than 1 m[6]. Thus it is important to place any collector of aerosols (a personal air sampler; PAS) as close to the face as possible; usually on the upper chest. The derived air concentration (DAC)[5] for some actinides, eg ^{239}Pu, is such that there are only a few particles per m^3 to detect and these particles will be randomly distributed in time and space. As the activity per particle usually follows a lognormal distribution, the activity collected on filter samplers will have a wide statistical spread. Other radionuclides do not necessarily present such problems and uranium detection by a PAS is probably the most sensitive and accurate method of dose assessment.

This paper will present the models used to provide dose estimates from personal air sampling, nose swabs, whole (and partial) body monitoring, excretion analysis; followed by an example. Also included are an outline of the proposed methods of dose record keeping and an initiative through Eurados-Cendos to improve the dose assessment.

MODEL DEVELOPMENT FOR ASSESSING INTERNAL RADIOACTIVITY

The model used at Harwell is essentially that described in ICRP30[5] by combining the lung, gut and other organs models. Each compartment is represented by a first order differential equation. There is no indication of excretion from organs, other than the gut, so the model includes a choice of excretion to urine or faeces from each organ. The excretion model for some radionuclides, such as ^{239}Pu, are presented in ICRP10[7] and 10A[8] as multiparameter function unrelated to particular organs and so the model output has been fitted to the function by adding additional compartments with appropriate input & output constants. The model has been programmed in two forms:

(a) a completely analytical solution for one pass through the system;
(b) numerical form for either one pass or the recycling of material as occurs with iodine.

(It should be noted that the ICRP30[5] model is designed for dose estimation and not as a representation of metabolism including excretion and a better model is required.)

The input to the body can be via lungs, with variable particle size, mouth or injection into the blood. The output from both programs is both tabular and graphical and gives the activity in individual organs and sums of organ contents, for assessing nose swabs and body monitoring measurements; and excretion in urine and faeces: all information as a function of time. Over the first few days it is important to integrate the excretion over the day rather than take the average organ content times the excretion constant because some of the latter are of the order of 0.25 d.

ANALYSIS OF A URANIUM AND THORIUM CASE

This is an example of a chronic exposure at various times throughout 1986, and the case is used for illustrative purposes to demonstrate the approach used. A personal air sampler (PAS) is used for all operations which involve uranium fuel fabrication. Urine samples are taken at approximately quarterly intervals but because the intake is a chronic one it is not possible to interpret the urine data directly in terms of intake. Thus for comparison purposes, the excretion rate in urine has been calculated from the PAS data and is as follows:

Date	Bioassay Measured mBq	PAS(Y) Calculated mBq	Ratio PAS/BIO (Y)	PAS(W) Calculated mBq	Ratio PAS/BIO (W)
27.01.86	-	0.0	-	0.32	-
17.02.86	22	0.4	0.02	11.65	0.53
12.05.86	67	4.3	0.06	116.3	1.74
11.08.86	-	1.3	-	26.2	1.31
3.11.86	-	5.0	0.07	121.9	1.74
8.12.86	254	94.7	0.37	2446.	9.62

There is probably a mixture of class Y and W material[5] but since the ALIs differ by a factor of 15 and the urinary excretion per unit intake is approximately 25 times higher for class W than Y, a small amount of class W material can change the urinary excretion rate dramatically.

Taking the conservative assumption of class Y, the resulting total intake for 1986 is shown below

Limit Nuclide	Intake Bq	ALI %	CDE mSv	CI	Organ
Non-Stochastic					
Nat U	705	35	176	0.35	Lungs
Th	2.6	0.5	2.4	0.005	Lungs
Total		36	179	0.36	Lungs
Nat U	705	0	0	0	Bone Surface
Th	2.6	2.6	13	0.026	Bone Surface
Total		2.6	13	0.03	Bone Surface
Stochastic					
Nat U	705	35	17.6	0.35	Whole Body
Th	2.6	1.3	0.7	0.01	Whole Body
Total		36	18	0.36	Whole Body

It is interesting to note that the organs for the non-stochastic limits are different for uranium and thorium. Thorium contributes a small dose to the lungs but there is no contribution to the bone surfaces from uranium. Also noteworthy is the similarity between compliance indices for non-stochastic and stochastic doses.

DOSE RECORD KEEPING

The assessment of dose equivalent or more commonly compliance index is entered on to the person's record. Compliance intake is defined for stochastic, CI(s), and non-stochastic intakes, CI(NS) as:

$$CI(S) = \frac{\text{Intake in Bq}}{\text{ALI (Stochastic) in Bq}} \quad \text{and} \quad CI(NS) = \frac{\text{Intake in Bq}}{\text{ALI (organ) in Bq}}$$

The non-stochastic compliance index may be needed for several organs if the person is exposed to several radionuclides and/or external radiation. If a person exceeds a total CI(S) or CI(NS) in a year of 0.30 then the exposure(s) is investigated by management to ascertain whether doses are being kept as low as reasonably practicable (ALARP). There are facilities within the regulations to allow an adjustment of the internal dose as more information on excretion becomes available.

The dose record for an individual is already on computer but a new data base, Harwell Approved Dosimetry Services (HADES)[9], is being established to provide on line information to Health Physicists to enable them to maintain day to day dose control. Information from film dosemeters and personal air samplers (PASs) is already stored in machine readable form and bioassay data will soon be available on the control computer. (Body monitor measurements are only made as part of an assessment.) Measurements are at or below the threshold of measurement or a CI <0.05. Results from PASs or bioassay measurements are handled automatically and compared to check if they come from the same person over the same time period. If there is an indication of a significant intake eg PAS > 100 DAC.h, then a detailed dose assessment is made based upon all data available.

HADES is used to combine all external and internal dose data into regular reports for the Health Physicist and other reports for the Central Index for Dose Information (CIDI) held by the UK National Radiological Protection Board[10]. This confidential index starts with a Registration Report to notify CIDI of monitoring for an individual; an Entry Report is returned from CIDI to provide any previous history on that individual; the Annual Return provides summarised doses and classification of the individual to CIDI within 3 months of the end of the year; a Termination Record notifies CIDI that the individual has left employment and gives details of doses received. It is hoped that HADES will be fully operational by mid 1988.

EUROPEAN INITIATIVE TO IMPROVE INTERNAL DOSE ASSESSMENTS

Eurados-Cendos, whose objectives are to coordinate dosimetry research programmes in the EEC through exchanges both within Europe and outside, instituted committee 6 on the Assessment of Internal Dose to prepare guidance on the interpretation of monitoring data relating to internal exposures of radiation workers and the implementation of ICRP recommendations on this topic within Europe. The committee drawn at present from the Federal Republic of Germany, France and the UK has set up a programme of work to: devise more

realistic models for excretion; improve interpretation of air sampling, in-vivo monitoring and bioassay data; examine autopsy data in relation to intake; ensure cross-frontier compatibility of dose records; provide for information exchange on the limited number of cases that arise in the nuclear industry in Europe. It is recognised that existing information is sparse and any improvement in knowledge of internal dosimetry and metabolism will assist European Dosimetry Laboratories in the rapid assessment of cases and so improve dose estimates to limit exposure to individuals.

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HUMAN TISSUE STUDIES OF THE UNITED STATES TRANSURANIUM AND URANIUM REGISTRIES: A PROGRESS REPORT

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INTRODUCTION

The United States Transuranium and Uranium Registries are parallel human tissue analysis programs dedicated to the study of the biokinetics and dosimetry of uranium and the higher actinides in man. Both are operated by the Hanford Environmental Health Foundation under contract to the U.S. Department of Energy. The U. S. Transuranium Registry (USTR) began in 1968 as the National Plutonium Registry. In 1970 the name was changed to reflect a broader programmatic concern with the other transuranic elements. The U. S. Uranium Registry (USUR) was established in 1978.

To evaluate human experience with the actinides, both Registries rely on voluntary donation of selected tissues or, in certain cases, the whole body after death. The routine autopsy tissue protocols of both Registries are basically the same and involve collection of tissues in which the actinides are known to concentrate. These include the liver, lungs and tracheobronchial lymph nodes, samples of bone, and kidney. Details of the autopsy protocol as well as the general operation of the Registries have been previously described (Breitenstein 1981; Kathren 1987; Marks 1981; Moore and Breitenstein 1983; Swint and Kathren 1986).

TRANSURANIUM ELEMENTS IN THE SKELETON

From the earliest studies, the skeleton has been identified as a major depot for plutonium and the higher actinides. Accordingly, the skeleton is a major area of emphasis of the human tissue studies of the USTR. One of the earliest findings related to human skeletal deposition of plutonium and americium was reported last year and described a clear inverse proportionality between calcium content (as represented by the ratio of ash weight to wet weight) and the concentration of Pu or Am in the ash (Kathren, McInroy and Swint 1987). Review of previously published data revealed a similar relationship in monkeys but not dogs. This study has now been extended to four whole bodies; in three of these cases, data are available for both Pu-239,240 and Am-241 while only Am-241 was measured in the fourth. All four cases were occupationally exposed individuals in whom exposure had occurred two or more decades prior to death. Inhalation was the route of entry in three cases; in the fourth, a wound some 25 years prior to death was the primary source of exposure. The inverse proportionality was found to hold generally for the entire skeleton for both nuclides in all four cases and, with few exceptions, for individual bones and groups of bones as well. Excellent linear correlation was found in the long bones ($-0.81 < r < -1.0$); the poorest correlation was found in the ribs. This relationship suggests that the quantity of Pu or Am in the entire

skeleton can be estimated with reasonable accuracy from radiochemical analysis of a few (and perhaps even a single) bone samples.

Am and Pu distributions throughout the skeletons of all four cases were similar. The percentage of Pu or Am activity in certain specific bones or groups of functionally related bones (e.g. hand phalanges) relative to the total activity in the skeleton was remarkably constant and correlated well with the percentage of the total skeletal wet weight. This is an important finding in that it enables good estimates of the total skeletal burden to be made from a single bone. The patella is particularly well suited to this purpose; it is easily obtained at autopsy and was found to contain 0.36 $\pm$ 0.04 per cent (0.72 $\pm$ 0.07 for both patellae) of the total skeletal burden.

To better evaluate leukemogenic risk from skeletal deposition of actinides, the USTR has undertaken a study of the amount of plutonium in the bone marrow relative to that in the rest of the skeleton. The first phase of this work, now complete, was an evaluation of various extraction techniques for separation of the organic and mineralized components of bone. Vertebral bodies from an individual known to have had occupational exposure to plutonium were used for this study. Ethylenediamine extraction provided the most complete separation of the organic component, removing not only the red and yellow marrow but the periosteum and endosteum as well. Mechanical removal, either by sonication in saline or washing out the marrow cavities with a water jet were about equally effective and seemed to remove the cellular marrow and associated fat and water. The water jet also appeared to remove small spicules of bone, and hence a small amount of the inorganic component as well. The poorest method was autoclaving, which appeared to remove only the yellow marrow.

This study revealed only a small fraction of total skeletal plutonium--on the order of 3%--was contained in the marrow. The dose to the marrow is thus relatively low by comparison with the dose incurred by the rest of the bone suggesting the risk of solid bone tumors such as osteosarcoma from skeletal deposition of plutonium is several fold greater than the risk of leukemia. The study also implied the dose to the sensitive cells lying on the bone surfaces--the periosteum and endosteum--may be several fold greater than the dose to other portions of the skeleton suggesting these tissues may, in fact, have the greatest risk.

BIOKINETIC MODELLING OF PLUTONIUM AND AMERICIUM IN MAN

The importance of human tissue data for biokinetic modelling cannot be understated. Evaluation of the first whole body donation led to important findings with respect to the metabolism and fate of Am in the body, and a detailed analysis was the subject of a special issue of Health Physics (Roessler 1985). Preliminary evaluation of the radiochemical results from three subsequent whole bodies, all Pu exposure cases, generally supports the ICRP 30 and ICRP 48 models with respect to initial partitioning of plutonium between skeleton and liver.

Evaluation of Pu-238, Pu-239+240 and Am-241 concentrations

and content in the skeleton and liver of more than 40 occupationally exposed chronic inhalation cases for whom suitable samples had been collected at autopsy revealed different partitioning between skeleton and liver for each of the three nuclides. For Am-241, the mean ratio of activity in the skeleton relative to that in the liver was 3.14; for Pu-238 and Pu-239 the corresponding ratios are 1.73 and 1.14, respectively. Only the Pu-239 data fit the ICRP model well. More significantly, the data suggest that Pu-238 and Pu-239 may be handled differently within the body, an observation that could have significant implications for dosimetry and standards setting. Further mathematical evaluation of these data indicates the differential partitioning between the two plutonium isotopes may be attributable to a much shorter effective half-time for Pu-238 in the liver as compared with Pu-239. On the basis of these data, the effective half-time for Pu-238 in liver has been calculated to lie somewhere in the range 4-9 years. This value should be compared with the values of 40 and 20 years for the biological half-time in liver for plutonium irrespective of isotope given in ICRP Publications 30 and 48, respectively.

In another study involving postmortem analyses of tissue from former plutonium workers, the USTR is examining the activity in the lungs and tracheobronchial lymph nodes in relation to the applicability and appropriateness of various lung models. The results to date are as yet preliminary but indicate relatively large, long-term depots in the lungs and tracheobronchial lymph nodes. As this study progresses, it may provide important new insights into lung clearance of plutonium in occupationally exposed persons.

Finally, the comparison of postmortem tissue depositions with estimates made in-vivo from either direct counting or excreta analysis with suitable models, merits some mention. In a study just published (Kathren, Heid and Swint 1987), it was observed that autopsy estimates of systemic plutonium were typically several fold lower than estimates made in-vivo using urinary excretion data and various models. The magnitude of the discrepancy was inversely related to deposition; typically the smaller the estimated deposition, the greater the discrepancy. For depositions estimated to be in the KBq region, the autopsy and in-vivo estimates were relatively close; for autopsy estimates of a few tens of Bq or less, the in vivo estimates were on the order of 5 to 10 times greater.

URANIUM KIDNEY TOXICITY

Despite more than a century of human studies with uranium, there are still unresolved questions with respect to kidney toxicity of this element, and there have been few, if any, direct studies of kidney pathology in uranium workers. In an effort to evaluate the possible long-term nephrotoxicity of uranium, the U.S. Uranium Registry submitted histological kidney specimens obtained at autopsy from four registrants with known occupational uranium exposure and similar samples from 5 persons without known exposure to a pathologist skilled in recognizing renal tissue changes associated with uranium toxicity. Occupational uranium

exposure was estimated from urinary excretion data to be below detectable limits in two cases and a few milligrams in the highest exposed individual. All registrants had working lifetimes with uranium in excess of 20 years. The consultant pathologist was asked to review the slides for toxic changes which might be attributable to uranium. These histopathologic changes represented the residue of a mild chronic toxic process, i.e. tubular atrophy, patchy chronic inflammation, tubular dilatation, calcified tubular casts and hydronephrosis. He was unable to recognize specific histological changes that would reliably discriminate between the two groups. Although inconclusive to date, this project is being enlarged to encompass registrants with larger tissue uranium depositions.

ACKNOWLEDGEMENTS

The research reported in this paper was supported by the Office of Health and Environmental Research, United States Department of Energy. The authors also acknowledge with thanks the contributions of our colleague James F. McInroy and his staff at Los Alamos National Laboratory, who perform radiochemical analysis of tissue and radiobiological consultation vital to the work of the Registries.

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INTAKE RETENTION FUNCTIONS AND THEIR APPLICATIONS TO BIOASSAY AND THE ESTIMATION OF INTERNAL RADIATION DOSES

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INTRODUCTION

Intake retention functions that give the fraction of an intake expected to be present in a specified in vivo or in vitro bioassay compartment at any time after a single acute exposure or after the onset of a continuous exposure are needed for the proper design and conduct of bioassay programs. This paper summarizes the derivation of these functions from a multicompartmental model and a recursive catenary kinetics equation that completely describe the metabolism of radioelements from intake to excretion, accounting for the delay in uptake from compartments in the respiratory and gastrointestinal tracts and the recycling of radioelements between systemic compartments. Because the metabolic model of a worker will not in general be known and because bioassay data from an exposed worker generally is not sufficient to obtain the worker's model, we recommend that ICRP Publication 30 or any other appropriate metabolic models or excretion functions be used to obtain parameter values for our model. In a separate paper we show how to transform a known excretion function into a pseudo systemic uptake retention function and pseudo constant fraction of systemic excretion, which then can be incorporated into our model and algorithm directly[1]. The estimation of intakes and internal radiation doses and the use of intake retention functions in the design of bioassay programs are discussed along with several examples in the oral presentation of this paper at the 7th IRPA Congress.

MODEL DESCRIPTION

The multicompartmental model describing the metabolism is depicted in Figure 1 by various one-way catenary pathways from intake to excretion. Intake pathways can include, for example, inhalation, ingestion, and absorption from a wound. The word catenary refers to a chain of compartments, and a one-way catenary system means one in which the radioelement is modeled to move in only one direction. The last compartment of all catenary systems is designated in Figure 1 as the 'total excretion compartment', which may be thought of as a 'bucket' where all excretion is collected. Although only one-way transfers between compartments are depicted, the model does in fact account for the recycling of elements between systemic compartments by use of a systemic uptake retention function for the whole body whose parameter values incorporate this recycling. This function is shown at the bottom of Figure 1 as a sum of m exponential terms. The word compartment is used in its mathematical sense, and it may or may not represent a real structured physiological entity in the body.

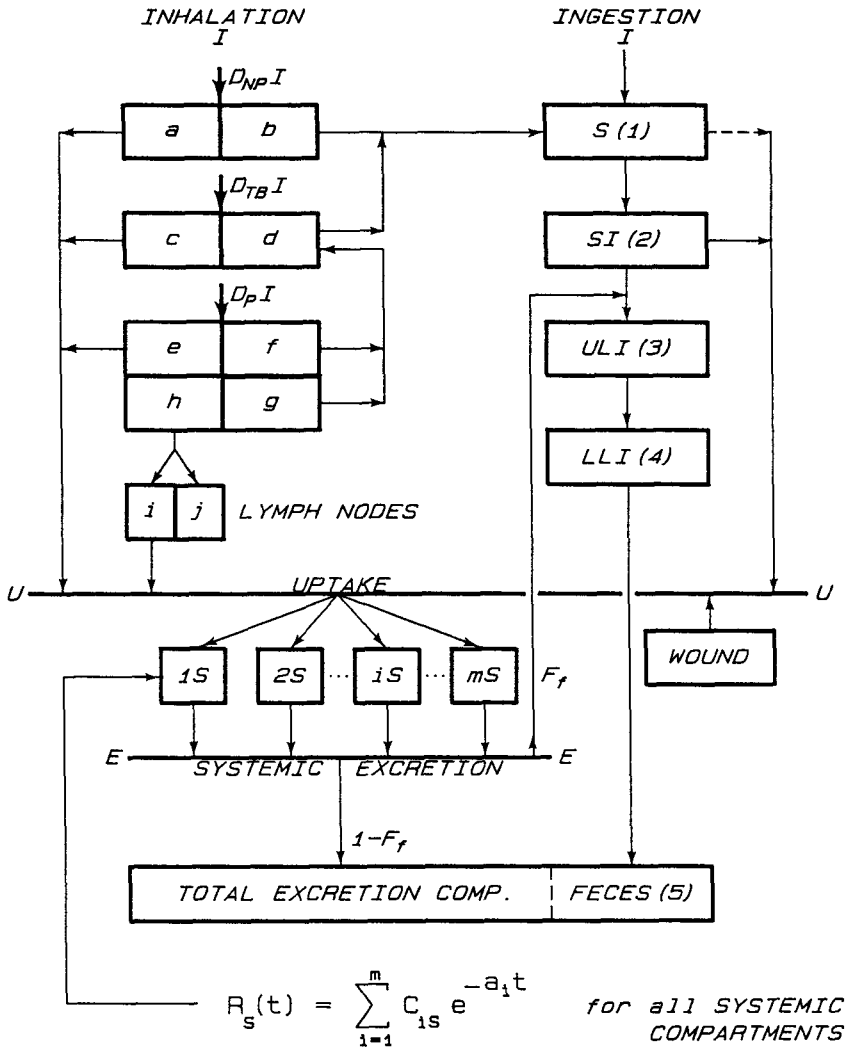

Figure 1. Catenary pathways from intake to excretion. Respiratory tract compartments are 'a' to 'j' inclusive. Gastrointestinal tract compartments are 1 to 4. The stable element uptake retention function $R_S(t)$ for the whole body is given by a sum (i=1 to m) of exponential terms with constant coefficients C_{iS}, which give the effective fractions of an uptake U that deposit in each ith systemic compartment. Each ith exponential $\exp(-a_i t)$ of $R_S(t)$ is treated as a deposition retention function of a systemic catenary compartment, which is cleared directly to systemic excretion E at an instantaneous fractional rate given by the eigenvalue or effective rate constant a_i of the exponential term. The fractions F_f and $(1-F_f)$ are the effective fractions of systemic excretion by the fecal and all other excretion pathways respectively.

Arrows in Figure 1 shown leaving a compartment indicate the specific compartmental biological removal pathways, each of which is characterized by a specific biological translocation rate constant. In addition, radioactive elements are removed from each compartment by radioactive decay, which is implicit and characterized by the decay constant λ of the particular radioelement. Because the metabolism can be described in terms of simple one-way transfers between compartments composing various catenary systems, the recursive catenary kinetics expression shown below by Equation 1 can be used to obtain explicit equations for the fraction of an intake expected to be present in all *in vivo* and *in vitro* bioassay compartments of interest. Although this description of the metabolism may seem oversimplified, it is mathematically consistent with more sophisticated models that rely on the use of convolution integrals for predicting the contents of systemic and excretion compartments following intakes.

CATENARY KINETICS EQUATION

By application of the following catenary kinetics equation to the model depicted in Figure 1, the fraction of an intake I expected to be present in an *in vivo* or an *in vitro* (i.e., accumulated excretion) compartment of interest is obtained:

$$i_n(t) = \sum_C F_C \prod_{p=1}^{n-1} k_{p,p+1} \sum_{j=1}^{n} \frac{\exp(-k_j t)}{\prod_{\substack{p=1 \\ p \neq j}}^{n} (k_p - k_j)} \qquad (1)$$

where:

$i_n(t) = \langle q_n(t) \rangle / I$ = fraction of acute intake I of radioelement expected to be present at time t in nth compartment as a result of contributions from all chains C,

$\langle q_n(t) \rangle$ = expected content of radioelement in nth compartment,

F_C = fraction of intake deposited in first compartment of chain C,

$k_{p,p+1}$ = rate constant giving the instantaneous fraction of the content of compartment p translocated per unit time to the (p+1)th compartment,

k_j = rate constant describing total removal of radioelement from jth compartment and given by $K_j + \lambda$, where K_j is the total of all biological translocation rate constants and λ is the decay constant, and

n = numerical index for that catenary compartment in chain C whose intake retention function is being obtained.

After values are substituted for the deposition fractions

F_C and rate constants in Equation 1, factors are multiplied, and the resulting coefficients of common exponential terms are added, then the intake retention function is expressed by a simple sum of exponential terms with constant coefficients. The coefficients are independent of the decay constant or half-life of the radioelement; so, they also apply to the intake retention function for the stable element, which can be obtained from Equation 1 by replacing each total removal rate constant k_j in each exponential by the total biological removal rate constant K_j.

OTHER INTAKE RETENTION FUNCTIONS

By applying the general catenary kinetics equation to the appropriate chains in Figure 1, single acute inhalation intake retention functions can be obtained for specific organs, organ systems, and excretion by summing the functions for the appropriate compartments. This includes: (1) the nasal passage (compartments a and b), (2) the lungs (compartments c-j), (3) the GI tract (compartments 1-4), (4) the systemic whole body (compartments 1S-mS), (5) a specific systemic organ x, e.g., the thyroid, by replacing $R_S(t)$ in Figure 1 by the organ's stable element uptake retention function $R_x(t)$ expressed by a sum of m exponential terms with constant coefficients (compartments 1x-mx), (6) accumulated total systemic excretion (compartment E), (7) accumulated total fecal excretion (compartment 5), (8) accumulated urinary excretion (F_u times the intake retention function for compartment E if F_u is constant or if a pseudo function $R_S(t)$ and pseudo constant F_u value are used), and (9) the total body (the sum of (1) - (4) above).

Once the single acute intake retention functions have been determined for the radioelement, other functions such as continuous intake functions, incremental excretion intake functions for single or continuous intakes, excretion rate intake functions for single or continuous intakes, and functions that yield the number of disintegrations in any *in vivo* compartment can be obtained by simply replacing the exponentials in the terms for the single intake functions by another time function. We have summarized these replacement functions and have discussed their applications in a more detailed manuscript of the subject of this paper, which we will provide upon request.

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COMMENTS ON THE RAT LUNG AS A HUMAN SURROGATE IN INHALATION STUDIES

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INTRODUCTION

The laboratory rat is often used as a surrogate to estimate the hazard to human health following inhalation exposure to ambient aerosols. Extrapolation of rat deposition data to humans depends, however, on the similarities and differences between the morphometric structures of the two airway systems. The main structural difference between the lungs of the two species, aside from dimensions per se, is their respective airway branching pattern: while the human lung is a rather symmetrically, dichotomously dividing system, the rat network is a more monopodial branching structure (Yeh et al. 1979).

In our stochastic modeling approach (Koblinger and Hofmann 1986) to defining suitable morphologies for human and rat lung, we utilize measured morphometric dimensions as the data base upon which a rigorous statistical analysis is performed, instead of forcing them into a formalized, average pathway scheme. This stochastic approach allows us, therefore, to account for structural irregularities, such as asymmetric branching, monopodial structure, and inter- and intra-subject variability.

STATISTICAL ANALYSIS OF MORPHOMETRIC DIFFERENCES

Our comparison between the human and rat lung will focus on the symmetry/asymmetry aspect of the branching pattern in the tracheobronchial (TB) region and on the applicability of the generation concept to characterize properly the position and physiologic function of each airway in the respective system of bifurcating generations. Morphometric data for both tracheobronchial regions have been derived from measurements on silicon rubber replica casts at the Lovelace Inhalation Toxicology Research Institute (Raabe et al. 1976). The common origin of both data sets, i.e., using the same laboratory techniques for the human and rat lung, facilitates the interspecies comparison of data.

For a symmetric branching scheme of airways of circular cross section, the theoretical ratio of the parent-to-daughter (both daughters) cross section is 0.794 (D'Arcy Thompson 1942). Our calculated average value for fuman bronchial generations 6-17 of 0.82 (Koblinger and Hofmann 1985) is close to the symmetric value, suggesting that the human TB tree is reasonably approximated by a dichotomous, symmetric airway system.

A distinct asymmetry of the TB branching pattern, however, is found in the rat lung. In Fig. 1 the cumulative distribution functions for the ratios of the major daughter-to-parent and minor daughter-to--parent diameters are plotted for different parent airway diameters. A consistent difference between major and minor daughters is apparent for all parent diameters, illustrating a more asymmetric structure of the rat TB region relative to the human. An additional indicator of structural asymmetry is the distribution of branching angles for major and minor daughters, shown in Fig. 2. In the rat lung, in most cases, major daughters deviate from the parent by only a small angle, whereas minor daughter branching angles are much larger. This is a clear indication of the monopodial structure of the rat tracheobronchial region. There is another important interspecies difference in the respective branching patterns: in the rat lung the branching angles are practically independent of the generation number, while in the human lung branching angles decrease with increasing generation number.

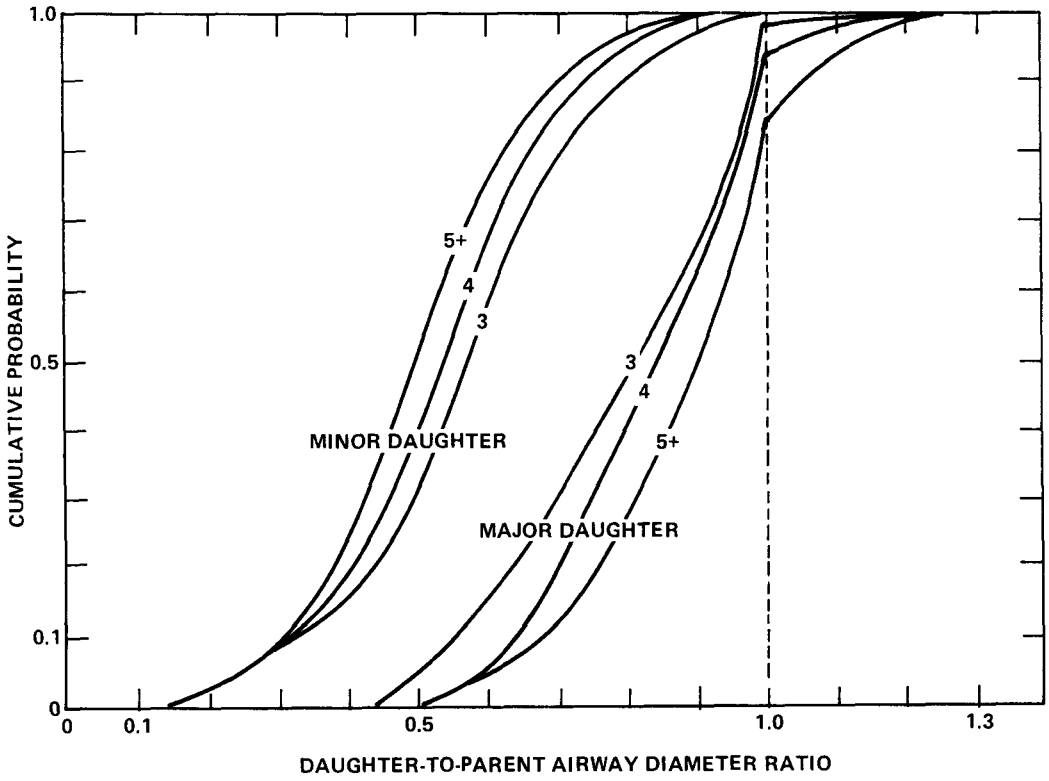

Figure 1 Cumulative distribution functions for major daughter-to-parent and minor daughter-to-parent diameter ratios for different parent diameters (in O.1 mm units) in the rat lung. The curves denoted by 5+ contain all parent diameters >O.5 mm, for curves do not change significantly above O.5 mm.

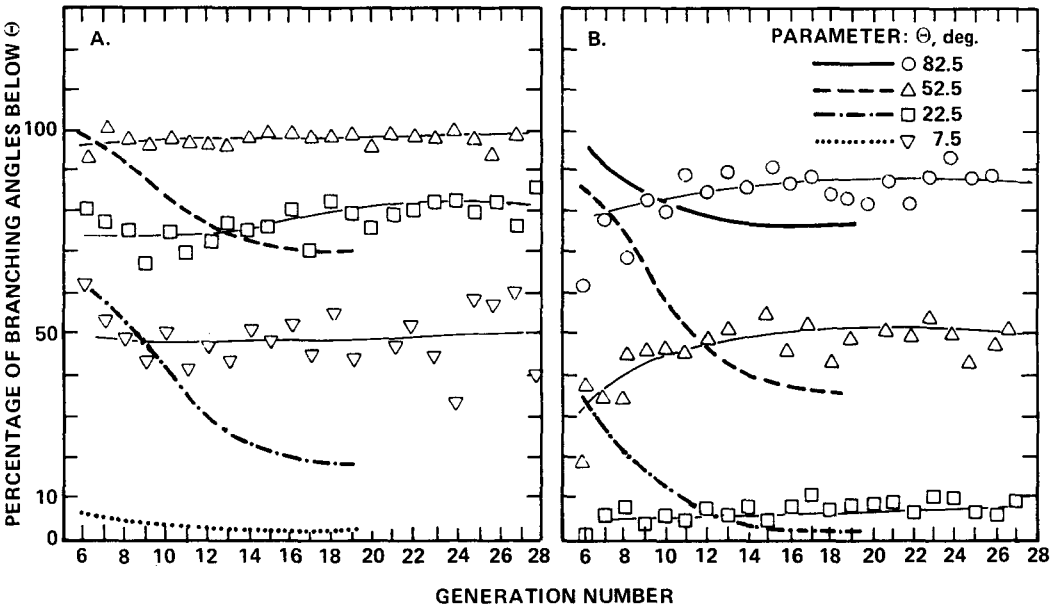

Figure 2 Cumulative branching angle distributions for major (panel A) and minor (panel B) daughters plotted versus generation number for arbitrarily selected branching angles. Solid lines denote rat lung, broken lines signify human lung.

Let us now investigate the termination probability, i.e., the probability that an airway of given linear dimensions or generation number is the terminal bronchiole and that the following airway belongs already to the alveolated, acinar region of the lung. For the human lung the termination probability depends on diameter and generation number: at a given diameter, it increases with increasing generation number and, for a given generation, it increases with decreasing diameter (Koblinger and Hofmann 1985).

In the rat lung the termination probability depends practically only on airway diameters but not on generation numbers; thus, the generation concept provides an inadequate description of the monopodial structure of the rat lung.

Considering all the arguments above, we propose to classify bronchial airways in the rat lung by their diameters (or diameter classes), and not by their generation numbers.

CONCLUSIONS

The statistical analysis has shown the above mentioned structural differences. The question, whether the rat lung is an applicable surrogate for human lungs for inhaled particles deposition calculations can be answered after comparing results calculated by the use of the two airway geometries.

We intend to carry out such comparative calculations with our stochastic lung model, in the near future.

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INDOS - AN INTERNAL RADIATION DOSIMETRY ASSESSMENT COMPUTER CODE

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INTRODUCTION

Bioassay is the final quality control procedure that is used to assure adequate protection of workers from internal radiation exposures. Intake retention functions, which give the expected fraction of an intake in a bioassay compartment, are needed for the proper design and conduct of bioassay programs, including the interpretation of bioassay data in terms of estimated intakes and internal radiation doses. In cases of accidental exposures, estimated intakes can be compared to either regulatory intake limits or the ICRP Publication 30 Annual Limits on Intakes (ALIs). The quotient of an estimated intake of a radioelement by its stochastic based ALI value when multiplied by 0.05 Sv gives an estimate of the whole body committed effective dose equivalent. This dose assessment of bioassay data can be used to identify those exposures that require further investigation from those that do not require such action and expenditure of time and resources.

This paper describes the main features of a bioassay/internal dosimetry IBM PC computer code called INDOS, which provides a rapid and efficient way of calculating values for intake retention functions and estimating intakes from bioassay data, including a complete documentation of all input and output data used in each individual exposure case. The derivation of the intake retention functions is outlined in a companion paper. Examples of the input and output from actual exposure cases are shown in the poster session for this paper. A demonstration diskette, which allows evaluation of essentially all features of the actual INDOS software, is available from the authors upon request.

INTAKE RETENTION FUNCTIONS

As shown by Equation 1 and Figure 1 of our companion paper, intake retention functions for all in vivo and in vitro bioassay compartments are obtained by application of a single catenary kinetics equation to a multicompartmental model that describes the metabolism of radioelements from intake to excretion. For a single acute intake, functions are expressed by sums of exponentials with constant coefficients, including the function for an accumulated excretion compartment. The coefficients of the exponentials are independent of the decay constant λ of the radioelement; so, they also apply to the intake retention function for the stable element. Replacement functions for the exponentials provide an efficient way of calculating the values for other types of intake retention functions that are described

in INDOS. For example, the exponential exp(-kt) in each term of the radioelement accumulated urine function, which gives the fraction of a single acute intake that is expected to be present in the accumulated urine compartment, is replaced by

$$\left[e^{-(k-\lambda)t} - e^{-(k-\lambda)(t-\Delta t)}\right] e^{-\lambda t}$$

to obtain the incremental urine function, which gives the fraction of a single acute intake expected to be present at time t in an incremental sample of urine that is collected from $(t-\Delta t)$ to time t after the intake, where:

$k = K+\lambda$ = total removal rate constant, which is shown to be calculated by the sum of the compartment's total biological removal rate constant K and the radioelement's decay constant λ.

PROGRAM DESCRIPTIONS

All of the programs in INDOS allow the user to input metabolic parameter values different from the standard ICRP Publication 30 values for the respiratory tract, GI tract, and systemic compartments. When more specific models and parameter values have shown to provide greater accuracy, their use will provide a better estimation of intakes and internal radiation doses than can be obtained from standard ICRP 30 models. Default values are provided for parameter values associated with the ICRP 30 respiratory and GI tract models and inhalation compound classes.

The programs in the INDOS software include:

1. Program INT: This program estimates an intake from one or more bioassay measurements of a radioelement. The user specifies the intake conditions and other pertinent data. Based upon one of three selected assumptions regarding the variance of the bioassay data, the program then performs a best fit to the data to obtain an estimate of the intake, exposure, and associated dosimetry. This includes the estimated exposure in units of MPC-hours or DAC-hours and the committed effective dose equivalent. All or part of the input data, including default values, can be stored on a disk or printed with the output to document the models and parameter values that were used.

2. Program DIL: This program generates tables of Derived Investiation Levels (DILs) for in vivo and/or in vitro bioassay compartments of interest. The user specifies the intake that corresponds to the specified investigation level (e.g., an intake of 0.02 ALI, which corresponds to an exposure of 40 DAC-hours) and inputs other pertinent data. The program then calculates the activity of the specified

radioelement that is expected to be present in particular bioassay compartments at various chosen times after a single acute intake or after the onset of a specified continuous intake interval. One can use such tables, for example, to screen those routine bioassay measurements that may require further investigation. By selecting that monitoring frequency and those procedures that have sufficient sensitivity and accuracy for the detection of a DIL value, a routine bioassay program can be designed to help ensure that significant intakes are detected, recorded, and properly investigated.

3. Program IRF: This program generates tables of Intake Retention Fractions (IRFs) for in vivo and/or in vitro bioassay compartments of interest. The input for this program is similar to that of Program DIL. The program calculates the fraction of an intake expected to be present in particular bioassay compartments at various chosen times after a single acute intake or after the onset of a specified continuous intake interval. One can use such a table, for example, to make a preliminary estimate of an intake from a single bioassay measurement of a radioelement by dividing the measurement by the IRF value given in the table for the radioelement and bioassay compartment of interest.

4. Program NOD: This program generates tables of values for the Number of Disintegrations that are expected to occur over a chosen time interval in various in vivo compartments of interest due to an acute intake specified by the user. Such a table can be used, for example, to calculate individual organ doses over any time interval after an intake, including the 50 year standard interval that is used for the calculation of committed doses and the associated ALIs. Estimated doses may be useful as guidance for medical and follow-up bioassay procedures. For a single radionuclide having no progeny that significantly contribute to the doses, the value of NOD over the standard 50 year interval along with other parameter values from ICRP Publication 30 can be used for calculating custom ALIs and DACs. This can be done for various particle size distributions and chemical compound classes or mixtures for a radioelement that is known to be present in a given working environment. In addition, if metabolic parameter values that reflect the metabolism of exposed workers are available, then they can be used as input to derive more realistic ALIs and DACs.

SPECIFICATIONS BY THE USER

The INDOS programs allow the user to specify the following at various prompts on the input screens:

1. Exposure Condition: (a) single acute intake or (b) contin-

uous intake (excludes program NOD);

2. Intake Pathway: (a) inhalation, (b) ingestion, (c) injection, or (d) delayed uptake through a wound;

3. Metabolic Data and Physical/Chemical Characteristics of Intake: (a) ICRP 30 compound classifications D, W, and Y or mixtures, (b) activity median aerodynamic diameter (AMAD) of inhaled aerosol and (c) ICRP 30 metabolic models or customized models;

4. In Vivo Bioassay Compartments or Measurements: (a) lungs, (b) GI tract, (c) systemic whole body whose uptake retention function is specified, (d) whole body excluding nasal passage (i.e., a + b + c), and (e) individual organ or tissue whose uptake retention function is specified; and

5. In Vitro Bioassay Compartments or Measurements: (a) accumulated urine, (b) incremental urine, (c) urine concentration, (d) accumulated feces, and (e) incremental feces.

With respect to the in vivo bioassay compartments, the somewhat redundant term "systemic whole body" is used to represent all internal organs and tissues and the entire systemic circulation including extracellular fluid. A radionuclide previously absorbed into the systemic circulation and present in the systemic whole body represents systemic internal contamination. A radionuclide deposited and present on the epithelial tissue in the respiratory tract or present in the contents of segments of the GI tract is treated separately and is not considered as part of this systemic contamination.

SUMMARY

Our experience with INDOS has proven it to be a practical way for the design and conduct of bioassay programs that will (1) help to provide adequate protection of workers from internal radiation exposures and (2) provide a rapid and efficient way to obtain estimates of intakes and internal radiation doses.

COMPARISON OF PREMORTEM AND POSTMORTEM ESTIMATES OF PLUTONIUM IN THE SKELETON AND LIVER OF SIX INDIVIDUALS

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Assessment of organ burdens after internal exposures to radionuclides is often necessary to evaluate the health and regulatory implications of the exposure. The assessment of plutonium activity in skeleton and liver is usually estimated from measurements of plutonium excreted via urine. As part of the overall evaluation of internal dose assessment techniques, it is useful to compare the results of organ burden estimates made from evaluation of urinary excretion data with those made at death from tissue samples collected posthumously from the individual.

Estimates of plutonium in the skeleton and liver, based on postmortem analysis of tissue samples for six individuals, were obtained from the U.S. Transuranium Registry (USTR). Bioassay data and other radiation exposure information obtained from the individuals' files were used to estimate their skeleton and liver burdens at the times of their deaths, and these estimates were compared to those obtained through tissue analysis.

PREMORTEM ASSESSMENT OF INTERNAL ORGAN BURDENS

At the U.S. Department of Energy's Hanford Site, an empirically derived excretion model is used to estimate the uptake of plutonium to systemic circulation, and the metabolic assumptions in Publication 48 of the International Commission on Radiation Protection (ICRP) [1] are used to predict the resulting organ burdens at various times post uptake. The urinary excretion model uses the function recommended by Jones [2] to describe the expected daily excretion of plutonium via urine at times after an acute injection into the blood. However, in many cases, uptake by the blood is protracted rather than acute; i.e., at intake the plutonium is not immediately taken up by blood but is deposited at a presystemic site such as the lung or a wound. The transfer of plutonium from the presystemic deposition site into the systemic circulation is assumed to be governed by linear first-order kinetics and thus can be described in terms of a transfer rate constant, λ. Using Healy's method [3], Equation (1) describes the expected daily excretion of plutonium, E(t), after the deposition of Q(0) activity in the presystemic compartment.

(a) The work performed by Battelle, Pacific Northwest Laboratory is supported by the U.S. Department of Energy (DOE) under contract DOE-AC06-76RLO 1830. The U.S. Transuranium Registry research is supported by the DOE Office of Health and Environmental Research under contract DE-AC06-76RLO 1837.

$$E(t) = Q(0)\lambda \int_{0}^{R} e^{-\lambda t} J(R-t)dt \qquad (1)$$

where J(R-t) is the value of the Jones excretion function at time R for plutonium entering the blood at time t post intake.

The excretion function is fit to the observed urinary excretion data by varying the transfer rate constant, λ, and the initial presystemic deposition quantity, Q(0). The λ and Q(0) values, providing the best fit to the excretion data, are then applied to a retention model to estimate the amount of plutonium in the skeleton and liver at times post intake. The retention model contains the presystemic compartment into which Q(0) is assumed to be deposited at the time of intake. Plutonium leaves the presystemic compartment enroute to the skeleton and liver according to the transfer rate constant, λ. Organ deposition fractions and retention half-times are based on recommendations in ICRP 48 [1].

This excretion model does not allow for the enhanced excretion that occurs during and after chelation therapy. Hanford experience has shown that the enhanced excretion rate after the administration of DTPA decreases with time post treatment and returns to levels predicted by the excretion model within 90 days of the end of treatment.

POSTMORTEM ASSESSMENT OF ORGAN DEPOSITION

The USTR accepts donations of organs and tissues from deceased plutonium workers for histopathological and biokinetic studies. At autopsy, donated tissues are removed and analyzed radiochemically. Tissue analysis procedures have been documented by Boyd et al. [4]. Tissue sample activities are scaled to either the known or Reference Man organ size [5]. The overall uncertainty in the organ burden estimates is not expected to exceed a factor of 2 [6].

EVALUATION OF EXPOSURE CASES

Assessments of skeleton and liver burdens at death were made for six individuals with well-defined exposure and bioassay histories. Premortem estimates were based on available urinalysis data and other information obtained from their files. Postmortem estimates were provided by the USTR. All six cases involved exposure to a mixture of plutonium isotopes consisting mainly of ^{239}Pu. In five of the six cases, exposure was via inhalation. In the sixth case (USTR 212) exposure was via a wound. The urine excretion function was fit using methods described previously. Only urine data representing 24-hour excretion periods was used. Sample results deviating from the excretion curve by a factor of 3 or greater were considered to represent either an erroneous measurement or an additional intake. In only one case did urine excretion data exceed the expected rate by this amount; because this result was not confirmed by follow-up samples it was determined to be erroneous. The initial selection of the presystemic

transfer rate constant, λ, used to fit the excretion model to the data was based on the solubility characteristics of the involved material. Final λ and Q(O) values for curve-fitting were obtained through regression analysis. Chelation therapy with DTPA had been applied in only one of the six cases (USTR 212). For this case, urinalysis results within 90 days following treatment were excluded.

Sample tissues of bone (vertebrae and rib) and liver were obtained at autopsy for all cases except USTR 212, for which the entire organs were donated for analysis. The entire liver and half of the skeleton were analyzed and thus the estimates of organ burdens for this case are expected to be the most accurate.

DISCUSSION OF RESULTS

Table 1 lists the premortem and postmortem estimates of the skeleton and liver burdens at death for the six cases. General consistency existed among the cases; typically, premortem estimates of skeletal burden were higher than postmortem estimates (five of six cases), while the converse was true for estimates of liver deposition. The premortem assessments averaged 1.6 and 0.6 times the postmortem estimates for skeleton burden and liver burden, respectively.

TABLE 1. Estimates of Plutonium in Skeleton and Liver, Bq

	USTR Case No.	Premortem Urinalysis	Postmortem Tissue Analysis	Ratio of Premortem-to-Postmortem Estimates
Skeleton	002	11	3.3	3.3
	006	1.8	0.89	2.0
	007	430	260	1.7
	010	130	96	1.4
	018	14	27	0.5
	212	130	100	1.3
				Mean=1.7±0.9

	USTR Case No.	Premortem Urinalysis	Postmortem Tissue Analysis	Ratio of Premortem-to-Postmortem Estimates
Liver	002	4.4	12	0.37
	006	0.85	0.56	1.5
	007	170	500	0.34
	010	57	270	0.21
	018	6.7	24	0.28
	212	56	87	0.64
				Mean=0.56±0.48

Skeleton-to-liver activity ratios at death (Table 2), were consistent for the premortem estimates averaging 2.3±0.2. Greater variability was observed among the postmortem data where skeleton-to-liver ratios averaged about one-third of the premortem ratio

average. In all cases, the postmortem estimates of skeleton-to-liver burden ratios were less than would be predicted using the distribution and retention recommendations in ICRP 48 [1].

TABLE 2. Skeleton-to-Liver Activity Ratios Based on Premortem Urinalysis and Postmortem Issue Analysis

USTR Case No.	Premortem Urinalysis	Postmortem Tissue Analysis	Age at Death	Years from Intake to Death
002	2.4	0.29	80	17
006	2.1	1.6	72	13
007	2.5	0.52	66	18
010	2.2	0.36	58	14
018	2.1	1.1	65	12
212	2.3	1.2	56	17
	Mean=2.3±0.2	Mean=0.85±0.53		

CONCLUSIONS

The assessment of organ burdens from evaluation of urine excretion measurements is roughly consistent with estimates based on postmortem tissue analysis. Individual estimates were within a factor of 3.2 for skeleton and a factor of 5 for liver with an overall average of about a factor of 2 for both skeleton and liver.

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SOLUBILITY CLASSIFICATION OF YELLOWCAKE PRODUCED BY A BRAZILIAN URANIUM MILL

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INTRODUCTION

The end product of uranium milling yellowcake, can vary in composition. Variations from ammonium diuranate to mixtures of uranium oxides, sulfates, nitrates and sodium compounds occur among different mills. Ore body characteristics, chemical extraction techniques as well as drying temperature vary and the uranium coupounds produced are expected to be process or site specific. Variations among batches are also expected (Eidscn and Mewhinney, 1978, 1980; Kalkwarf, 1989; Eidson and Griffith Jr., 1984) and so the dissolution characteristics of yellowcake in lung fluid. All this variability in solubility complicates the interpretation of bioassay results obtained during occupational monitoring of uranium mill workers.

This study was performed to classify the dissolution rate of yellowcake samples produced at the Poços de Caldas uranium mill according to the ICRP categories (ICRP, 1979), that is, class D, dissolution half-time less than 10 days, class W, from 10 to 100 days and class Y, greater than 100 days.

MATERIALS AND METHODS

Samples were automatically taken from 6 different yellowcake batches produced under routine operation of the mill. Two 6 mg samples of each batch were sieved and grounded and placed in tubes with 6 ml simulated lung fluid, SLF, prepared according to Moss (1979). The physical and chemical process was based on the work of Thein et al. (1982). After 3 minutes in a sonic bath to disperse aglomerates and/or avoid wall losses, the tubes were transferred to a shaker water bath and kept agitating and at 37ºC, to maximize dissolution rates. Samples were taken after 11 time intervals, equal to 0, 2, 4, 6, 12, 24, 72, 120, 336, 504 and 720 hours. Once removed from the water bath the samples were centrifuged. The supernate was pipetted off the tube using a plastic seringe and filtered through a 22 μm, 13 mm Millipore filter. Fresh SLF was replaced every 2 to 3 days. The pH was maintained at 7.3 - 7.4 by addition of dilute HCl acid. After a 30 day period, the residual uranium was dissolved with 6 ml concentrated nitric acid and analysed by neutron activation. A thermal neutron flux (2×10^{12} n/cm^2.s) was used to irradiate the samples. They were counted with a germa-

nium detector (ORTEC GMx-10180) and analysed using a Nuclear Data ND65-system. The software used for espectral analysis is detailed elsewhere (Vasconcellos, 1972 and Delgado, 1984).

RESULTS AND DISCUSSION

The results of the dissolution studies are summarized in Table 1. The dissolution behaviour of uranium in simulated lung fluid follows an exponential relation with time and can be described by the following equation (Mercer, 1967).

$$F = f_1 \cdot \exp\left[\frac{-0{,}693}{T_{1/2}\,(1)}\right]t + f_2 \cdot \exp\left[\frac{-0{,}693}{T_{1/2}\,(2)}\,t\right]$$

where F is the fraction of uranium remaining undissolved in the sample at time t and f represents the fraction of the sample dissolving with dissolution half-time $T_{1/2}$.

Table 1: Dissolution Parameters of Yellowcake in S.L.F.

Sample Number	More Soluble Percentage $f_1 \pm \sigma$(%)	More Soluble Dissolution Half-Time $T_{1/2} \pm \sigma$(hours)	Less Soluble Dissolution Half-Time $T_{1/2} \pm \sigma$ (days)
1	86 ± 2.1	14 ± 0.78	14 ± 0.99
2	83 ± 2.8	7.7 ± 0.028	12 ± 2.3
3	69 ± 1.0	8.2 ± 0.049	33 ± 1.8
4	76 ± 1.0	9.5 ± 0.014	30 ± 1.2
5	66 ± 2.0	9.4 ± 0.13	31 ± 0.50
6	72 ± 1.0	8.0 ± 0.34	30 ± 0.35

The generated non linear regression fit for all the 132 fractions presented correlation coefficient greater than 0.996. The pairwise comparisons among the fractions f_1 and f_2 and dissolution half-times were performed by the Newman-Keuls (N-K) test provided the variances were homogeneous (Bartlett's test). According to the N-K test on f_1, f_2 and $T_{1/2}(2)$, samples 1 and 2 were not statistically different at $\alpha = 0.01$ as well as samples 3, 4, 5 and 6. The test on $T_{1/2}(1)$ showed statistical difference only for sample number 1 with no significant difference ($\alpha = 0.01$) for all the others.

The differences encountered when $T_{1/2}(1)$ was tested seem too small (hours) when one looks at the solubility classification (days) presented by ICRP (ICRP, 1979). When compared to Kalkwarf (1979) results, the short half-time component behaves as ammonium diuranate, a form of UO_3. A range varying from zero to 24 hours can be used as the criteria to interpret the results obtained (Eidson and Griffith Jr., 1984). Based on the above discussion one can present the results of the short dissolution half-times as a single group and the long dissolution half-times and dissolution fraction as two distinct groups as indicated in Table 2.

Table 2: Results of the In Vitro Dissolution Experiment

Batch Number	f_1 (%)	σ(%)	$T_{1/2}$ (1) hours	$T_{1/2}$ (2) days
1 and 2	84	2.8	9.4	13
3, 4, 5 and 6	71	2.7	9.4	31

CONCLUSIONS

The yellowcake samples present a fast dissolution component, class D, and a slow component, class W. The fast component indicates that the yellowcake sampled contains ammonium diuranate, ADU, dissolving with half-times varying around 9.4 hours. The slow dissolution component (greater than 10 days) may characterize big ADU particles or small U_3O_8 ones dissolving in the lung fluid. The results indicate the presence of two specific chemical forms in the batches sampled but variable relative quantity of dissolved uranium among batches.

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TAILS FROM PREVIOUS EXPOSURES: A GENERAL PROBLEM IN SETTING REFERENCE LEVELS FOR THE ASSESSMENT OF INTERNAL CONTAMINATION

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SUMMARY

Reference levels for retention and excretion are evaluated for routine and special monitoring following the intake of a fraction of ICRP annual limits (ALIs) or of a unit activity. Methodologies are also suggested for taking into account the contribution by previous intakes to excretion or retention.

1. The ICRP recommendations suggest the use of reference levels (RLs) for planning and interpreting the monitoring of internal contamination. Direct measurements for X-ray and gamma-emitting radionuclides (WBC monitoring) give information on the activity existing in the body, but only in a few cases they are able to inform about its internal distribution. Indirect measurements on excreta (urine, faeces and exhaled air) (bioassay) cannot be directly related to the activity existing in the body but they can give information about the amount recently taken up. If the monitoring programme is not properly planned, direct measurements or excretion data alone cannot give any reliable information about the intakes and related doses.

2. Reference Levels (RL) may be established for any of the quantities determined in monitoring programmes - whether or not there are limits for this quantity - and can be used to determine the course of action when the value of a quantity exceeds, or is predicted to exceed, the given level. Measured values lower than the "Recording Level" should be disregared for assessing the annual dose equivalent or intake; values of dose equivalent or intake above the 'Investigation Level' (IL) become sufficiently important to justify a review of circumstances causing it or an assessment of the consequences. Derived Reference Levels (DRLs) could be defined for the derived retention and excretion values.

3. To be useful an Investigation Level has to be set in relation to a single measurement but, for a correct application, a well planned monitoring program shall be clearly established and observed. Generally an IL could be based on the expected

consistency of results during normal conditions or on the selected, optimized level of radiation protection. When an IL has been exceeded, it is generally more significant that there has been a failure of control than that one or more individuals have slightly exceeded a certain expected dose or intake.

4. For **routine monitoring** it is important to verify that the average rate of intake during a year does not exceed some pre-established values, generally less than the Annual Limits of Intakes: we are so allowed to employ the methodology and the models utilized by ICRP for computing ALIs.

5. It can be assumed that a worker is continuously exposed to a more or less uniform intake or that the intake takes place at the mid-point of the monitoring interval. RLs for routine exposure are referred to a suitable fraction (q) of the ratio of the ALI to the number (n) of controls in the year and the DRLs for retention or excretion are computed, for various sampling interval T° (days), as

$$DRL\ (T°) = q.(ALI/n).F(T°/2) = q.ALI.(T°/360).F(T°/2) \qquad (1)$$

where F(T) is the relevant function of (organ or total body) retention or (urinary or fecal) excretion.

6. For **special monitoring** as after an acute contamination, a reasonable evaluation of the intake of a radioelement can be obtained only by following the retention and/or excretion over short time intervals. This should be done in any case of important accidental contamination, even if only suspected.

7. To gain an idea of the importance of an acute contamination, as in screening to establish if contamination is negligible or if further examination or specific health intervention is necessary, the "routine" models could be used with some variations. In the respiratory system model, compartment (b) of the NP region is divided into two parts for the classes W and Y, assuming that 15% of the fraction of deposited material reaches the gastro-intestinal tract and that the 16% of the global activity deposited in the NP region is recovered in the first nasal blowing or swabbing; the endogenic faecal elimination has been also calculated, taking into account the delay during the passage into che gastro-intestinal tract.

8. Questions are open today regarding the uncertainty by which the internal contamination could be assessed and the meaning to be assigned to RLs in this assessment: they are not a tool for computing the dose resulting from internal

contamination, but are useful to verify che achievement of the optimized protection level. Efforts could be dedicated to interpretation of the monitoring results in order to reduce uncertainties; perhaps it would be worth to direct larger efforts to keep internal contamination well down, to levels where such uncertainties could be deemed as insignificant. This goal can be easily obtained by means of an appropriate and practicable choice of the sampling intervals, on the basis of detection limit of the measurement techniques.

9. The contribution of tails from previous exposures is also a constrain in setting the sampling interval T: if a formal Recording Level is adopted, as a fraction (f) of the Investigation Level (a value 1/3 is suggested by the ICRP), all the results of individual monitoring below the Recording Level shall be disregarded. As the exposure time increases, the tails from unrecorded intakes could build-up and lead, for some long-lived slow clearence radionuclides, at excretion or retention values comparable or larger than the DILs above discussed.

10. In this situation the DILs apply to the difference between the n-th result and the tails from previous intake I_k, i.e. the value of the ratio "f" between the Derived Recording Level and the Derived Investigation Level shall be lower than

$$F(T°/2) / \sum_{1}^{n-1}{}_{k} \; F\,(kT°+T°/2) \qquad (2)$$

Only a suitable, very low, value of (f) could solve in general this problem for some radionuclides (Table 1), by assuming as Recording Level the detection limit of the measurement technique.

11. As an other approach, all the results of individual monitoring are recorded, without adopting a formal Recording Level. For a selected individual, the result of the n-th control after the beginning of the exposure shall be corrected for the tails from the previous intakes, evaluated by means of the usual models. This methodology is highly individual related and is practicable only if computer based, as by using a computer on-line with the measuring instrumentation. Some efforts are dedicated in our organization to develop a suitable software for applying this methodology.

12. Some values for routine monitoring are shown in Table 1, where the ICRP value of 3/10 of the fraction of ALI corresponding to the sampling interval has been adopted as Investigation Level. The methodology can also be applied to radionuclides with recycling between the transfer compartment and the organ or

tissue compartments, as the three-compartment model for iodine, the Beach-Dolphin excretion functions for plutonium, the Balonov models for tritium.

Table 1
Urine DILs(/days) for inhalation (q= 3/10) and excretion tails (discrete and (continuous) intake)

Nuclide	T° (days)	DIL	number of periods "n" 3	6	12	30
^{60}Co	60	2.8 Bq	1,0(1,0)	1,7(1,7)	3,0(3,0)	5,1(5,1)
Class Y	90	2.5 Bq	1,3(1,3)	2,3(2,3)	3,7(3,7)	5,6(5,6)
^{125}I	60	50 Bq	11(11)	11(12)	11(12)	11(12)
Class D	90	66 Bq	6,3(6,9)	6,3(7,0)	6,3(7,0)	6,3(7,0)
U-nat	60	54 µg	50(50)	100(100)	200(200)	420(420)
Class Y	90	76 µg	75(75)	150(150)	280(290)	530(530)
^{239}Pu	60	190 µBq	94(95)	140(140)	190(190)	280(290)
Class W	90	240 µBq	100(100)	150(150)	210(210)	310(310)
^{239}Pu	60	23 µBq	19(19)	39(39)	83(83)	220(220)
Class Y	90	31 µBq	29(29)	61(61)	130(130	310(310)

13. For special monitoring DRLs refer to the unit of activity (ingested or inhaled), owing to the doubts about the use of ALI as an index of the committed dose following an acute intake. The applicability of the model to a specific contaminated person shall also be examined.

14. The problem of tails is not important for many radionuclides with a reasonably short effective retention time, as 137 Cs, 60 Co and 90 Sr, or in general for Class D and W radionuclides. For Class Y radionuclides and for a continuous intake, the contribution from the tails to the total urinary excretion can be comparable to the recent excretion when the systemic retention is not very short, compared to the sampling interval. For other radionuclides, as ^{239}Pu, the build-up of tails could be relevant and individual routine monitoring could be useful in providing information on the accumulation of internal contamination, rather than on recent intake, and thus in testing long term compliance with the selected level of protection.

EXCHANGE OF DOSE DATA WITHIN NUCLEAR ACTIVITIES IN FINLAND AND SWEDEN

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Abstract

In the Nordic countries, i.e. Denmark, Finland, Iceland, Norway and Sweden, only Sweden and Finland have introduced nuclear power into energy production. The first still operating nuclear power plant was commissioned in Sweden in 1972 and in Finland in 1977. In 1987, over 40 % of the total generation of electricity in both countries was generated by nuclear power.

It was soon noticed that there was a growing tendency that small groups of workers used to move at short notice between Finland and Sweden to work in the nuclear power plants in both countries during maintenance periods.

In 1983, the regulatory authorities for radiation protection, National Institute of Radiation Protection in Sweden and Finnish Centre for Radiation and Nuclear Safety in Finland, surveyed the radiation exposure to those workers. The authorities have brought about an arrangement by means of which the central dose data bases in the other country since 1984 have been able to record without delay the radiation doses received by her own citizens in the nuclear power plants of the neighbouring country. In addition, the authorities have confirmed the procedures of controlling dose data on workers from the neighbouring country, before those workers start working in a nuclear power plant regulated by the national authorities in question.

The paper describes the starting point of the activity, the established practice and the experience achieved. Until now, the practical experiences are positive. The total radiation exposure to the workers in the Swedish and Finnish nuclear power plants has been relatively low at each plant site. Thus, the main objective in the exchange of dose data, is to achieve a good radiation protection control.

Introduction

In Sweden there are 12 nuclear power plant units in operation; 9 BWR units of Swedish design and 3 PWR units of US design. In Finland there are four nuclear power plant units in operation; two BWR units of Swedish design and two PWR units of Soviet-Finnish design.

The methods of radiation protection applied at the nuclear facilities in the two countries are based on ICRP's recommendations and are quite similar in both countries. The activity is characterized by well-established active cooperation between both the Finnish and Swedish regulatory authorities and the persons responsible for radiation protection in the plants of the countries.

The Finnish and Swedish regulatory authorities for radiation protection have set forth a prerequisite that all persons working at the nuclear power plants shall use appropriate personal radiation dose meters. In addition, the personal dose data are required to be recorded and reported to the central dose data bases and authorities.

The dose meters and read-out units used in the Finnish and Swedish power plants are of the same type. The calibration and quality control procedures of the dose meters are approved and regulated by the authorities.

In Finland, the Finnish Centre for Radiation Protection and Nuclear Safety maintains a computer based national dose data base. The exchange of information between the central dose data base and the data bases at the two nuclear power plants takes place by reports, for the present.

In Sweden, the nuclear industry with its four nuclear power plants and the nuclear fuel factory have computerized the reading and transfer of dose data. From each site there is a direct connection to a computer based central nuclear dose data base.

Reporting personal dose data between Finland and Sweden

According to a mutual arrangement negotiated in 1983 by the authorities for radiation protection, the radiation doses received by the citizens of one's own country in the nuclear power plants in the neighbouring country, are recorded without delay in the central home data base. Certain country-specific features have an impact on the reporting procedure, which for the time being is as follows:

a) The selected sender in Sweden, the Forsmark nuclear power plant, shall report all occupational radiation exposures to Finns, entered in the central dose data base for the Swedish nuclear power plants, directly to the national dose data base at the Finnish Centre for Radiation and Nuclear Safety in Finland.

The report shall comprise dose records per month and it shall be transferred as soon as possible after the 10th day of the following month. Reporting takes place in the form of output data by mail. A copy of the report is transmitted also to the radiation protection authority in Sweden.

The output list shall, in accordance with the Swedish data base, include up-to-date personal radiation exposure, the identity number and the name of the employer, as well as data on the accumulated annual dose received in the Swedish nuclear power plants. In addition, information is given about passing medical examination including the examination date.

b) The senders in Finland, the Loviisa and Olkiluoto nuclear power plants, shall correspondingly on a monthly basis (a) transmit the dose data on all occupational radiation exposures to Swedes to the receiver in Sweden, Asea-Atom and to the Finnish authority. Asea-Atom is, thereafter, responsible for entering the data in the central nuclear data base in Sweden. The input is recorded as a personal dose received abroad.

The procedure involves certain specifications in order to, among other things, clarify the definition of citizenship and differencies in the identity numbers used.

Any other way of recording in the transmission of dose data between the countries must not be used.

Prodecures followed in dose inquiries

The radiation protection personnel at nuclear power plants shall prepare to maintenance outages. As part of this work, e.g. the previous radiation doses to the workers are to be considered.

The Finnish and Swedish nuclear power companies may inquire the facilities in the neighbouring country for particular questions concerning work and dose data of persons coming from that country.

Another way of finding out dose data concerning workers from the neighbouring country is that the nuclear power companies ask the contractors, before the work is started, for the latest recorded dose data. This is possible because all contractors operating in Finland and Sweden receive monthly or after their work is completed, a dose report concerning the nuclear power plant work of their employees.

If these two first methods, in some case, do not work, it is possible to ask directly the central dose data base in the neighbouring country (Finland-Sweden) for radiation doses of persons by giving the identity number and the name of the person.

Experiences learned from the procedure

The experiences show that the quick migration of workers between Finland and Sweden is primarily concentrated on certain nuclear power plants. Factors that have a permanent impact in this respect are, among other things, the location and design origin of the plant. Temporary factors include e.g. special work at some plant. The migration of workers is naturally peaked during the annual maintenance periods of the nuclear power plants.

The dose data transferred show that the collective radiation dose of citizens received in the other country has usually been under 0,1 manSv per year. The values, as regards Swedes, have been below 1 % and as regards Finns, below 5 % of the total collective radiation dose received at the same time in the nuclear power plants in the home country. About 50 - 100 workers from both Finland and Sweden have annually worked in the neighbouring country.

The highest individual doses received during one year in the neighbouring country have been as follows: 23 mSv (1982) and, after the arrangement was introduced, at its highest, 16 mSv (1986). The majority of the annual individual doses received in the neighbouring country have been, within the range of a few mSv. In general, the individual doses have been received in one or two months.

Discussion

The exchange of dose data on nuclear power plant workers between Finland and Sweden have systematically taken place since 1984. The exchange arrangement ensures that the individual dose limits are not unawares exceeded when workers quickly move between the neighbouring countries.

The workers, and also the associations representing them have taken a very positive attitude towards the arrangement. In preparing for the arrangement, it has been taken into account the legislative rules and regulations concerning the confidentiality of information on the persons in question. Contradictions with legislation have accordingly been avoided during the implementation of the procedure.

In addition to Finland and Sweden, there has not been a similar need for arranging a comparable procedure with other countries. The prerequisite for the exhange of dose data of this type is that the methods and follow-up systems of radiation dose monitoring are compatible and comparable with each other. In practice, this results in a requirement for high quality in the radiation protection procedures.

EXPERIENCE OF DOSE REDUCTION PROCEDURE USING TARGET DOSE MANAGEMENT

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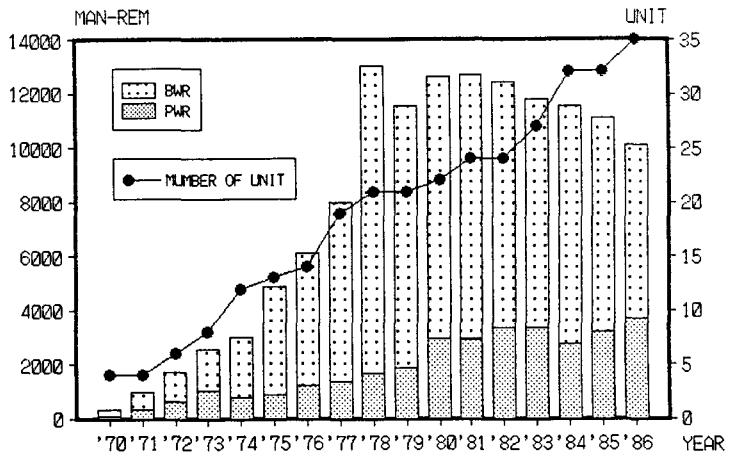
INTRODUCTION

In Japan, 35 units of nuclear power plant are now under operation. The exposure dose of plant workers showed a tendency toward increase year by year along with the increase of number of operating units. However, the tendency has shown a slight decrease of late (See Fig.1). The annual collective dose per reactor in 1986 was approx. 360 man•rem at BWR and approx. 230 man•rem at PWR. The dose could be limited to such a low level as a result of various dose reduction measures implemented in order at the existing plants as well as newly constructed plants. The efforts of concerned parties toward further reduction the dose and, at the same time, establishment of dose management and control system are requested.

Fig. 1 Trend of annual collective dose in Japan



DOSE REDUCTION MEASURES IMPLEMENTED AT TOKAI No. 2

At Tokai No. 2 power plant (BWR 1100MWe - start of commercial operation in 1978), various dose reduction measure have been positively implemented ever since the start of commercial operation, based on the actual results of preceding plants. These measures included crud reduction measures such as control of feedwater oxygen concentration, filling water during shutdown, introduction of CRD remote exchanging device, automatic ISI device, etc. and, furthermore, employment of such measure as improvement of the work environment including pipe shielding.

As a result of implementation of these measures, the collective dose at Tokai No. 2 power plant now ranks in a middle level among all operating BWR plants in Japan, and the dose at this plant keeps the same level rather than a slight increase.

The program for facility-wise measures necessary for reduction is now under study so that the dose at this plant can be further reduced. On the other hand, it is a matter of the first importance for reduction of the dose during the work to implement the dose management and control system, so that radiation

protective measures can be systematically and surely executed from the work planning stage.

On basis of such a viewpoint as stated above, arrangement and strengthening of target dose management organization and method have been recently executed at Tokai No. 2 power plant. The current status of the execution is as follows:

DOSE REDUCTION BY TARGET DOSE MANAGEMENT

a. Establishment of ALARA Coordinate Group

ALARA Coordinate Group (ACG) was established at site from the 7th Annual Maintenance Outage for the purpose of collectively control and reduce the dose during the annual maintenance work, which shares a greater part of the annual dose. ACG consists of the members of the plant and contractors assigned for radiation control and work management sections, and the meeting of ACG is held periodically starting from about 2 months before the annual maintenance work begins and throughout the maintenance work.

An objective of ACG activity is to introduce target management and connect result of the management with the actual dose reduction in the work control in accordance with the Plan-Do-Check-Action procedure and, at the same time, to promote ALARA philosophy through its activity.

The target management concept has been formally incorporated in Quality Contorol and Quality Assurance activity of general industrial field, and it is a common procedure used in the QC circle in our country. ACG just applied its concept into the dose reduction measures.

b. Scope of ACG Activity

The following activities are performed by ACG at each step from start of maintenance work and after completion of maintenance work:

Prior to start of maintenance work, anticipated man·rem for the whole work is calculated and the calculated result is studied and evaluated from a viewpoint of ALARA program (according to the "ALARA Check List"). In particular, for the work items involving the total dose of 5 man·rem or more the dose reduction measure items for each work step are studied and evaluated (See Flow Chart 1). The target value and weekly target value assigned to each contractor by each wrok item are established, based on the result of study mentioned above.

During maintenance work, weekly meeting is held for checking the actual result against target value and recommendation and warning for dose reduction purpose are given to contractors as necessary so that they can be reflected into such actions as flushing, decontamination, etc.

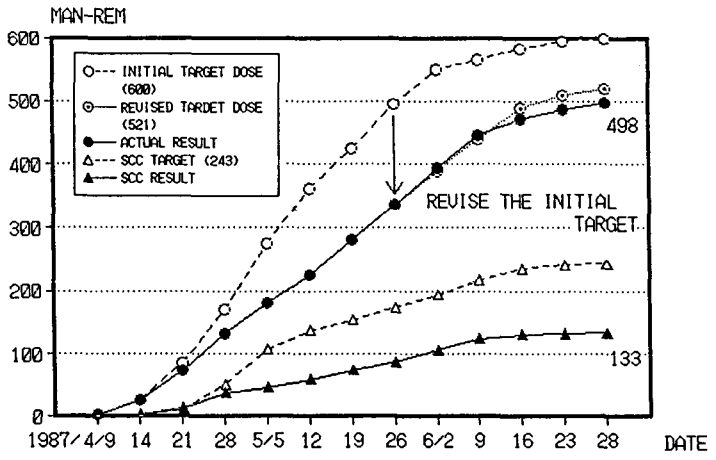
After completion of maintenance work, execution status of ALARA program is confirmed in order that the actual status can be reflected into the next maintenance work. In particular, for the work item which involved 5 man·rem or more, evaluation of comparison between target dose and actual dose as well as the difference thereof are studied in detail (See Flow Chart 2). In addition, for the work to be performed every year, the model cases have been properly arranged and consolidated in order for standardization of ALARA techniques.

c. Actual Result of Dose

Fig. 2 shows the actual dose received during 8th annual maintenance outage. For this maintenance work, ALARA study was made on the initial value of 650 man•rem estimated based on the past actual results and, as a result of study, a target value of 600 man•rem, which was about 10 % less than the estimated value, was established. The actual dose however was 498 man•rem, which was far less than a target value. This was because the effect of dose reduction measures applied to the work of SCC countermeasures, which was the largest modification work item done during this maintenance outage, was far greater than expected. These measures included chemical decontamination of the pipe performed prior to the pipe replacement work and, as a result, the dose received during pipe cutting and removal work was successfully reduced to a low level. In a midway of the maintenance work, a target value was revised to a lower value of 520 man•rem, but the actual result was far below even the revised value.

Evaluation of results of each of dose reduction measures applied during this maintenance outage indicated such effects as reduction of 19 man•rem by installation of a temporary shielding and reduction of 11 man•rem by adjustment of work location, in addition to reduction of 120 man•rem resulted from chemical decontamination mentioned above.

Fig. 2 Accumulated exposure dose in Tokai No.2 8th annual outage

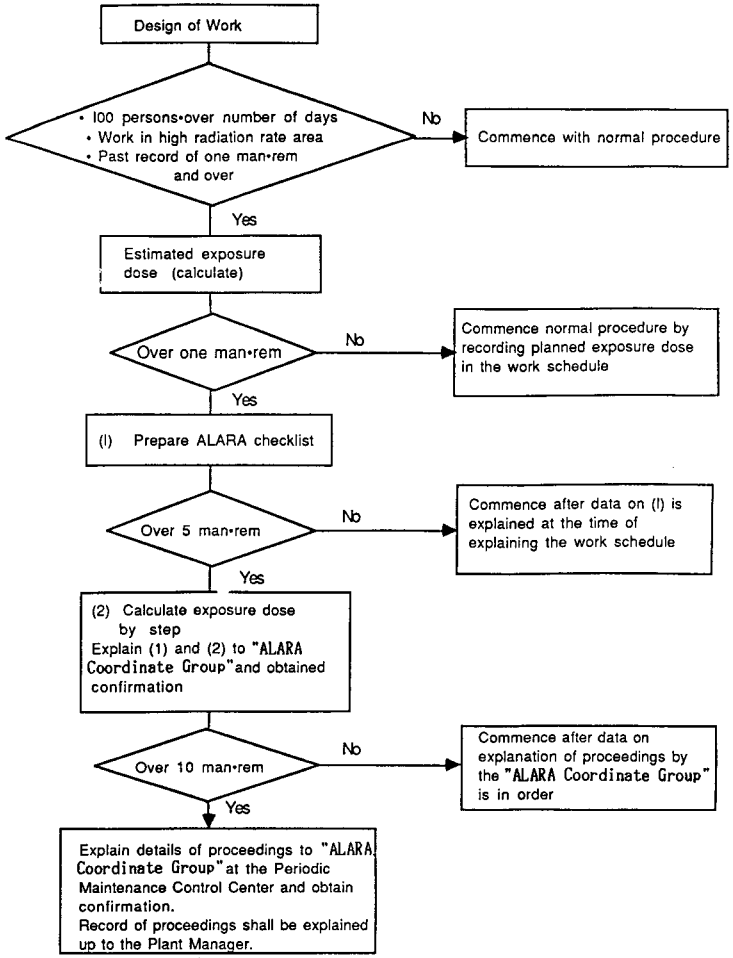


CONCLUSION

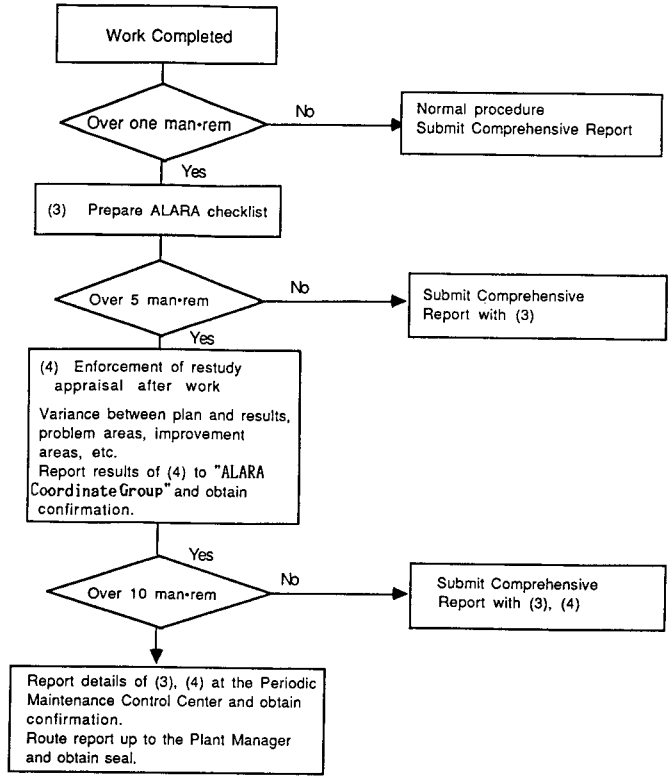
Though estimation method of the dose in special cases like a large plant modification work remains unresolved, expected effect of dose reduction measures is obtained through the work management, by setting up a target value and performing careful and refined integrated dose control. It is of extremely importance to firmly fix and maintain such dose management and control system in the future.

Further, in order to drastically reduce the dose to a half or more of the current dose, we are planning to set the long/medium - term target dose and application programs of necessary measures, such as installation of permanent and/or temporary shielding for primary loop recirculation piping, duplexed condensate cleaning system using hollow fiber filter, etc.

Flow Chart 1 ALARA plan procedure before initiating work



Flow Chart 2 ALARA plan procedure after work



REDUCTION OF OCCUPATIONAL DOSE USING THE UKAEA SIMVIDOSE SYSTEM

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INTRODUCTION

The ICRP recommendations emphasise the need to keep doses as low as reasonably achievable. In order to make the best uses of resources in reducing operator dose it is important to know accurately how the dose is incurred. To allow the radiological protection staff to link dose rate with the particular task being carried out the SIMVIDOSE (SIMultaneous VIdeo and DOSE) system was developed (reference 1). SIMVIDOSE allows the external gamma dose rate to be displayed on a video image of the worker involved.

By recording the combined display the exact details of the exposure can be seen and protection measures targeted to the maximum effect. Experience with a development system has demonstrated has demonstrated its use as a training aid and/or on the spot dose control.

THE SIMVIDOSE EQUIPMENT

The monitoring consists of a GM tube and radio transmitter fitted in a small portable unit worn by the operator. The pulses are transmitted to a base unit with a countertimer module which converts count rate to dose rate and superimiposes a character string of time, dose rate and total dose on the video display from the camera.

In early versions (figure 1) a black and white camera was used. The only record of the dose rate was on the video recording. While this was useful in a qualitative way it was tedious to recover the data for quantitative studies.

A serial interface was added which enabled the data to be collected on a portable microcomputer and recorded on tape. Simple graphs of dose rate against time were then produced using either a printer or a graph plotter.

In low contrast situations a considerable advantage is offered by colour. Unfortunately the equipment had been designed for a camera with external synchronisation. Domestic colour cameras do not offer such a facility but an alternative system using a BBC microcomputer was available. The data from the counter timer is sent via the serial interface to the BBC microcomputer which saves the data and generates the numerical display which is then added to the video image (figure 2). Periodically the dose data is transferred to a floppy disc.

USE OF SIMVIDOSE

Throughout the development of the equipment trials have been carried out on active work. Both routine and one-off tasks have been studied.

Routine Operations:

(a) Silicon Ingot Irradiation. Pure silicon ingots are irradiated in the DIDO and PLUTO reactors at the UKAEA Harwell Laboratories to produce P type doping. Both vertical and horizontal irradition facilities are used. Unloading of irradiated ingots from the vertical facility is carried out by winching the can containing the ingots into a flask. The ingot is unloaded in an active handling cell, a new can is loaded into the flask and transferred to the reactor.

Slight differences exist between the equipment and operations on the two reactors. On Pluto the flask was located over a collar placed on the reactor top and the operator approached the flask during the process. The complete operation took 22 minutes but 60% of the dose was received in the 2½ minutes when the flask was being loaded with the irradiated ingot. On DIDO the flask locates in a larger collar on the reactor top. Transfer operations were observed from behind a shielded screen. In this case the higher doses were incurred in the transfer to the active handling cell (25% of the total), although the total dose was only 70% of that on the other reactor. The dose involved in unloading irradiated silicon from the horizontal facility on the PLUTO reactor and loading a new ingot was about double that for the vertical facility. The whole operation took about 35 minutes. Manipulation of irradiated ingots accounted for 45% of the dose and manipulation of new ingots for 25%. Both operations were carried out within a shielding tank, working from above.

These studies identified the parts of the operation giving most of the dose. While this could have been done using hand held instruments the results would not have been so easy to interpret and extra exposure would have been incurred.

(b) Cropper and Dissolver loading. The spent fuel from the DIDO and PLUTO reactors is reprocessed at the UKAEA Dounreay site. Operations involve the use of long tongs to manipulate elements within a pond into a cropping machine and then into the elevator to the dissolver cell. Results showed a fairly constant dose rate although there were increased levels when sediment was disturbed. In this case the only way to reduce the operator dose was to reduce the overall background level by reducing the residual contamination in the pond.

One-off Operation - Boiling Tube Replacement

During the annual shutdown of the SGHWR (Steam Generating Heavy Water Reactor) at the UKAEA Winfrith establishment a section of one of the boiling tubes below the reactor was replaced to allow tests for hydrogen embrittlement to be carried out on it. Dose rates in the lower lagging box where the tubes run are high due to "crud" deposition.

The camera was mounted on a remote control pan-tilt unit fitted to a specially fabricated bracket in the lower lagging box. The rest of the equipment was set up outside the biological shield. The calibration was adjusted to match the personal radiation monitors worn by the operators.

The SIMVIDOSE equipment was set up in a low dose rate area which effectively served as a control room for the operation. The work of removing the tube section and fitting a new piece was carried out by contractors supervised by UKAEA personnel. A number of entries were required. The use of SIMVIDOSE enabled health physics staff to continuously monitor the dose rate to which the worker was exposed without incurring dose to themselves.

Contractors about to take over could see recordings of earlier entries and locate high dose rate areas. Modifications could then be planned to take account of the dose rates.

At one point the dose rate increased sharply when a bung was removed from the cut pipe. However, the dose rate fell off quickly after an initial peak and the health physics staff allowed the worker to remain avoiding the extra dose from another entry.

It was also valuable to know when the operational dose control limit was being approached so that the worker to be instructed to stop at a convenient point in the job. In one such instance a bolt on a restraining bar broke, the decision was made to cut the entry short and leave the repair for a further entry as insufficient margin was available on the dose limit for the entry.

FURTHER DEVELOPMENTS

The SIMVIDOSE system, as it stands, is very much a development version. Some problems have been encountered with loss of signal from the dose meter. Interference from for example, overhead cranes can give rise to spurious results. The current design also precludes the use of more than one detector. In addition the BBC microcmputer is slow by modern standards and lacks the power and memory of later machines. One possible approach which is being studied (reference 2) is to use dosemeters which record total dose. Each dosemeter is polled by the controlling computer via a "transparent" radio data link. The personal dosemeter transmits the total dose. The current dose rate is then reconstructed by the computer. As the transmission is digital the likelihood of interference is considerably reduced and any loss of data can be recovered on the next transmission. Although this system is intended for emergency use without simultaneous video it will be possible to adopt it for this use.

Another possible development is to combine SIMVIDOSE with an automatic location system which would enable a map of dose rate to be produced for the area under investigation.

CONCLUSION

Simultaneous display and recording of dose rate together with a view of the operations being undertaken has enabled protective measures to be applied effectively to routine tasks. In addition the equipment has proved useful in controlling doses during individual operations and as a training aid.

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2. UK Patent Application 145254. Monitoring of personnel in hazardous environments. S F Hall.

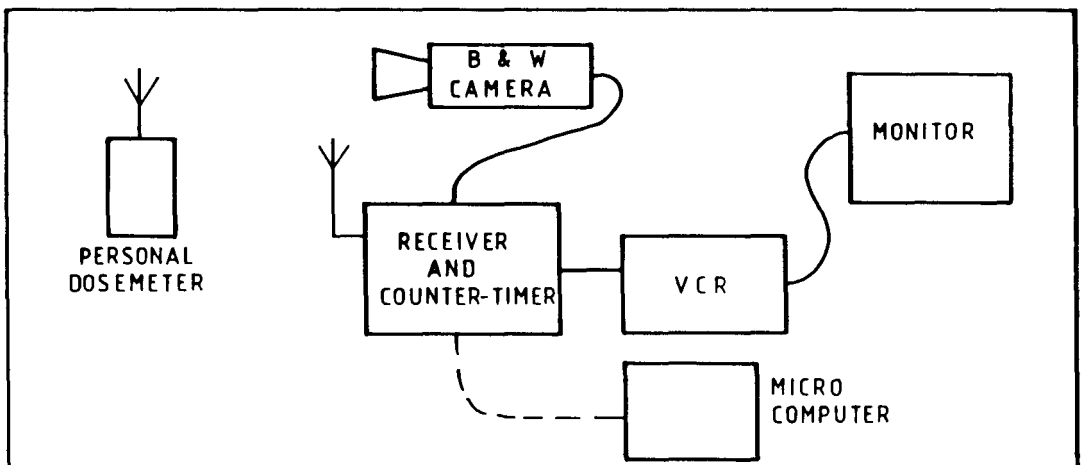

FIGURE 1 SIMVIDOSE VERSION 1

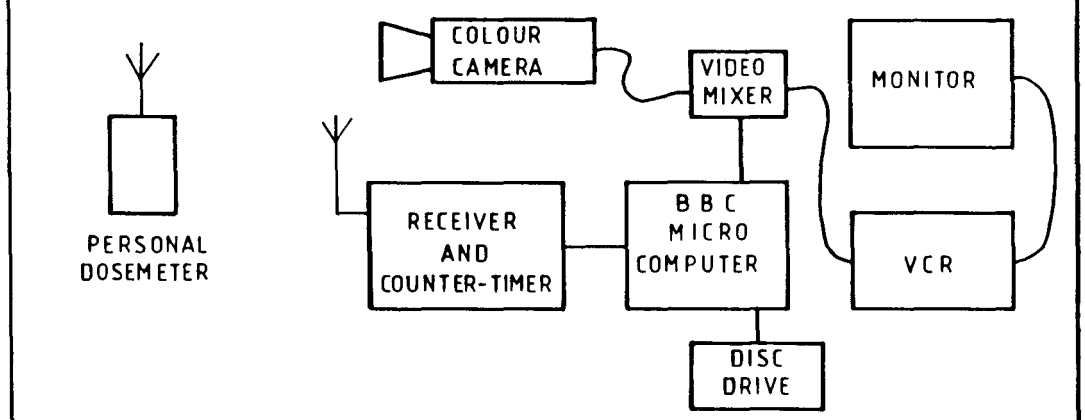

FIGURE 2 SIMVIDOSE VERSION 2

THE CURRENT STATUS AND IMPROVEMENT ON THE REGISTRATION SYSTEM FOR RADIATION WORKERS AT THE NUCLEAR FACILITIES IN JAPAN

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ABSTRACT

The number of radiation workers engaged in the nuclear field in Japan has already reached about 210,000, and that of nuclear facilities 31. As a considerable number of these workers are doing their work in more than one facility, it is requested for us to certify radiation workers and manage their records in one system throughout the country.
For those purposes the Registration System for Radiation Workers and the central office for it were established in 1977.
The current status and a new improvement scheme are as follows.

1) current status of the system
   A) main activities
      a) registration of workers
      b) issuance of radiation passports
      c) registration and keeping of the personal dose records
      d) response to inquiries about the records
   B) participating organization
      a) nuclear laboratories
      b) nuclear Power stations
      c) nuclear fuel fabricating companies
      d) maintenance contractors for nuclear facilities
   C) problems
      Telephone, facsimile and mail are used for the communication between each facility and central office in the system now, so
      a) it takes a lot of time to answer to inquiries, and
      b) the dose data for short period-for instance, monthly date-havn't been dealt with.

2) improvement scheme
   A) establishment of an on-line network system which enables each terminal computer to have direct access to the master computer.
   B) issuance and maintenance of radiation passports by the terminal computers installed at facilities.

SYSTEMATICS OF GAMMA-RAY ENERGY SPECTRA FOR CLASSIFICATION OF WORKPLACES AROUND A NUCLEAR FACILITY

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1. INTRODUCTION

Radiation dosimetry in workplaces has been carried out both for assurance of the doses complying with the acceptable values and for improvement of protection methods to minimize detriments of the exposed population. This means that it is very impotant not only to determine dosimetric quantities in workplaces but also to know features of radiation fields because information for radiation protection can be often derived from the radiometric quantities.

Classification of workplaces based on the fearure of gamma-ray energy spectra is one of practical ways to realize radiation protection being taken into consideration of the radiometric quantities. Furthermore, demarcation of workplaces based on these radiometric quantities may be effective for improvement of radiation protection practice such as estimation of radiation doses, designing of radiation shields and other activities.

From these points of view, gamma-ray energy spectra have been determined in various workplaces in nuclear facilities, and systematics of gamma-ray fields were tried for classification of workplaces on the basis of the feature appeared in health physical quantities such as effective dose equivalents and reponses of dosemeters.

2. EXPERIMENTAL

2-1 Measuring System and Measurements

Gamma-ray energy spectra in some workplaces of the Research Reactor Institute of Kyoto University were determined by the scintillation spectrometer with a 7.6 cm diameter and 7.6 cm long cylindrical NaI(Tl) crystal covered with a 2 mm thick bakelite cover. The scintillation detector was supported with a tripod about 1.2 m high. Data from the detector were recorded in the multi-channel pulse-height analyzer. Pulse-height distributions were analyzed into gamma-ray energy spectra with an unfolding technique using mini-computer system.

Workplaces chosen for measurements of gamma-rays were (1) TL (laboratory for physical, chemical and biological treatment of low level radioactive materials), (2) HL(laboratory chemical processing of high level radioactive materials) and (3) RR(reactor building containes the light water cooled reactor and related experimental equipments). Measurements were carried out at 19 places in these laboratories.

2-2 Determination of Dosimetric Quantities

Dosimetric quantities discussed here were exposures(Ex), effective dose equivalents(EDE) and ambient dose equivalents(ADE). Ex could be determined after well-known formula using gamma-ray energy spectra and energy absorption coefficients of air. EDE and ADE could be derived from the exposure multiplied by the Ex to EDE and ADE conversion factors, respectively/1,2/. Responses of a Geiger-Muller serveymeter and a filmbadge dosimeter were calculated using energy dependences of these counter and dosimeter sensitivities/3,4/.

3. RESULTS AND DISCUSSION

3-1 Gamma-ray Energy Spectra

Examples of gamma-ray energy spectra obtained in TL and HL are illustrated in Fig.1 and Fig.2. Broken histograms in these figures show the energy spectra due to natural radionuclide in the laboratory. It is evident from these results that the artificially

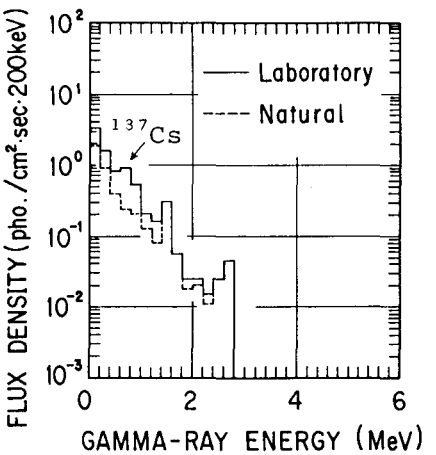

Fig.1 Gamma-ray energy spectra around chemical laboratories.

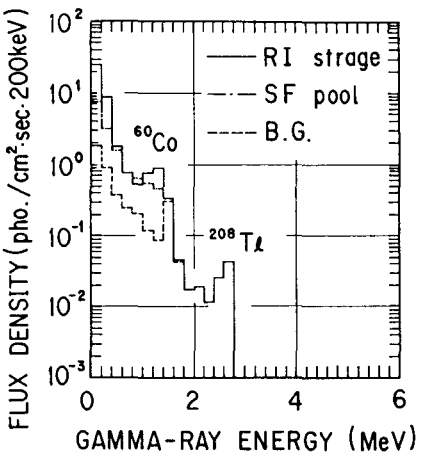

Fig.2 Gamma-ray energy spectra around RI/SF strage facilities.

increased gamma-rays, which are attributable to radionuclides such as ^{60}Co and ^{137}Cs, are observed below 1.4 MeV and that intense low energy photons are observed near the strage facility for radioactive isotopes or spent fuels of the reactor. On the other hand, as illustrated in Fig.3, higher energy gamma-rays are observed aound a neutron source strage facility. This seems to be caused by an Am-Be neutron source which emits about 4.4 MeV gamma-rays as well as netrons after nuclear reaction of $^{9}Be(\alpha,n)^{12}C$.

Examples of gamma-ray energy spectra observed in the reactor building are illustrated in Fig.4. Solid histogram shows the energy spectrum obtained during reactor operation at the power of 5 MW and broken one shows a background gamma-ray energy spectrum. It is apparent from these histograms that gamma-rays in the reactor building were mainly caused by neutron capture reactions of argon,

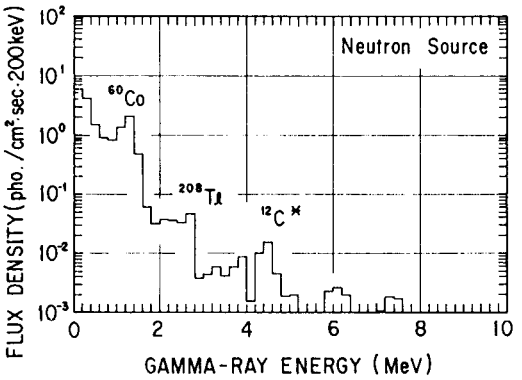

Fig.3 Gamma-ray energy spectrum around a neutron source strage facility.

hydrogen, iron and other nuclides, and that they distributed up to energy about 9.2 MeV. Gamma-ray energy spectra obtained at basement of the reactor building is illustrated in Fig 5. As shown by the solid histogram in Fig.5, gamma-rays from ^{16}N created by ^{16}O(n,p) reaction in the coolant circuit and from ^{41}Ar leaked from the reactor room of the building are clearly observed by the operation of the reactor.

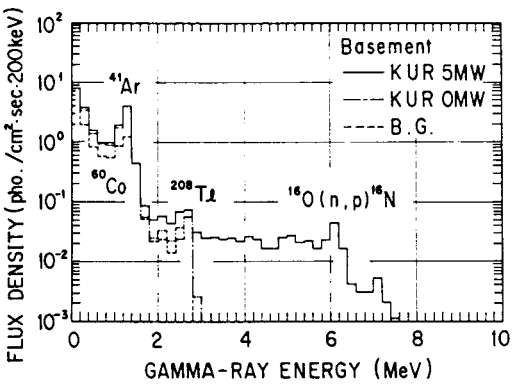

Fig.4 Gamma-ray energy spectra in workplaces around the KUR.

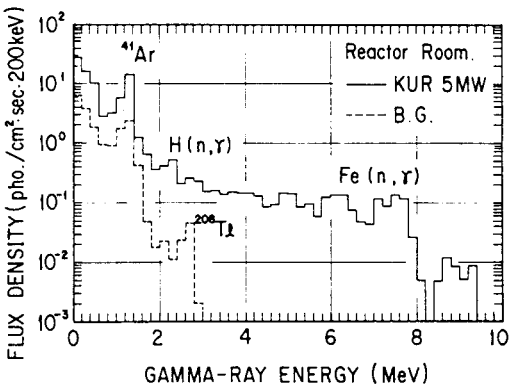

Fig.5 Gamma-ray energy spectra around coolant of the KUR.

3-2 Relation between dosimetric and radiometric quantities

Ex rates, EDE rates and ADE rates were determined on the basis of the energy spectra of workplaces and serveymeter responses or filmbadge readings were determined by accumulating exposures multiplied by the detector sensitivity. To discuss relations among these health physical quantities regardless of the total exposure rate, each dosimetric quantitie was divided by the Ex rate. The results are shown in Table 1. The number in each line was given by averaging over several results obtained at workplaces where gamma-

Table 1. Ratios of dosimetric quantities and dosimeter responses to exposure.

workplace	energy	EDE/Ex	ADE/Ex	film/Ex	GM/Ex
RI/SF strage	<0.4 MeV	0.90	1.14	0.44	1.34
chemi.res.labo.	<3 MeV	0.87	1.06	0.53	1.28
reactor room	>3 MeV	0.86	1.03	0.57	1.34

ray energy spectra showed similar characteristics. Typical workplaces and some features appeared in gamma-ray energy spectra are shown in the first and second column of the table. It is clear from the two dosimetric quantities in the table that the ratios shown in the first line is markedly defferent from the other ones and that smaller differences can be seen between ratios obtained in the medium and higher energy gamma-ray fields. Furthermore, it is to be noted that the ratios in the third and fourth columns become small with increasing maximum photon energy in the workplaces contrary to the feature appeared in the results in fifth and sixth columns. Though the difference of the minimum and maximum values in each column are within 30 %, it is very important to realize that low energy photons give larger Ex to EDE or ADE conversion factors and they can be shielded by a simpler protective way and that dosimetric quantities or detector responses show complicated charactors in higher energy gamma-ray fields. These two facts indicate that an application of a unique method is not appropriate to get reliable information for radiation protection This means that classification of workplaces based on the feature of the energy spectrum will be powerful for establishing reasonable methods for radiation protection.

4. CONCLUSIONS

It becomes clear from the investigation that gamma-ray fields can be classified into three groups: (1) low energy gamma-ray fields mainly composed by scattered photons, (2) medium energy gamma-ray fields formed by utilization of radioactive materials and (3) high energy gamma-ray fields due to neutron capture in mixed radiation fields. It is also quite possible that radiation protection practice will be reasonably improved by the demarcation of workplaces on the basis of these feature of gamma-ray fields.

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CHARACTERISTICS OF INTERNAL AND EXTERNAL EXPOSURES TO WORKERS BY JOB IN JAPAN ATOMIC ENERGY RESEARCH INSTITUTE

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INTRODUCTION

It is useful to investigate the characteristics of exposure to radiation workers by job category for the reassessment of the appropriateness and effectiveness of monitoring method and radiation protection practice. As a first step, we have analyzed annual collective dose to workers in JAERI by job category and type of exposure for the years of 1969-1986.

INDIVIDUAL MONITORING IN JAERI

For external exposure, film badge is issued every 3 months to all radiation workers. For internal exposure, monitoring is carried out by bioassay and external counting as follows [1]:

- Routine or confirmatory monitoring: for subjects selected from specific worker groups according to the possibility of internal contamination, by an annual survey on working condition,
- Special monitoring: including internal dose assessment, for workers supposed to be contaminated or exposed, as a rule, above 0.1 mSv of committed dose equivalent in any organ.

CLASSIFICATION OF WORKERS BY JOB

Workers are classified into 8 categories as follows:

a) Reactor operation: operation, utilization and maintenance of nuclear reactors,
b) Hot laboratory management,
c) Weste treatment: including the workers attended to decontamination of instruments, equipments and hot laboratories,
d) Radiation control,
e) Radioisotope production,
f) Research and development: relating nuclear reactor, accelerator, safety engineering, materials, nuclear fuel, radiation chemistry, physics, chemistry, fusion, etc.
g) Others: administrative and engineering services, trainees,etc.

YEARLY VARIATION OF THE NUMBERS OF WORKERS IN JOB CATEGORIES

The number of workers tends to increase, mainly due to development of fusion research, nuclear safety research, reactor fuel examination and decommissioning of a power demonstration reactor, that reflects in the increase of workers in categories c) and f). The numbers of workers are shown in Fig.1.

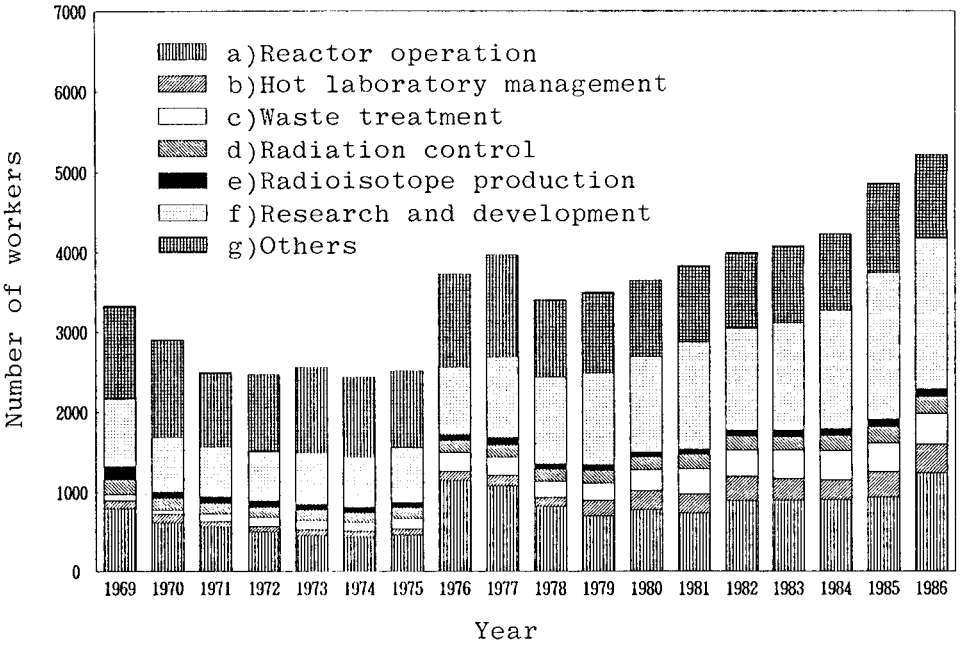

Fig.1 Yearly number of radiation workers.

CALCULATION OF COLLECTIVE DOSE

(1) External exposure

Effective dose equivalent was assumed to be equal to the dose estimated from the film badge reading. Collective dose to workers of each category was calculated by summation of individual doses above the minimum detectable dose of 0.2 mSv.

(2) Internal exposure

Intake of radionuclides of workers was estimated by bioassay and/or external counting. Annual collective dose was calculated by summation of committed effective dose equivalents above 0.1 mSv.

VARIATION OF ANNUAL COLLECTIVE DOSES

(1) External exposure

Variation of annual collective doses by job for years 1969-1986 is shown in Fig.2. The collective dose was the largest in category a), and the second largest in category f). The collective doses were larger before 1973 because of special jobs of maintenance of heavy water research reactors. After 1973, the annual collective dose has been relatively constant. In 1986, the annual collective dose is nearly equally shared by 4 categories: a), b), c) and f).

(2) Internal exposure

Fig.3 shows the annual collective committed dose equivalent to 38 workers among the total of 1,500 subjects of special monitoring. As for the result of routine monitoring of 18,300 measurements, only 7 subjects were found to be exposed to above 0.1 mSv

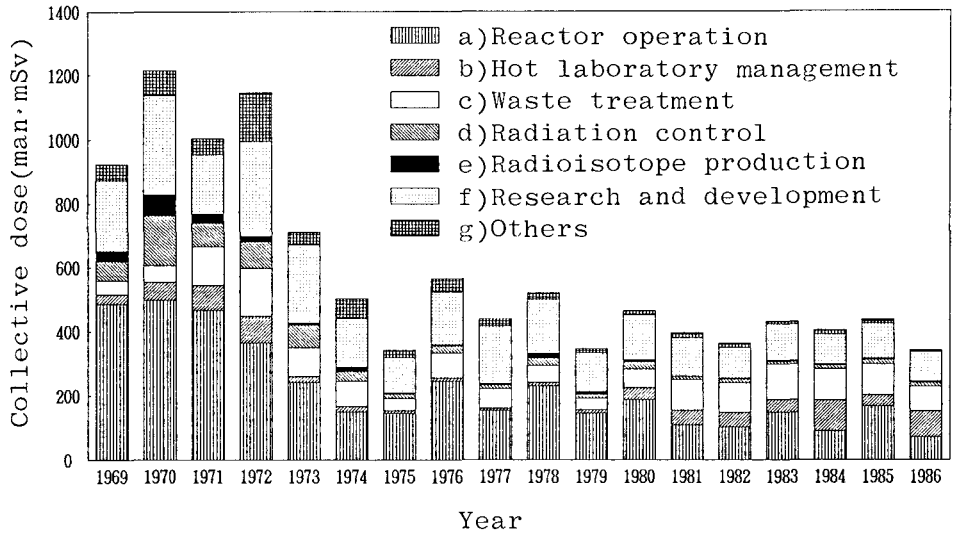

Fig.2 Variation of annual collective dose for external exposure.

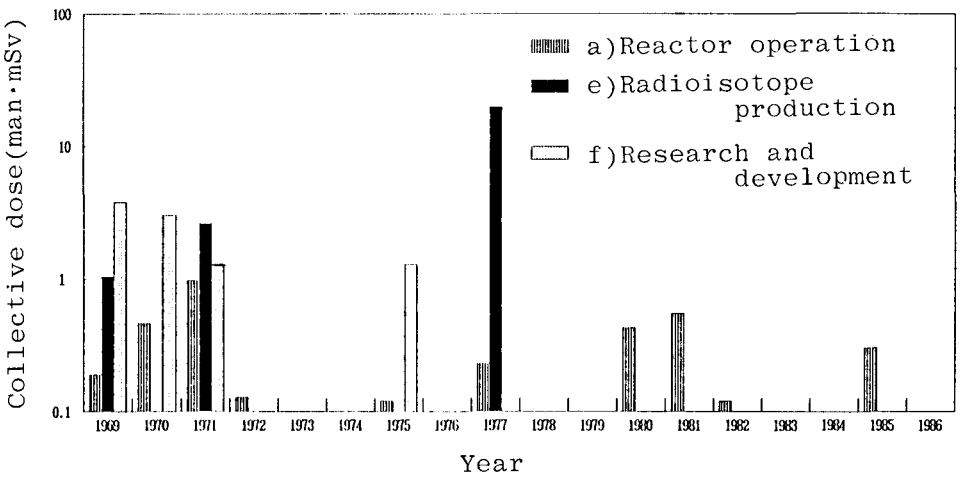

Fig.3 Annual internal collective doses for job category.

of effective dose equivalent, which are included in the collective dose. Before 1973, internal exposure occurred almost every year in 3 categories a), b) and f). After 1973, internal exposure occurred less frequently. The exposure of the year 1977 in category e) was an Am-241 inhalation case, 20 mSv received by a subject.

COMPARISON

Fig.4 shows yearly variation and ratio of internal and external

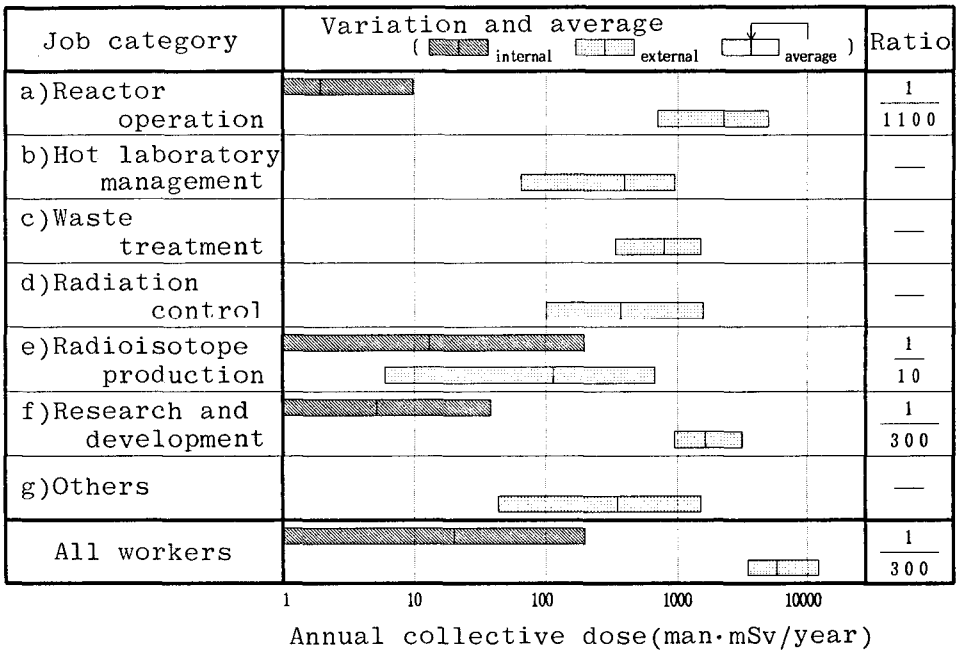

Fig. 4 Comparison of internal and external annual collective doses for job categories.

annual collective doses in each category. Each box represents the range of annual collective dose variation. The left and right ends of the box are minimum and maximum, respectively, and the average is shown with an interposed line. As shown in the figure, the ratio of internal to external collective doses in category e) was the largest, that was 1:10, to which the Am-241 case in 1977 made a dominant contribution. Internal collective dose was by 300 times smaller than external collective dose for all radiation workers in JAERI for years 1969-1986.

CONCLUSION

We presented a retrospective analysis of annual collective dose with special emphasis on comparison of internal and external exposures. We could characterize the working condition by job more clearly, that may be of some use in planning a monitoring program. A small proportion of internal exposure to external exposure in collective dose can be interpreted as a result of radiation protection practice in JAERI, that is, unsealed radioactivity is enclosed as far as possible to minimize contamination of workers and work places.

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PERSONAL COMPUTER INTEGRATED PROGRAMS - EFFECTIVE TOOLS FOR PLANNING COMPREHENSIVE RADIOLOGICAL SURVEYS

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Radioactive material processing facilities now abandoned or placed on inactive standby require periodic maintenance and surveillance. Institutional arrangements frequently become ineffective and result in the migration of radioactive materials to areas beyond site boundaries. In order to comply with current regulations regarding environmental degradation, various response actions are available to correct and limit future migration. Decisions on specific remedial measures must be based on sound engineering evaluations. In addition to site specific physical features, these engineering studies require, as input, a thorough description of the areal distribution of radioactivity outside site boundaries as well as a description of the radioactive source term. These data can only be obtained through the completion of a comprehensive radiological survey of facility (and vicinity) grounds, buildings, process equipment, sewers, sumps, tanks, drains, and facilities used to handle and store residual radioactive wastes. For facilities with 20 - 50 buildings and which cover over 100 acres, a comprehensive radiological and chemical characterization represents an immense undertaking. Determining an optimum distribution of labor and other resources can be a difficult and time consuming task. The purpose of this paper is to describe the use of an electronic spreadsheet to determine, on the basis of prescribed measurement and sampling intervals, parameters such as total number of measurements and samples, total time required for specific tasks, the total time in days, given the number of personnel available and realistic performance efficiency, and an estimate of the cost.

One of the most innovative and useful developments in personal computer software is the integrated program. For this work, the author chose Symphony, by Lotus Development Corporation, and all work was performed on a COMPAQ Portable Computer. This combination can be used by the novice with limited experience. Instruction may be accomplished in a few hours through review of Symphony documentation (Refs. 1,2) and other tutoral manuals (Ref. 3). Symphony's spreadsheet module is a powerful tool, an electronic grid of columns and rows which is ideal for planning sequential or integrated activities. In addition to the spreadsheet, word processing and graphics environments in Symphony offer flexibility in data presentations. Specific mathematical relationships between tasks and required labor may be applied to individual cells formed by the intersection of rows with up to 256 columns. There are a total of 8192 rows, hence a fully utilized spreadsheet would contain over 2 million individual cells.

Before one can apply the spreadsheet to the requirements of an actual survey, a site specific characterization plan must be pre-

pared. Such a plan requires the careful review of historical information as well as an evaluation of data obtained during site visits (Ref. 4). The scope of any characterization is influenced by project objectives, existing criteria for measurements and sampling, and available labor forces. Electronic spreadsheets facilitate the coupling of these latter considerations with facility size to establish overall project scope and budget. Furthermore, the spreadsheet may be used to monitor project performance so as to predict non-conformance with cost and schedule criteria.

Symphony was used to plan an actual survey of a facility formerly used to process radioactive material. For this discussion, the facility will be treated generically. It consisted of 33 structures with moderate to high levels of surface radioactivity (Category M,H) and 14 structures thought to be free of radioactive contamination (Category N), all situated on a fenced site of about 150 acres. The intervals for measurements and samples in the two categories were dictated by respective measurement and sampling criteria and were tied to a fixed grid system on floors, walls, ceilings, and flat roofs. Results of the survey of Category M,H structures were to be used as an aid in planning remedial activity and to establish areas where specific controls would be required to protect workers and to prevent the spread of radioactivity. For this reason, survey intervals were larger than for Category N where results could be used to base decisions regarding the demolition and disposition of those buildings. In addition to a greater number of measurement points per unit of inside area, the outside walls of Category N structures required survey. The spreadsheet was also used to plan measurements and sampling (surface and subsurface) activity on open land at specific points on a site-wide grid system.

Following is an explanation of the spreadsheet organization and execution for measurement and sampling activity in Category M,H structures. Column headings accross the spreadsheet's width provided details of each structure, specific tasks to be performed, the number of person-hours required to complete each task, the total person-hours for groups of tasks, and the total person-hours for each structure. Columns of the spreadsheet are identified by letters, and rows by numbers thus giving each cell an alpha-numeric code. For Category M,H the sheet was composed of 61 columns and 34 rows (2074 cells). Manual entry was limited, for the most part, to column headings, structure number in Column A, structure description in Column B, floor area in square meters in Column C, wall area in Column Z, and formulae for mathematical relationships in other cells along the first row of data. In a few cases, certain other data required manual entry because no real relationship existed with respect to building size. For example, the time required for a pre-entry radiation survey was a variable function based on the building's physical condition (some with no lighting, some with structural damage, etc.). The number of equipment items to be surveyed, the number of sumps and floor drains, and the time required for personal protective measures varied from building-to-building. An estimate of the number of person-hours for each of these was

made initially and updated as necessary to provide an accurate estimate of total labor hours. Once the number of above items were determined and entered manually, sheet calculations were used to estimate the number of measurements, samples, and respective labor. Individual tasks associated with floor area were as follows:

- Number of grid points to be marked a 2 m intervals
- Number of directly measured alpha readings at 2 m intervals on floors
- Number of directly measured beta readings at 2 m intervals on floors
- Number of pressurized ion chamber readings (1 m above the floor) at 4 m intervals on floors
- Number of bulk samples collected at 6 m intervals on floors
- Number of swipe samples at 4 m intervals on floors.

Tasks associated with wall area included:

- Number of grid points to be marked at 4 m intervals
- Number of directly measured alpha readings at 4 m intervals on walls
- Number of directly measured beta readings at 4 m intervals on walls
- Number of swipe samples at 4 m intervals on walls.

Ceiling and floor areas were the same for each building level, therefore ceiling grids were not established physically. Measurements and swipe samples were based on the floor grid, but with intervals of 4 m. Measurement and sampling intervals for roofs (flat roofs only) were the same as for floors, and grids were installed.

Structures in Category M,H contained areas (in square meters) of 51,000 for floors and ceilings, 47,300 for walls, and 28,300 for flat roofs. In addition, these 33 structures contained over 1400 pieces of equipment and 150 floor sumps and drains. Once the column headings were entered, column width was established and the cells in each column were formatted in accordance with the type data to be included (for example, fixed decimal, number of significant figures, currency, etc.). Starting with the first row of data (call it Row 10) in the spreadsheet, move right to Column E where the heading reads "Person-hours to Grid Floor". On the video screen, Cell E10 will be highlighted. Either a value or a formula must be entered in this cell. Because establishing a floor grid is related to floor area, a formula is typed and entered. Two people are needed to measure grid intervals, mark grid intersections, label the the grid, and document the grid on drawings. Approximately 1.5 minutes are required per grid node at 2 m intervals. The actual formula is as follows:

$$(C10/4)*1.5*2/60$$

where C10 is a cell which contains the floor area in square meters, 4 refers to 1 grid intersection per 4 square meters, 1.5 minutes to install each grid node, 2 persons, and 60 minutes per hour. The same formula applies to all 33 structures, therefore it is required

in all Column E cells in Rows 11 - 42. The formula need not be entered manually however. Using Symphony's Copy Command, the task can be accomplished with 7 key strokes, or by executing a previously determined macro, a procedure which Symphony can perform automatically upon command. As one progresses through the spreadsheet, similar procedures are applied to the leading cell in each column and copied to cells in Rows 11 - 42. In addition to determining the number of alpha, beta, and gamma-ray measurements, the number of swipe samples, bulk residue samples on floors and roofs, sump and drain liquid and sediment samples, the sheet also determines for each structure the number of measurements on items of equipment, samples from that equipment, the number of boreholes drilled through the floor at regular intervals, the number of profile measurements in boreholes, and the number of profile samples collected from boreholes. Labor requirements are also determined, as shown in the example above, for the installation of floor, wall, and roof grids, to perform each type of measurement, to perform quality control (duplicate) measurements, to collect and analyze each of the several types of samples, to label, photograph, and perform measurements on equipment items, and to document all field data records.

Similar operations were carried out to prepare the total scope of work for Category N structures and for open land areas. Once the total labor (person-hours) is obtained for all tasks, the number of days required to complete the characterization may be calculated on the basis of available personnel and their probable efficiency. The actual number of hours worked per day is governed by the time required to check and standardize radiation measuring instruments, to put on and remove protective clothing, for mandatory rest and heat stress monitoring, for measuring radioactivity on personnel upon exit from controlled areas, and for other administrative functions. In most projects of this type, one can assume a productive work period of 6 hours per day.

The Symphony spreadsheet is well suited for project control applications. Once the total scope of work and estimated project costs have been determined, a project cost plan and work schedule can be prepared in the spreadsheet format. Through the entry of a few values following the weekly (or monthly) closing of cost collection, automatic features built into a spreadsheet may be used to monitor actual performance and to spot trends in the different project activity codes. This latter feature is useful in identifying potential problems before they occur thus permitting corrective measures to maintain project objectives.

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SAFEGUARDS AND HEALTH PHYSICS

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ABSTRACT

Safeguarding essentially deals with controlling of fissile material stored and manipulated in health physics controlled areas of nuclear establishments. In this contribution the particular health physics aspects are dealt with which concern a safeguard inspector when performing his safeguard control duties. It also describes the problems a Health Physics Responsible runs into when dealing with radiation protection for safeguard inspectors or his organisation. The paper is the result of a ten years experience in the field of radiation protection gained at the Safeguard Directorate of the Commission of the European Communities controlling about 380 nuclear establishments and facilities for non-proliferation of fissile material in the European Community

THE ASSESSMENT OF RADIOACTIVE DOSE OF URANIUM MINERS

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1. Abstract

The main factors of radioactive hazard in uranium mines are radioactive ore dust, radon gas and tis daughters. These hazards often result in contracted sillicosis and lung cancer among miners, Howere, they can be fully avoided if some active radiation protection are taken up. According to the statistics (incomplete) over the past 30 years, average annual sillicosis rate in uranium mines of China was only 0.083%, far less than that of coal mines and metal mines. On the other hand, the average lung cancer rate was 108.3/million.men.annum, which is also very low.

2. Introduction

The first uranium mine was discovered in 1934 in Guangxi Chuang Autonomous Region of China, but recounaissance and exploration began in 1955, the mine construction was in 1958. After 1963, some uranium mines put into production and now there are more than twenty active uranium mines in China.

In the process of uranium exploration and mining, from the 30 years practice and experience, it is demonstrated that the main hazard factors are radioactive ore dust, radon gas and its daughters for uranium miners because of sillicosis and lung cancer intruced. In the initial period of the exploration and construction, some miners were found to be sillicosis by reason that the protection measures for the radioactive ore dust, radon gas and its daughters were insufficient.

3. The basic measures for controlling the radioactive harm of uranium mine

3.1 Adoption of dust-proof and radon-reduced measures

During the past 30 years, there are three periods for dust prevention and radon reduction in uranium mines of China.

The first period(1955-1963) included uranium exploration and initial construction. In this period, no mechanical ventilation system was adopted and the natural ventilation was the main way, the local stope and protection conditions were very poor. Some drilling were dry-type operating. The maximum dust concentration was up to 150 mg/m^3, and the average was 10 mg/m^3. The maximum radon gas concentration was 125 Bq/L, and the average

was 11.5 Bq/L.

The second period(1963-1975) indicated that overall dust prevention and radon reduction were developed comprehensively. After 1963, the uranium mine, one after another, was put into production. For prevention against occupational disease, some comprehensive measures of dust prevention and radon reduction were taken including "water supply at first and then ventilation" etc, and mechanical ventilation system was established and the approaches to dust-measuring, radon-monitoring and radiation monitoring were also set up. According to the statistics of the representative ten mines, averag ore dust concentration was 2.5 mg/m^3, and qualified rate was 65%, average radon gas concentration was 4.81 Bq/L, and qualified rate was 66% from 1963 to 1975 in the stopes of underground uranium mines. the incidence of sillicosis in a uranium mine of Hunan was dropped from 1.28% in 1964 to 0.17% in 1974.

In the third period(1976-1986), scientific research and development were strengthened and the experiences were summarized and promoted and applied. Since 1978, the ventilation system was adjusted in accordance with the different kind of uranium mines, the research for dust prevention and radon reduction was developed widely, the standard of dust prevention by ventilation and the technique of radon reduction were raised. From statistics of 13 mines, it shows that the average dust concentration was dropped to 1.19 mg/m^3 and qualified rate was raised to 74.1% from 65%; average radon concentration was dropped to 4.85 Bq/L; and qualified rate was 73.8%; radon daughters concentration was 2.27 WL, and qualified rate was 76.5% from 1976 to 1986 in the underground working faces.

3.2 Adoption of comprehensive measures for dust prevention and radon reduction

At the present ventilation is the main way for dust prevention and radon reduction and its daughters concentration radioactive hazard in uranium mines of China. In order to eliminate radon and its daughters timely and rapidly, mechanical ventilation is adopted day and night in uranium mines. The amount of ventilation is calculated in accordance with the National Standard for eliminating radon gas and its daughters. It is 4-6 times greater than that of non-uranium mines under the condition that there are the maximum workers in the underground or the amount of dynamite is equal.

3.3 Ventilation

Over the 30 years, a few of practical experiences in uranium mine ventilation have been integrated including ventilation method, air amount of calculation, administration, etc, at the same time, some technicologies were adopted, such as the application of positive pressure ventilation for controlling the emanation rate and the usage of air-guided plate, etc. Blowing ventilation is an effective method for the control of radon emanation and the decreasement of radon concentration. Forced and exhaust overlap auxiliary ventilation was adopted in the

local area and descentional forced ventilation was adopted in shrinkage and filling stopes, both of them were also effective for decreasing stope radon gas and its daughters concentration.

Sectional ventilation system and exhaust ventilation method are adopted widely in uranium mines and surfacial coating and isolation techniques are applied to reduce radon emanation and migration. The practice shows that coating on the surface of ore bodies with organic coating material may reduce 75% of radon emanation.

3.4 Ore dust prevention and radon reduction

The main measures in uranium mining are to restrain dust source and to prevent ore dust as well as to collect ore dust, and the wet operating, sprinkling water and waterseal explosion will be all effect to restrain dust source and quicken dust condensation.

The method mechanical ventilation is used directly for high dust concentration and air return ways radon pollutant, and dry-type dust catcher with superfine fabric cloth and the method of dust removol by super high-pressure electrostatics are adopted in some uranium mines with the result of depressing 99% Of dust and 89% radon daughters from air.

3.5 Personal health and protection

In order to safeguard the health of uranium miners, they must be inspected before their employment, and after that, the inspection must be carried out at regular intervals, on the other hand, a more subsidies for health, a better health foods and personal-protective equipments are free supplied. For any miner, personal dose monitoring should be carried out. And now protable and fast-speed air-ball radon detectors are used widely in uranium mines.

4. The evaluation of radioactive dose in uranium mines

China has a history of more than 30 years in uranium exploration mining. Over 30 years, in order to understand and control the hazard of radioactive ore dust, radon gas and its daughters in uranium mines, a great deal of inverstigation has been carried out so as to study the characteristics of sillicosis and lung cancer of uranium miners. It was reported that, from 1962 to 1985, the average annual sillicosis rate is only 0.083% in uranium miners of China, which was far less than that of miners in coal mines and metal mines.

According to census from 1955 to 1985 in uranium miners, average death rate of lung cancer was 19.5/million·men·year. A uranium mine of Hunan was the highest in uranium mines of China in which twelve patients of lung cancer was found from 1976 to 1983, among them eleven was dead. Average death rate was 357/million·men·year. The accumulated exposure of workers who worked in the underground for 2-19 years was equal to 130-450 WLM. The

age of miners who had lung cancer was from 40 to 62 years, average age of death was 46 years, which was 11 years lower than that of the civilian, death rate was 30 times higher than that of the vesident.

Meanwhile, 402718 employees in five uranium mines were investigated in epidemiology from 1958 to 1982, amoung which the death rate of lung cancer was 108.3/million·men·annum. in checking of the dead bodies, it was found that the radioactive composition in the lung organization indicated uranium concent was less than that of the ordinary resident and that the sillicosis for uranium miners would be caused mainly by the pathological changes of lung organ leading to the obstructive of lung function because radioactive ore dust could make the sillicosis originate and develop. Lung cancer from that uranium miner suffered was caused mainly by potential of radon daughters as a result of its internal exposure in lung organ. The result of reseach shows that the incidence of uranium miners is linear effect of uranium ore dust and radon daughters may promote the obstructive of lung function.

5. Conclusion

5.1 In uranium exploration and mining, the main radioactive harmful factors are uranium ore dust, radon gas and its daughters, resulting in that uranium miners will suffer from sillicosis and lung cancer.

5.2 Incidence of uranium miner is linear with the radon gas and its daughters concentration. The complex effect of radioactive ore dust and radon daughters will promote the obstructive of lung function.

5.3 The basic measures for protecting miners from radioactive hazard are to adopt mechanical ventilation, wet-type operating, sprinkling water so as to decrease the exposure surface of bodies and ore amount stored in underground and to apply surfacial coating and isolation of bodies, and to seal the mined-out area and abandonment tunnels.

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RADIOLOGICAL PROTECTION PROCEDURES RELATED TO THE HANDLING OF DIFFERENT URANIUM COMPOUNDS ENRICHED 20% IN 235U

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The obtainment of U3O8 starting from UF6 comprises a wet stage consisting in the extraction of gaseous UF6 by heating at 60-80°C, its hydrolysis to uranyl fluoride and precipitation to ammonium diuranate, and a dry stage where conversion to U3O8 powder takes place through calcination at 800°C and further treatment until the attainment of the raw material for fabrication of fuel elements.

The treatment of these compounds upon the stated process involves a prevailing toxicological risk due to exposition to UF6, FH and UO2F2, and a radiological risk of fundamentally non-stochastic nature for the critical accident and stochastic nature for the dispersion of materials.

Exposition to UF6 constitutes an emergency because in a few minutes UF6, FH and F2UO2 incorporations can reach toxicological limits. On the other hand, these incorporations represent only 10^{-2} times of ALI.

Uranium (20% 235U) air concentration

-Toxicological risk (UF6, UO2F2)	
continuous exposure	200 μg m^{-3}
TLV-EE 10 min	3 mg m^{-3}
-Radiological risk (UF6, UO2F2)	
continuous exposure	53 μg m^{-3}
ALI	134 mg m^{-3}

For the other compounds, ALI can be reached in a few minutes in case of inhalation of dry material dispersion.

	ALI
ADU	3.10^4 Bq
U3O8	103 Bq

For continuous exposure derived uranium (20% 235U) air concentration is determined by the radiological risk.

Uranium (20% 235U) air concentration

Toxicological risk	200 μg m^{-3}
Radiological risk	
ADU	53 μg m^{-3}
U3O8	1 μg m^{-3}

Adequate respiratory protection systems for the above risks are:

Fume hood ($1m^3\ s^{-1}$ air velocity)	Hazards involving UF6 escape have to be avoided, personal protection is necessary for emergencies.
Glove box with air extraction system	UF6 explosion has to be avoided for glove box integrity.

The protection system adopted for the Plant Production at the CNEA Constituyentes Atomic Center is glove box with 10-15 air renovations h^{-1}, $1m\ s^{-1}$ velocity for any opening. In view of the differences in the process, kind of materials and type of risk involved in case of internal contamination the wet stage and dry operations are performed in two different enclosures.

The main risks in handling these compounds are the probability of internal contamination with stochastic radiological effects and the probability of criticality with mainly non-stochastic radiological effects. The hazards involved in these risks are UF6 cylinder explosion, UF6 escape, dispersion of materials and critical excursion. The Plant was designed under the principle of totally avoiding the first and second hazards, and reducing to minimum the consequences of the last two.

Several events may lead to internal contamination if combined with gloves or bags rupture and glove box ventilation failure:

- the presence of HF or impurities in the UF6 cylinder may result in explosion while heating. Prevention consists in UF6 vapor pressure determination at room temperature before heating.
- failure in impurities detection would lead to an abrupt rise of pressure during heating for which the countermeasure consists in the automatic relief to a gas expansion system.
- heating at temperatures higher than 121°C would lead to hydraulic rupture of a full cylinder; redundant, independent controls of temperature linked to automatic stopping of heating for T above 110°C are needed.
- heating of an overfilled cylinder can lead to hydraulic rupture of the tube. Tube weight must be determined prior to heating.
- contact of UF6 with hydrocarbons generates explosive mixture. The use of hydrocarbons is forbidden in the plant and fluorinated oils must be used in pumps.
- cloggins in piping or valves and water vapor presence are events conducing to pressure rise in the gas transfer system with the probability of loss of primary confinement. Check of piping leakage and use of purge gas are necessary prior to gas transfer. In case of pressure rise, connection to the already mentioned gas expansion system favours the state of confinement of UF6 gas.

The prevention of criticality is given operatively by the mass control at the wet stage, and mass and moderator control at the dry stage. Always units of less than 2,4 kg of uranium are handled.

The most relevant hazard for criticality lies in an uncontrolled transfer of UF6 to the hydrolysis vessel. For its prevention hydrolysis takes place in a vessel with annular geometry embodying a cadmium sheet in its construction. No matter the amount of UF6 transferred, critical excursion cannot take place. Physical restraints avoid the inadvertent transference to unsafe geometry.

In case of UF6 dispersion, liquids or material spreading, inherent safety is achieved by safe design of the air extraction system, safe geometry of the glove box floor and of collector bowls, and safe distribution of subcritical units.

As shown in Fig.1 interactions between subcritical units are only relevant after a chain of events of low probability has occurred. (B) identifies an interaction between dispersed solid UF6 and UO2F2 and spread water. (A) and (B) identifies the same situation as (B) but the interaction is with a spread UO2F2 saturated solution in safe geometry.

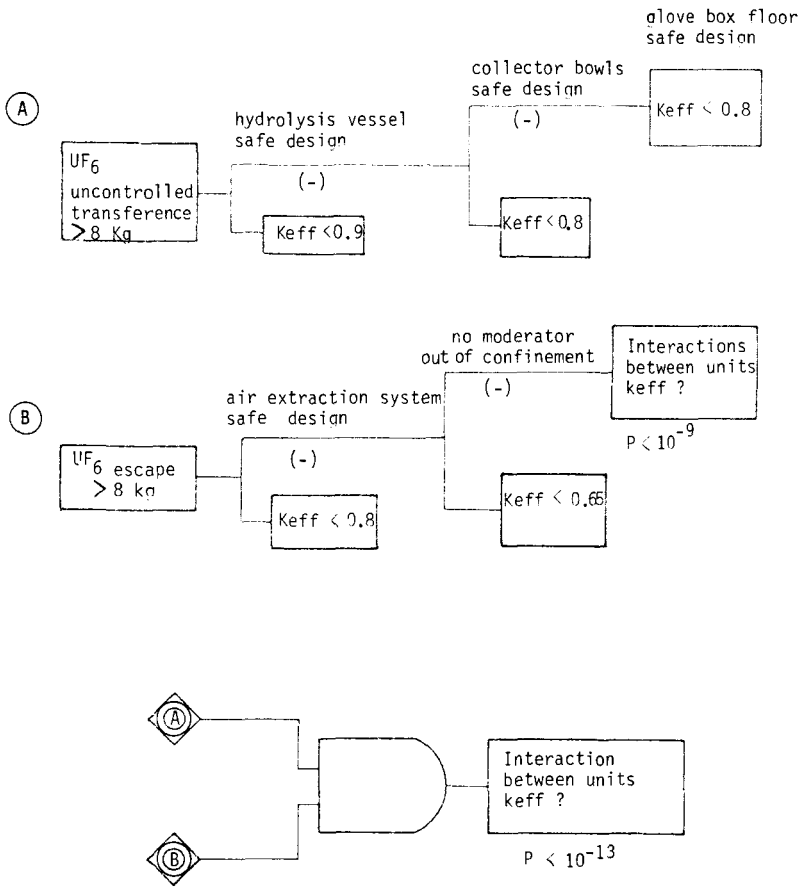

Figure 1

A simple administrative control of mass and moderation limits with wide margins of safety along the dry powder line was established. Ten kilograms of uranium are allowed, administrative mass control is aided by limiting the number and capacity of containers, so that operative control is reduced to respect the prohibition of moderator in excess of the minimum amounts allowed.

The operative experience showed that the principal hazards had been sufficiently taken into account. Incidents of UF6 escape to the glove box occurres coming from UF6 deposits in the connection pigtail to the transfer system and UF6 deposits in the nozzle through which the gas enters to the hydrolysis vessel and the hydrolysis vessel cover. As expected, monitoring showed absence of contamination out of the glove box. After these incidents an outlet for the glove box air extraction was placed close to the cylinder valve. Admissible levels of contamination were detected in empty cylinders coming out of the glove box. Liquid and solid spreadings occurred as expected and without radiological consequences. There were no incidents involving internal contamination.

Intensive monitoring during the fabrication campaigns demonstrated that the risks had been adequately evaluated and safely handled.

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Experience of personnel monitoring at the Plutonium Fuel Fabrication Facilities

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1.INTRODUCTION

Increasing use of plutonium in nuclear fuel cycle,it is important to measure precisely and to reduce properly occupational dose at plutonium fuel fabrication facilities.

At Tokai Works,Power Reactor and Nuclear Fuel Development Corporation (PNC),the plutonium fuel fabrication facilities have been operating since 1965. Plutonium and uranium mixed-oxide fuel for the Experimental Fast Reactor (JOYO) and the Heavy Water Reactor (FUGEN) have been manufactured continuously. About 79 tons of mixed-oxide fuel have been fabricated and shipped to reactors,and the fuel included 295 assemblies to JOYO and 332 assemblies to FUGEN until 1986.

Gamma ray and neutron have to be considered for external exposure to workers in the facilities. We have been making intense efforts to reduce occupational exposure of workers,such as shielding of radioactive nuclides and mechanization of fuel fabrication process. We had developed the self-designed personnel monitoring system using TLD badge and finger ring dosimeter, and occupational doses of workers have been measured routinely. Also the dose limitation system shown in Table.1 have been adopted for exposure control of workers based on ALARA concept. Consequently the doses of workers have been kept enoughly below the dose limit recommended by ICRP.

This paper describes the outline of the personnel monitoring system and the overall view of the occupational doses at the Plutonium Fuel Fabrication Facilities based on our experience for more than 20 years.

Table.1. Dose limitation system at PNC Tokai Works

Category	Investigation level	Action level	Dose limit
A	3 mSv/3 months	13 mSv/3 months	30 mSv/3 months
B	1 mSv/3 months	4 mSv/3 months	15 mSv/year

A: Workers who recieve occupational dose continualy
B: Workers who enter the controlled area occationaly

2.PERSONNEL MONITORING

1).Principle

At the facilities,a large quantity of PuO_2 and UO_2 are handled in the process of mixed-oxide fuel fabrication. Several kg order of PuO_2 and UO_2 are handled in a glove box through almost all process of the facilities. The dominant nuclides which cause external exposure are ^{239}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am. These isotopes emit gamma ray,low-energy X-ray,spontaneous fission neutron and (α,n) reaction neutron. Glove box is made of acry-

lite panels of 10 mm thickness and lead-contained gloves. Gamma and neutron dose rates locally become to a few hundreds of μSv/h order at the surface of glove boxes. To reduce the dose rates, lead-contained acrylites of 35 mm thickness are added at preferable places.

The dominant energy of gamma ray in working areas is 59.5 keV from ^{241}Am. Neutron energy spectrum measured by a spherical multi-moderated ^{3}He detector is shown in Fig.1. More than 90 % of dose equivalent is originated from the fast neutron above 15 keV.

Considering these radiation features in the facilities, personnel monitoring program have been designed and monitoring methods have been developed. And occupational doses of workers have been measured for whole body and extrimity(finger).

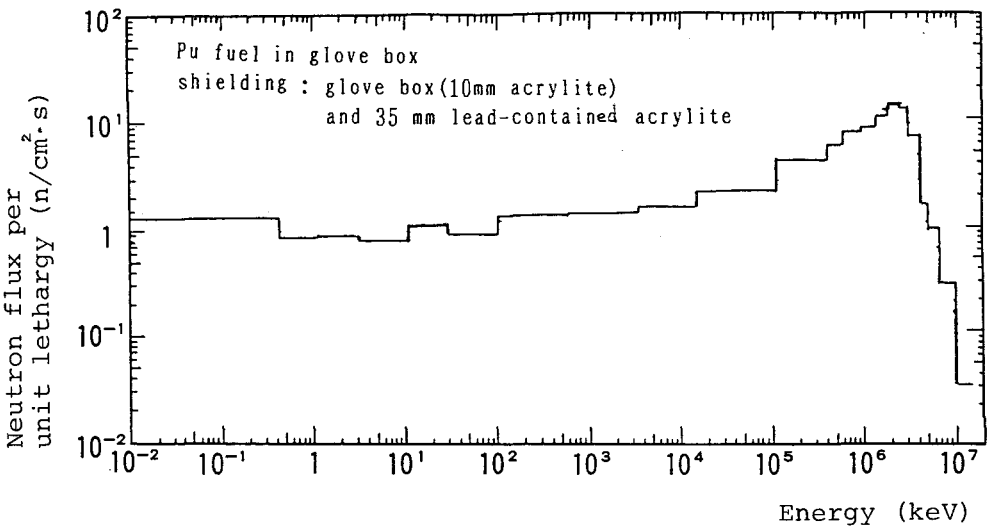

Fig.1. Neutron energy spectrum at the plutonium fuel fabrication facilities

2).Monitoring methods

The doses to radiation workers have been evaluating quarterly or monthly using TLD badge and the finger ring dosimeter.

For the measurment of whole body dose,TLD badge capable of evaluating gamma,beta and neutron doses had been developed and applied to the routine personnel monitoring. Gamma dose is evaluated as 1cm-deep dose equivalent by a $^{7}Li_2{}^{11}B_4O_7$(Cu) element behind a filter of 1000 mg/cm² thickness. Three $^{6}Li_2{}^{10}B_4O_7$(Cu) elements and one $^{7}Li_2{}^{11}B_4O_7$(Cu) element are used for evaluating thermal,epi-thermal and fast neutron doses separately by means of Albedo method. The recording level for TLD badge are 0.1 mSv for gamma dose and 0.2 mSv for neutron dose respectively.

Especially for workers who handle PuO_2 and UO_2 in glove boxes, the finger ring dosimeter of TLD type have been used. Finger ring dosimeter consists of $CaSO_4$(Tm) enclosed in rod-type glass ampoule and the element is surrounded by a energy compensation filter. The recording level is 0.2 mSv for gamma dose.

3.OCCUPATIONAL DOSE STATISTICS

The number of workers and annual collective dose for the period of 1972-1986 are shown in Fig.2. The total collective dose in these 15 years was 5.6 man·Sv. At the early stage in the fuel fabrication,the collective doses were relatively high. By the introduction of radiation shielding and mechanization of the fuel fabrication process,the collective doses had decreased in the period of 1974-1978. But the collective doses had incresed from 1978,because plutonium recoveried from high burn-up spent fuel had become treated.

The precise dose data of 1982-1986 are shown in Table.2. About 250 workers had doses above the recording level(0.1 mSv) every years. The ratio of gamma to neutron in annual collective doses were 0.95-1.19. Annual average doses were about 1.8-2.7 mSv for workers of measurable dose.

Fig.3 shows the probability vs dose of the individual annual doses for fiscal years of 1984-1986 in log-normal distribution. About 45 % of workers over the recording level were below 1 mSv and 80 % of them were below 5 mSv. As individual doses were controlled to be kept below the investigation level (3 mSv/3 months) defined in our dose limitation system,the individual highest annual doses have been get down to in a dose range between 8 mSv and 12 mSv. Therefore the dose distribution did not fit to log-normal distribution in higher dose range over 5 mSv.

Plutonium and uranium mixed-oxide fuel have been also used at the Heavy Water Reactor (FUGEN) of 165 MWe. In the period of 1978-1985,the cumulative electric output of the reactor was 0.495 GWe·y. The collective dose of fuel fabrication workers corresponding to this reactor operation was 0.416 man·Sv. The collective dose of plutonium and uranium mixed-oxide fuel fabrication per unit generated output of FUGEN was o.84 man·Sv/GWe·y.

The highest extrimity annual doses at finger were 17.7-61.0 mSv in 1978-1986.

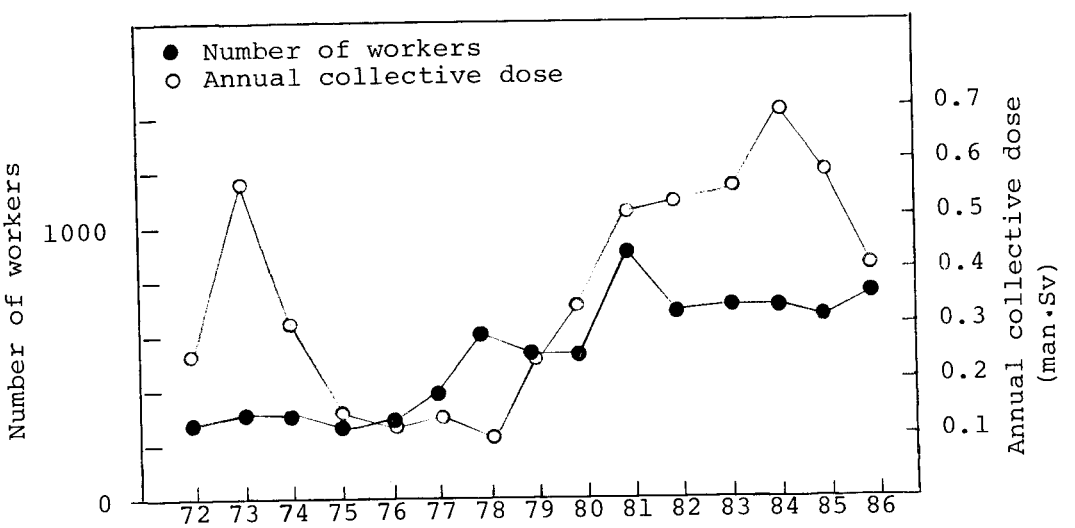

Fig.2. Number of workers and annual collective dose at the plutonium fuel fabrication facilities

Table.2. Doses at the plutonium fuel fabrication facilities

fiscal year	number of workers	number of workers with measurable dose	collective dose (man·Sv)			average dose (mSv)
			gamma	neutron	total	
1982	584	266	0.26	0.27	0.53	2.0
1983	721	254	0.28	0.30	0.58	2.3
1984	692	279	0.32	0.38	0.70	2.5
1985	660	220	0.28	0.31	0.59	2.7
1986	762	231	0.21	0.20	0.41	1.8

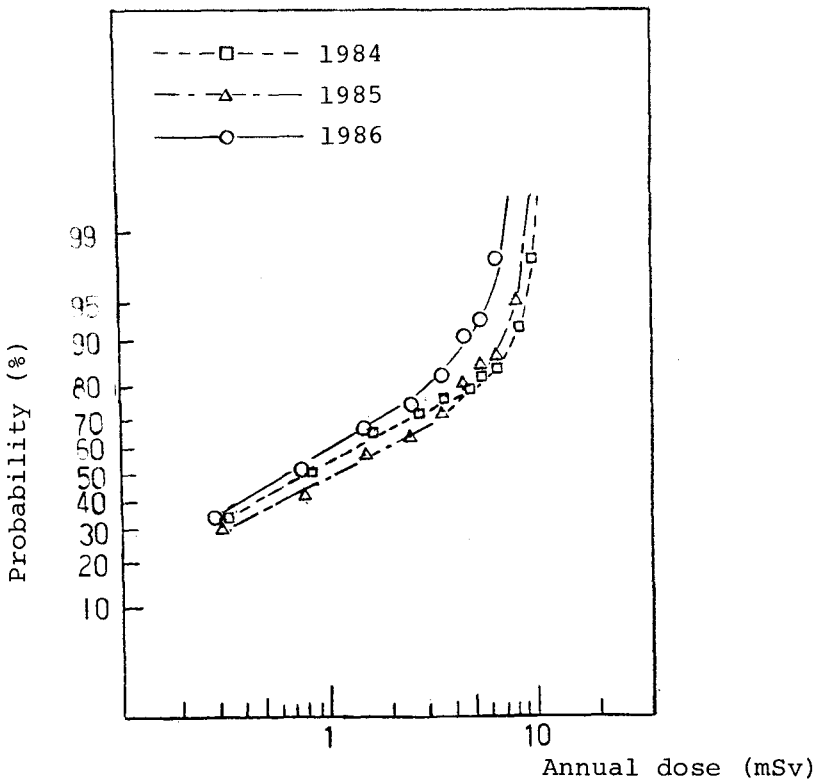

Fig.3. The log-normal distribution of annual individual doses

CONSTRUCTIONAL DESIGN PRINCIPLES ON RADIATION PROTECTION AND METHODS OF RADIATION CONTROL AT THE LARGE SCALE PLUTONIUM FUEL FABRICATION FACILITY OF FULLY REMOTE OPERATION

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1. INTRODUCTION

The large scale Plutonium Fuel Fabrication Facility (Pu-3rd) of PNC is designed to fabricate the Pu and U mixed-oxide (MOX) fuel for FBR "MONJU" and "JOYO". The fuel production capacity of Pu-3rd is enlarged to 5 tons MOX per year.

The fuel fabrication process is operated by fully remote operation.
The process consists of:

(1) Pellet Fabrication Process
(2) Pin Loading and Assembling Process
(3) Analytical Chemistry Process
(4) Storage for plutonium powder, pellets, pins and assemblies

The external exposure to the workers at a plutonium fuel fabrication facility is potentially larger than that of uranium fuel fabrication facility for LWR because of the gamma ray emitted from Am-241, the daughter nuclide of Pu-241, and of the neutron by the reaction $O(\alpha,n)Ne$ and spontaneous fission.

At the existing FBR Fuel Fabrication Facility of PNC (Pu-2nd) where the operation is done manually, there are three operating modes which bring external exposure to the workers. The operating modes are:

(1) Operation with gloves.
(2) Observation in front of glove boxes.
(3) Operation at instrument panels.

We have experienced that the collective dose per unit plutonium fuel fabrication at Pu-2nd had been increasing due to the use of recovered plutonium from high burn-up spent fuel. Therefore it is the essential issue at the design stage of Pu-3rd to reduce the external exposure from the viewpoint of ALARA.

Pu-3rd is constructed based on the experiences of Pu-2nd reflecting the changes for better and rationalization.

This paper shows the constructional experiences of Pu-3rd focusing on the design principles on external exposure reduction and methods of radiation control.

2. DESIGN PRINCIPLES

2.1 General design principles

At the design stage of Pu-3rd, the requirements given in the guide of the Nuclear Safety Commission of Japan [1] have been taken fully into consideration in order to ensure the safety features. This guide lays down requirements for radiation control and other safety measures.

The design criteria for radiation protection are as follows:

(1) Shielding appropriate to an individual process should be considered in order that the external exposure of the personnel does not exceed the level given in radiations.
(2) For the purpose of decreasing the internal exposure of personnel, a system of multiple confinement barriers should be applied
(3) The radiation control system should be designed to maintain a good operating environment.

2.2 Design principles for external exposure reduction

The principal causes of external exposure gotten from the experiences at Pu-2nd are as follows:

(1) Direct access to the glove box and mannual handling of the radioactive materials (exposure sources).
(2) Storage of the radioactive materials in the working area such as inside of glove box or storage box in the process room during the routine inspection and maintenance work.
(3) Surface contamination on the inside wall of glove box caused by leakage of materials from the process machine.
(4) Maintenance work at the process room staying in the high level radiation field.

To prevent the external exposure from the causes above mentioned, the following design principles are incorporated:

(a) Fully remote operation of the fabrication process from the process control room.
(b) Prepare the intermediate storages for feed powder and pellet. When the operators need to enter the process room, the radioactive materials are transfered to the storage before entrance.
(c) The powder process machines are improved with high containment ability and powder feeding is done by double containment system.
(d) Glove boxes are designed to be separable to each other. When it needs to maintain the machines, the glove box and the machine are separated in the lump and transfered to the maintenance room where the radiation level is kept to be at the background level.

3. RADIATION CONTROL DESIGN FEATURES

3.1 External radiation monitoring

Features of external radiation monitoring are as follows:

(1) A fixed area monitoring system is applied to the facility. Signals and alarms from the detectors can be observed on the centre panel in the radiation control room.

(2) Each monitor consists of couple of gamma ray monitor and neutron monitor. The G-M detector for gamma monitor is improved to have good respose to the dominant gamma ray energy of 59.5 keV from Am-241.
(3) Each monitor is installed at the entrance of the process room to be able to judge whether it is permissble to enter or not.
(4) Entrance to the process room is limited under the normal operation mode that the radioactive materials exist in the room. The entrance limitation indicators are installed at the entrance of the rooms.

3.2 Air contamination monitoring

Features of air contamination monitoring are as follows:

(1) A continuous airborne radioactivity monitoring and sampling system is applied to the facility. Signals and alarms from the detectors can be observed on the centre panel in the radiation control room.
(2) Alpha ray energy to be detected is limited to the range of 3.5 MeV to 5.5 MeV focusing on the alpha ray from the plutonium nuclides.
(3) The monitoring system has alpha ray energy spectrum analyzors to be able to distinguish between natural alpha emitters such as Rn-Tn daughters and plutonium nuclides as early as possible just after alarm.
(4) The airborne trap for air sampling of the process room is installed outside of the room to prevent external exposure of workers who change the airbone sampling filters.

4. BENEFITICAL ESTIMATION OF EXPOSURE DECREASE RESULTED FROM FULLY REMOTE OPERATION

The cumulative collective dose for FBR Mox fuel fabrication at Pu-2nd is about 2 man Sv during the period from 1981 to 1986. Total amount of fuel production in this period is about 2.5 Ton・MOX. Therefore the average collective dose per unit fuel production is estimated to be about 0.8 man Sv/Ton・MOX.

If the operation at Pu-3rd is done by manual handling, the annual collective dose is estimated about 4 man Sv/y based on the exposure at Pu-2nd above mentioned. This value means that the individual annual collective dose amounts to 40 mSv/y for each of one hundred workers at Pu-3rd.

It can be done to calculate benefitical estimation of exposure decrease resulted from fully remote operation of Pu-3rd.

Benefitical estimation is measured by quantity of
a) collective dose per unit fuel production ; man Sv/Ton・Mox
and
b) collective dose per unit energy production ; man Sv/GW(e)y
comparing between the actual result of Pu-2nd and design objective of Pu-3rd

The results of estimation are shown in table 1.

Table 1 Benefitical estimation of exposure decrease

	result of Pu-2nd	objective of Pu-3rd	Pu-3/Pu-2
Amounts of fuel production	2.5 Ton · MOX (1981~1986)	5.0 Ton · MOX/y	
Collective dose	2.0 man Sv/2.5 Ton · MOX	2.3×10^{-2} man Sv/5 Ton · MOX	
Collective dose per unit fuel production	0.8 man Sv/ Ton · MOX	4.6×10^{-3} man Sv/ Ton · MOX	$\frac{1}{180}$
Design burn-up of fuel	50,000 MWD/MTM* (JOYO MK-II)	80,000 MWD/MTM* (MONJU)	
Collective dose per unit evergy production	18 man Sv/GW(e)· y **	6.3×10^{-2} man Sv/GW(e)· y **	$\frac{1}{280}$

* MTM(Metric Ton Metal) = 0.9 Ton ·MOX

** GW(e) ·y = MWD ÷365(day/y) ×0.3 electric efficiency $\times 10^{-3}$(Giga/Mega)

5. CONCLUSIONS

The large scale plutonium Fuel Fabrication Facility (Pu-3rd) has started its uranium test operation on November 1987. The effect of constructional design principle on radiation protection will be demonstrated through the following fullscale operation.

It will be the subject to the health physicist to develop and establish the administrative standards and procedures for radiation protection and control at a fully remote operating facility based on the experiences of Pu-3.

Reference

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ANALYSIS OF MEASUREMENTS IN EUROPE FOLLOWING THE ACCIDENT OF CHERNOBYL NUCLEAR PLANT.

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INTRODUCTION

A long-range pollutant transport and deposition model, developped by Electricité de France (Aquatic and Atmospheric Evironment Department) and Meteorologie Nationale (Franch Weather Service), is used to analyse the Chernobyl radioactive plume dispersion over the European continent. Model predictions are compared to field measurements of ^{137}Cs activities in the air from April 26th to May 5th, 1986.

THE DATA BASE

During and following the accident on the Chernobyl Nuclear Power Plant RBMK 1000, European countries have done many measurements, either on a permanent basis used for environmental survey, or exceptionally taking into account the importance of the Chernobyl disaster.

Measurements presented in this paper have been done from April 28th to May 12th 1986. The results are concentrations of Caesium 137 in air, in term of radioactivity expressed, in Becquerel per cubic meter. Atmospheric concentration has been measured using an air pump connected to a filter retaining particles of aerosols. The period of measurement is 24 hours, generally from morning to next morning.

As regards the results of measurements done, they should be modified by the uncertainties due to : the yield of the filter, the calibration of the γ counter, the sampling, the measurement itself.

In French laboratories, the yield of the paper filter is near 65 % if the size of particles is represented by a log-normal distribution with a mean and a standard deviation respectively of 0,3 μ m and 2 μ m. Granulometric measurements have shown that the size of measured particles is near 1 μ m, and consequently the yield of the filter can be estimated to 80 %.

The uncertainty due to γ counter calibration is about 10 % of the value of the measurement. The uncertainty due to the measurement itself is included in 25 % to 2 % in function of the value of the radioactivity of Iodine 131. This same uncertainty concerning Caesium 137 is between 40 % and 2 %.

II AIR MASS TRAJECTORIES

II-a) Air Mass Trajectories

The computation of three-dimensional trajectories was accomplished using the synoptic wind field and vertical velocities obtained from the analyses of the E.C.M.W.F.

The use of vertical velocity eliminates constraints associated with the level of the trajectory (isobaric trajectories, for instance). Only the exact location and altitude of the trajectory at the starting point are required. General studies have been made to evaluate this trajectories computation (Martin et al. (1987)). It is shown that the kinematic and geographical location of air masses were improved with this method.

The computed trajectories for the Chernobyl accident provide interesting qualitative information (Strauss B. and Gros J.M. (1987)), but are not sufficient to explain all the measured data. In consideration of mean plume rise, three sets of trajectories beginning respectively at 925, 850 and 700 mb have been computed. These levels characterize the diurnal evolution of the mixing height over the emission area.

To simulate the meausrement taken ten days after the Chernobyl accident (April 26 th, 0 GMT), eighty trajectories (i.e. a new trajectory every three hours) for the three plume rise levels have been determined.

II-b) Dispersion and Deposition

Assuming the plume centerline follows a trajectory defined by straight line segments, a Gaussian concentration field is generated, at a given time, around each segment. A plume segment is characterized by its initial polluant content and its total travel distance. The effects of material removal and plume processes are taken into account to respect the pollutant balance at each segment.

II-c) ^{137}Cs Source Term

The total source term was evaluated by the USSR authorities and published at the Vienna Conference in August 1986. For the ^{137}Cs, the value is 3.7 10^{16} Bq.

The daily emission percentage were provided by the French atomic Energy Agency : 24 % on April 26th, 8 % on April 27th, 6,8 % on April 28th, 5,2 % on April 29th, 4 % on April 30th, 4 % on May 1st, 8 % on May 2nd, 14 % on May 3rd, 14 % on May 4th, 1986.

A relation between pollutant emissions at different initial heights (925, 850 and 700 mb) was established in regard of mixing layer evolution. This evolution, determined by the radiosoundings taken, over Kiev at midday and midnight, allows to built the chronology of the initial emission on each level (925 mb, 850 mb, 700 mb).

III - COMPARISON WITH MEASUREMENTS

As not enough dry wet deposition measurements were available, only the actual daily air activity averages are compared.

To quantify the correlation between measured data and computation results, the order of magnitude (i.e $\log_{10}$ of the ^{137}Cs activity) is considered.

In figure 1, measured points are plotted against computed values. Three linear regressions are drawn. The correlation coefficient is 0.57. The total number of plotted values equals 402.

The number times the model results agree, over or underestimate measurements are also counted and presented in contigency tables. The first line and the first column are reserved for values lower than 10^{-3} $Bq.m^{-3}$.

Table 1 shows that :

- The model underestimates the measured data in 11 % of all cases.
- The model overstimates the measured data 35 % of cases.
- The model predicts no pollution where some ^{137}Cs was measured in 10 % of cases. The model predicts pollution where no activity was measured 2 % of cases.

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La Météorologie N° 15 pp 14-15.

TABLE I : EUROPE (10°W, 40°E, 35°N, 65°N)

	Am <-3	-3≤Am<-2	-2≤Am≤-1	-1≤Am<0	0≤Am<1	1<=Am<2	2<=Am<3	
Ac<-3	17	3	18	33	22	0	1	94
-3≤Ac<-2	2	0	0	1	3	0	0	6
-2≤Ac<-1	4	0	6	9	14	0	0	33
-1≤Ac<0	1	0	6	5	35	1	0	48
0≤Ac<1	1	1	2	25	188	4	0	221
1≤Ac<2	0	0	0	0	0	0	0	0
2≤Ac<3	0	0	0	0	0	0	0	0
	25	4	32	73	262	5	1	402

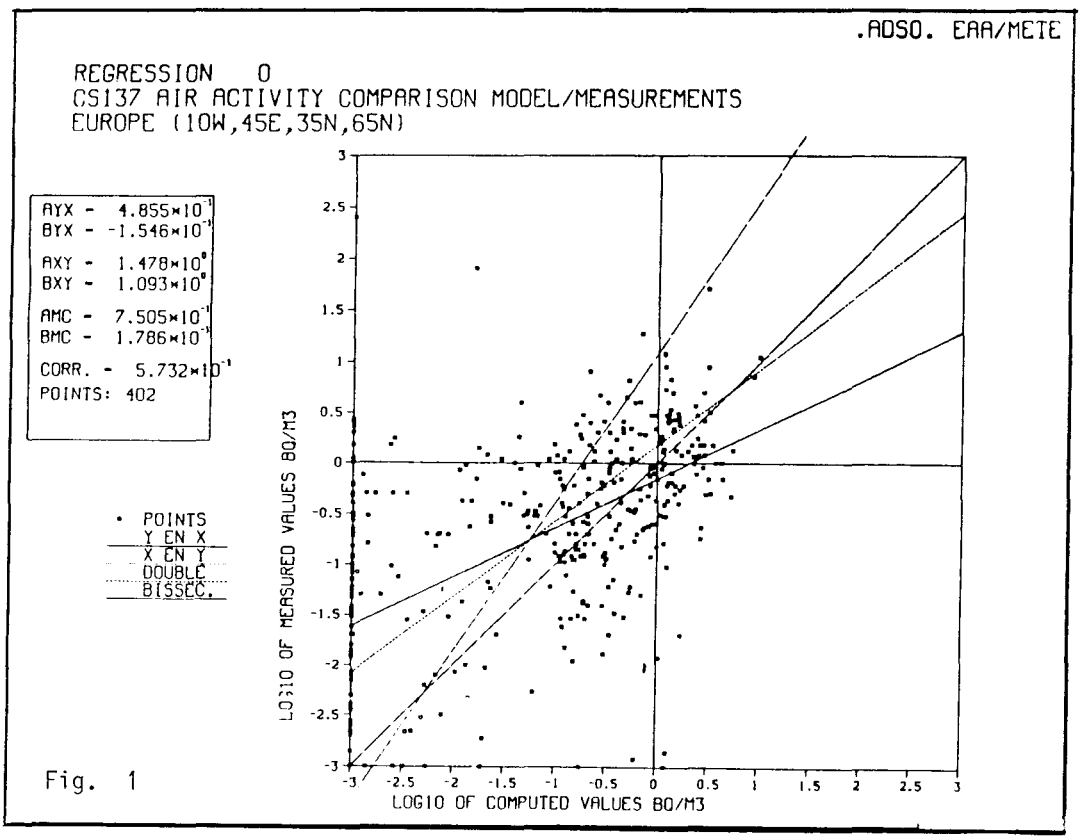

Fig. 1

THE SWISS EMERGENCY REFERENCE LEVELS AND THEIR APPLICATION IN THE CHERNOBYL-CASE

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ABSTRACT

The Swiss radiological emergency response planning system has been adopted by the Federal Commission of Atomic and Chemical Protection (KAC) in the year 1982. The system recognizes four phases of response; the phase of the accident, the preventive phase (predetermined protective actions), the early and the long-term phase of the contamination. The principal exposure pathway for the preventive phase is the whole-body external exposure from the plume (NPP-accidents) and from deposited material (nuclear weapons fallout and NPP-accidents); for the early phase (about the first two days) mainly the external pathway but also the ingestion of contaminated food and water; for the long-term phase the ingestion and with decreasing significance whole-body external.

The system of the external emergency reference levels specifies the action-levels for the predicted external dose (dose equivalent) to initiate immediately predetermined protective actions in predetermined areas. The predicted dose is the dose that would be received by individuals in the population if no protective action were taken. The primary protective actions may be staying in the house or sheltering. They are preventive actions and should be initiated in advance of any irradiation hazard. Evacuation will be carried out after a deposition of radioactive material has occured and will be based on measuring results.

The reference level for the ingestion defines the projected first-year dose (dose commitment) resulting from the consumption of contaminated food and water below which countermeasures are unlikely to be justified. In the swiss concept this reference level is the basic standard. Derived levels will be established only in the real case of an accident, considering the special situation of it like amount and concentration of critical nuclides in the different foodstuffs, nourishing behavior of critical population-groupes, vegetation-period, possibilities to replace the contaminated food by other one's or to store it, until the acitivty has decayed, and so on. When it seems likely that a radiation dose will exeed the reference level, countermeasures should be undertaken provided that a substantial reduction of dose is to be achieved. If the projected doses will only be moderately in excess of the reference level, the countermeasures should be such that they do not involve unacceptable risk or loss of income to the community.

During the Chernobyl-contamination parts of this emergency response system had to stand the test, specially the concept for the ingestion pathway. It showed clearly its advantages, e.g. flexibility to react specifically to the given situation, but also the limits. It was mearly impossible to explain to the public, why we didn't have well defined and published activity levels for the food like our neighbouring states.

MEASURED TRANSFER FACTORS IN MILK AND MEAT AFTER THE CHERNOBYL REACTOR ACCIDENT

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INTRODUCTION

After the nuclear reactor accident at Chernobyl the radioactivity in the environment in Aachen was measured in detail at the Lehrgebiet Strahlenschutz in der Kerntechnik /1/, /2/, /3/. The change of the different radionuclides in the eco-system made it possible to obtain radioecological parameters especially for iodine and caesium. The knowledge about the transport of iodine into cow's milk could be very much improved.

TRANSFER FACTOR OF IODINE INTO MILK AND MEAT

After the enhancement of the specific activity of I 131 in grass (starting May 1, 1986 /3/), the cow's milk of 4 larger farms and 1 small farm has been analyzed continuously using γ-spectrometry. Fig. 1 shows the change of the activity concentration of I 131 in the milk of farm B (36 cows) in the south of Aachen, starting with May 4. Because of the bad weather, the cows had only been on the meadow since a few days. Until May 9, they always grazed on the same part of meadow. Since may 10, the animals were lead to a new separated part of meadow every day. Since approx. May 23, the cows fed only grass, approx. 65 kg/d, except for approx. 3 kg/d of concentrated feed, , see Fig. 4. The right part of Fig. 1 shows the change of the specific activity in grass. Detailed information of the first 10 days can be found in /3/.

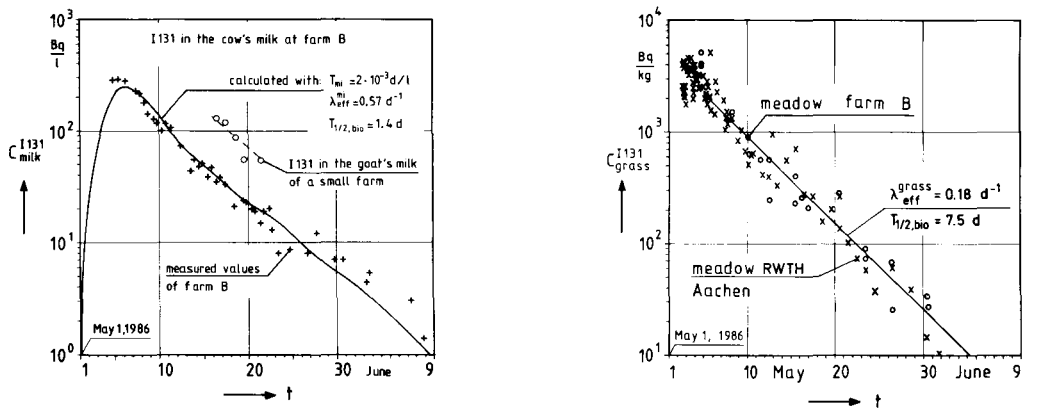

Fig. 1: Activity concentration of I 131 in cow's milk and specific activity of I 131 in the grass of farm B in Aachen

The transfer factor T_{mi} and the effective removal rate constant λ^{mi}_{eff} have been calculated using the least squares method. Surprisingly, the transfer factor of 0.002 d/l is about a factor of 5 smaller than it has to be used in the Federal Repub-

lic of Germany in licensing procedures. Using the results in /3/ it could be clarified why the transfer of I 131 from the air into the milk is overestimated by a factor of approx. 5 /4/ when using nowadays parameters /4/. λ_{eff}^{mi} = 0.57 1/d for I 131 corresponds approximately with the values given in various publications.

Measurements of goat's milk of a small farm give the activity concentration in May 1986 shown in Fig. 1. Compared with the milk of a cow grazing on the same meadow an I 131 activity concentration in goat's milk results which is approx. 3.4 times higher. Assuming a feed rate of 6 kg/d, a transfer factor for goat's milk of approx. 0.07 d/l results. Ewe's milk from the area of Aachen was not available for radioactivity measurements. From some iodine measurements in beef it can be derived that the transfer factor into meat is approx. equal to the transfer factor of iodine into milk being 0.002 d/kg.

TRANSFER FACTORS OF CAESIUM INTO MILK AND MEAT

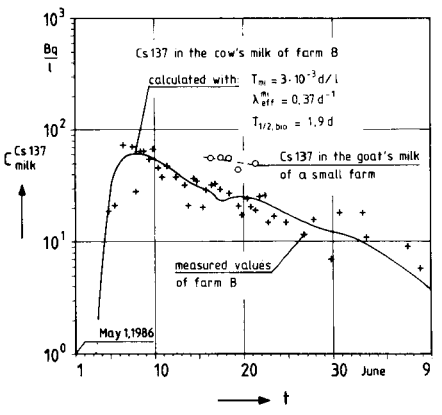

Fig. 2: Activity concentration of Cs 137 in cow's milk of farm B in Aachen

Fig. 2 and 3 show the change of the specific activity of Cs 137 in the cow's milk of farm B in the south of Aachen. The activity supply can be derived from Fig. 4. In June 1986 it was tried with the same method used for I 131 to calculate the transfer factor into cow's milk and the effective removal rate constant using the least squares method, see Fig. 2. The transfer factor is 0.003 d/l. In the frame of licensing procedures a value of 0.01 d/l is used in the Federal Republic of Germany. The measurements of the milk after June 1986 showed, that the transfer factor in milk became larger, see Fig. 3. Solubility tests on aerosol particles showed that only approx. 45 % of the caesium was soluble in distilled water. A transfer factor of approx. 0.007 d/l resulted for the two periods of September and October 1986 and 1987 when a certain equilibrium occured. During the feed with grass ensilage from Jan. 1 until April 25, 1987 with grass that was cut in the middle of June 1986, a transfer factor of approx. 0.003 d/l resulted similar to May 1986. In goat's milk the activity concentration of Cs 134 and Cs 137 was about a factor 1.4 higher than in cow's milk, see Fig. 2. In ewe's milk the enhancement was about a factor 1.5.

Concerning beef the information are not so detailed as for milk. Up to the end of 1986 we only analyzed the meat of animals, that had been slaughtered at the Aachen slaughter house. After that more meat of two butchers which sell only meat from animals from the Aachen area, was measured. Although the deposited activity was nearly constant in Aachen /2/, the specific activities

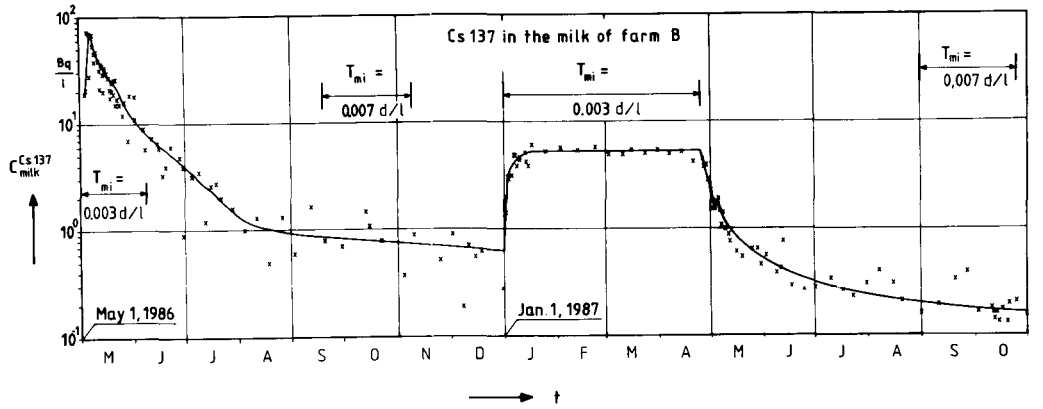

Fig. 3: Activity concentration of ^s 137 in the milk of farm B in the south of Aachen

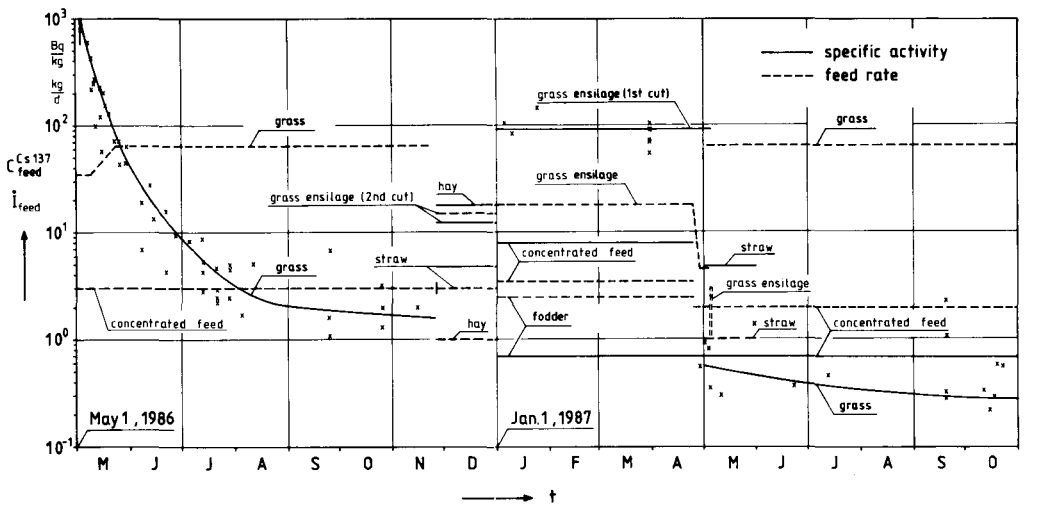

Fig. 4: Specific activity of Cs 137 in feed of cows and feed rate of the different feed of farm B in the south of Aachen

differ a lot. This is caused by the fact that in the Aachen area the animals are fed as well with grass ensilage of higher specific activity as with indian corn ensilage of turnip ensilage with lower specific activity. The development of the specific activity concentration in Fig. 5 and of the activity concentration in Fig. 3 shows that the transfer factor into beef is about 2.5 times higher than into cow's milk. If the caesium in the plants is totally available for the animals, the transfer factor is approx. 0.017 d/kg. The effective removal half-life of Cs 137 in the meat of beef is approx. 50 d. The specific activity in the meat of

bulls is up to a factor 5 higher than in cow's meat.

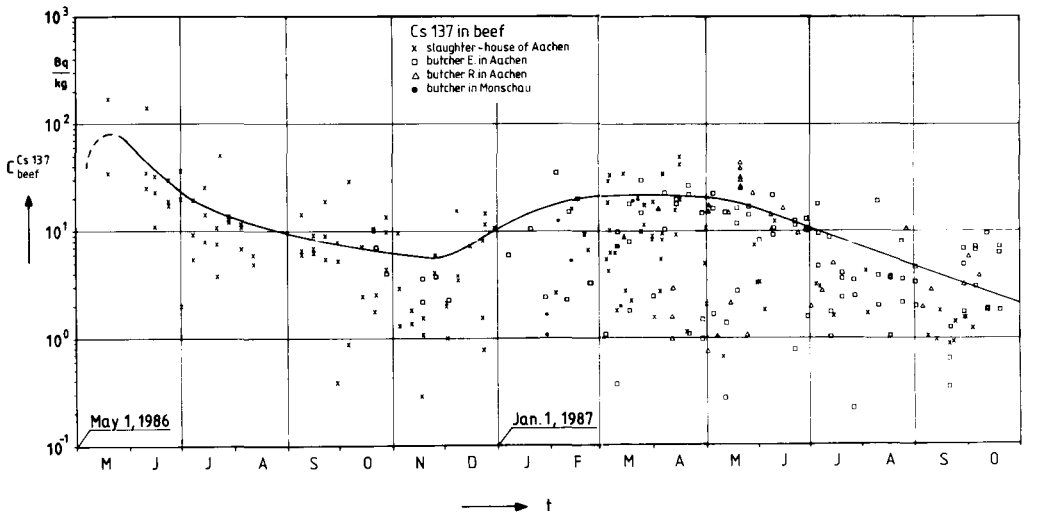

Fig. 5: Specific activity of Cs 137 in beef of animals of the vicinity of Aachen

Measurements of veal showed that the specific activity is approx. 1.5 times higher than in beef. The specific activity in pork was nearly constantly approx. 7 Bq/kg. In goat's meat the specific activity of caesium is approx. 1.4 times higher than in beef. The variation with the time of the specific activity of Cs 134 and Cs 137 in the meat of deers and wild boars is shown in /2/ for 1986.

The measurements were sponsored by the Bundesminister für Umwelt, Naturschutz und Reaktorsicherheit under the number St. Sch. 1026.

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RADIOLOGICAL IMPACT OF THE REACTOR ACCIDENT AT CHERNOBYL ON THE HUNGARIAN POPULATION

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ABSTRACT

The accident of the Chernobyl nuclear power plant at the end of April 1986 has resulted in the release of radioactive substances in considerable amounts. The meteorological conditions that prevailed at the time of the accident and during the subsequent days facilitated dispersion of the airborne pollution and contamination of the environment over almost the whole continent. The first contaminated air masses appeared over the territory of Hungary from north-east during the night from 29 to 30 April. A second, somewhat smaller and a third, somewhat higher contamination occurred from the south - south-east direction on the 3rd of May and between the 6th and 8th of May, respectively. In determination of the magnitude of the environmental contamination and assessment of its radiological significance as well as in elaboration of recommendations on the measures which were believed to be necessary for the protection of the public, surveillance systems and institutes of several national authorities and organizations have participated, including also the Radiological Controlling and Data Providing Network of the Ministry of Health. Results of radioactivity and radiation measurements, assessment of the situation developed and recommendations on protection of the population made by this system are summarized in this paper.

1. CONTAMINATION OF THE DOMESTIC ENVIRONMENT

Arrival of radioactively contaminated air masses was first detected in the northern and north-western regions of the country during the night from 29 to 30 April. A somewhat less pronounced second contamination occurred on the 3-4 May and a more significant third one on the 6-8 May from the direction of south and south-east.

With the aid of gamma spectrometry analysis, typical fission products were found in the aerosol samples, such as Zr/Nb-95, Mo-99, Ru-103, I-131, Te/I-132, Cs-134 and 137, Ba/La-140, Ce-141 and 144.

These airborne radionuclides settled down on the ground surface by either dry or wet deposition influenced by the local meteorological conditions and, in particular, by rainfall patterns. The site specific increments in dose rates of environmental radiation measured outdoors at 123 selected points of the country in the early post-accident period have given a reasonable good information on the geographical distribution of the surface contamination over the whole Hungary.

According to these measurements, the region of Budapest was one of the most heavily contaminated areas of the country. Thus, the results obtained in Budapest on the levels of radioactive contamination of the environment and food-stuffs could be taken as

representative values for the most contaminated parts of our country.

Time integrated radionuclide composition of the aerosol in Budapest can be seen in Table I.

Table I. Radionuclide composition of the aerosol and fallout in the region of Budapest

Sample	Radionuclides						
	Mo-99	Ru-103	I-131	I-132	Cs-134	Cs-137	La-140
Aerosol 29 April-8 May $Bq.d.m^{-3}$	1.28	13.95	11.20	16.25	1.89	3.48	0.72
Fallout 29 April-9 May $Bq.m^{-3}$	1.09	16.11	3.55^{+}	10.19^{+}	2.28	4.71	2.55

$^{+}$About 5/6 fraction of the radioiodine isotopes has been lost due to evaporation and incineration of the fallout samples.

In addition to the radioactive iodine absorbed to the airborne particulates (aerosol), about 1.5-2.5 times as much radioiodine was present in the atmosphere in vapour form. Accordingly, the maximum I-131 activities that could be inhaled in the region of Budapest by infants, children, adolescents and adults between 29 April and 8 May were about 150, 400, 750 and 1000 Bq, respectively, in case they spent all the time outdoors, and considerably lower if a significant fraction of time was spent indoors. Assuming that, in reality, at least 3/4-2/3 part of the day is spent indoors, the committed dose equivalent for the thyroid was about 100-150 uSv, and the committed effective dose equivalent about 3-4.5 uSv, at most.

The radioactive contamination of the atmosphere was washed out almost completely by a widespread rainfall during the night from 8 to 9 May. The radionuclide composition of the fallout can also be seen in Table I. This composition changed rapidly in the subsequent days and weeks depending on the half-life of each radionuclide. Gamma-emitting radionuclides deposited on ground surface were primarily responsible for the increase in the environmental dose rate observed in the early post-accident period.

The outdoor gamma dose rate in Budapest reached a maximum of 430 $nGy.h^{-1}$ on 1 May. This was about 4.5 times higher than the average background level during the preceding years. From the 2nd of May, it started to decrease rapidly. The country-wide average of environmental dose rate of 86.8 ± 10 $nGy.h^{-1}$ measured outdoors in the years of 1983-85 increased to 106.4 ± 22 $nGy.h^{-1}$ in 1986. No significant increase was detected in the indoor dose rates. The per caput increment in annual effective dose equivalent received by the Hungarian population from external sources in 1986 was estimated to be about 25 uSv.

2. CONTAMINATION OF WATER AND FOOD-STUFFS

Activity concentrations of radionuclides in surface and underground water resources remained so low throughout the whole post-accident period that contribution of drinking water to the exposure of the population was considered to be negligible.

During the first days of May, the same radionuclides could be detected in samples of grass and vegetables as in air and soil samples. Maximum concentration of I-131 of about 10 $kBq.kg^{-1}$ wet weight was measured in grass samples on 1-2 May. Concentration of Cs-137 in grass samples went up to 2-2.5 $kBq.kg^{-1}$ wet weight at the beginning of the month. These concentrations decreased fast to about 0.05-0.15 $kBq.kg^{-1}$ till the end of May. Radioactive contamination of vegetables was lower than that of grass by a factor of 2-3, at least, and was mainly attributable to surface contamination that could be partially removed by repeated and careful wash in running water. No contamination was found on vegetables grown in green-houses or under plastic tent. Relatively low and readily removable radioactive contamination was on the surface of fruit, such as cherry and strawberry.

Based on these contamination data, state farms and agricultural cooperatives were requested by the Government to discontinue grazing of animals, and the population was adviced to wash all leafy vegetables (lettuce, sorrel, spinach) carefully prior to consumption.

Grazing of farm animals in the early post-accident period resulted in a rapid increase in concentrations of I-131, Cs-134 and Cs-137 in fresh milk. Maximum concentration of I-131 was found to be about 1.5 $kBq.l^{-1}$ in the first days of May, and that of Cs-137 around 50 $Bq.l^{-1}$ between 5 and 15 May. Concentrations of these radionuclides remained significantly lower in milk collected from differently contaminated areas, but blended and put on market by the dairy industry. The maximum concentration of I-131 in milk marketed in Budapest was less than 200 $Bq.l^{-1}$ and that of Cs-137 less than 50 $Bq.l^{-1}$ all the time. Milk with a particularly high contamination level was withdrawn and either converted into milk products of a longer shelf-life or used for animal feeding.

No particularly high concentration of I-131 was detected in meat of animals, such as pig and cow, except in their thyroid glands, during the second half of May and the first half of June. Concentration of Cs-137 was 25-60 $Bq.kg^{-1}$ in beef and 10-70 $Bq.kg^{-1}$ in pork during the same time period. A rather slow decrease in Cs-137 concentration of meat could be observed thereafter.

Average concentrations of Cs-137 in cereals, such as wheat and rye, produced in various regions of the country in 1986 were about the same and fell within the range of 10-80 $Bq.kg^{-1}$. Radiocaesium was detected in much lower concentrations in crops, milk and meat products of the subsequent year in spite of the fact that the upper 5 cm layer of soil still contained about 50-60 $Bq.kg^{-1}$ Cs-137.

3. RADIONUCLIDES IN THE HUMAN BODY

Monitoring of 35 adults and 4 children during the first half of May, who had spent the days of radioactive contamination of the environment in various regions of the country, revealed that the maximum detectable content of I-131 in their thyroid glands was 0.4 kBq and that in their bodies was 0.93 kBq. Some other radionuclides, including Mo-99, Ru-103 and Cs-137 were just de-

tectable in a few cases.

From May 1986 to September 1987, additional 160 persons of both sexes and different ages have been examined in regard to the Cs-137 content of their bodies. Of these, 23 were adolescents of 12-17 years old. The results of these studies are given below.

Table II. Results of personal monitoring for whole-body content of Cs-137

Parameters	adults		adolescents	
	males	females	males	females
Number of persons	69	68	15	8
Number of measurements	142	144	16	11
Average Cs-137 content, Bq	1024	638	963	779
Average body-weight, kg	77.5	62.2	60.9	55.2
Average Cs-137 concentration, $Bq.kg^{-1}$	13.2	10.3	15.8	14.1

In 1986, the average body content of Cs-137 in both sexes increased steadily from about 530 Bq to 570 Bq. This increase became much faster at the beginning of 1987 resulting in a maximum of about 1050 Bq in July and followed by a rather steep decrease afterwards, reaching a level of somewhat less than 700 Bq in September.

A loose correlation could also be established between the body content of Cs-137 and its daily excretion rate in the urine. According to our measurements, 1000 Bq body content corresponds to an average excretion rate of 8.5 $Bq.l^{-1}$ of Cs-137.

Average concentration of Cs-137 was found to be about 2.7 $Bq.kg^{-1}$ in embryonic tissue, 4.25 $Bq.kg^{-1}$ in placental tissue, and 8.75 $Bq.kg^{-1}$ in the body of adult females in the period of June 1986 to April 1987. These data seem to suggest a discrimination ratio of 1:3 between embryonic and adult tissues and 1.2 between placental and adult female tissues.

4. INCREMENT IN EXPOSURE OF THE POPULATION

On the basis of the preceding data, the national average increment in the annual exposure rate of individual members of the Hungarian population in 1986 might have been about 100 uSv effective dose equivalent, somewhat higher for children and somewhat lower for adults. The maximum increments in the most heavily contaminated areas might have been about 2-4 times higher.

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BHOPAL AND CHERNOBYL - A COMPARISON

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ABSTRACT

Mankind witnessed two catastrophic industrial accidents in this decade, one in the chemical plant at Bhopal due to release of the toxic gas methyl-iso-cyanate, and the other in the nuclear power plant at Chernobyl. For both the events the extent of damage has been well documented. This communication attempts to compare the consequences of the two types of accidents and to critically examine how legitimate such a comparison is?

A software programme developed for a personal computer to predict quickly the plume propagation and exposure levels in the event of an accidental radioactivity release in air has been utilised to create mock nuclear reactor conditions with release characteristics corresponding to Chernobyl but meteorological and demographic parameters simulating those of Bhopal. The consequences of this simulated accident have been compared with those of the Bhopal disaster. It turns out that although a Chernobyl-type accident at Bhopal would have affected a larger area than in the case of the actual spread of MIC, the immediate death toll would have been much smaller.

Delayed effects of Chernobyl may be estimated from the committed dose equivalent using accepted risk coefficients. The corresponding situation for MIC release consequences is quite unsatisfactory. Preliminary data obtained by surveys of the Indian Council of Medical Research reveal incidence of delayed effects (like abortions, stillbirths and chromosomal abberrations) which are normally associated with radiation. A tentative attempt has been made to correlate the damage in terms of equivalent radiation doses.

THE IMPACT OF THE CHERNOBYL ACCIDENT ON NORWAY

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INTRODUCTION

As the fallout from the atmospheric nuclear weapons tests gradually decreased during the 1970s, the national preparedness and analytical capacity in Norway gradually disintegrated as well. The Chernobyl accident was therefore met without any overall contingency preparedness plan. The affected governmental bodies and other institutions had to improvise their first steps, including information to the public, until necessary coordination had been established. A complicating factor was the change of government during the first days of May 1986, the reasons for this had however nothing to do with the reactor accident.

A great deal of uncertainty prevailed about the accident and its consequences, especially during the first days after the accident. The Ministry of Health and Social Affairs and the Ministry of the Environment in May 1986 both appointed committees to report on the accident and its impacts and on a future preparedness system, although their given terms of reference were not identical. A third committe was appointed in June by the the Ministry of Health and Social Affairs to report on the "information crises" in connection with the accident.

AIR CONCENTRATION AND DEPOSITION ON THE GROUND

The peak concentration of the fallout from the Chernobyl reactor accident was first detected by permanent air monitor stations in the Oslo region in the morning of 28 April 1986. Probably contaminated air of lower concentration had already reached the country during the night. The maximum air concentration, 2-3 Bq Cs-137/m^3, was recorded during 28-29 April. The concentration of total I-131 at the same time was about 19 Bq/m^3. The existing air monitoring stations in other parts of the country had no facilities for immediate evaluation of the activity. It therefore took some time before the air concentrations outside the Oslo region were known, and that was heavily criticized in local newspapers. The air concentrations in the Trondheim region, and probably in other regions as well, had in fact been just as high as in Oslo.

The deposition pattern turned out to be rather complex, due to local precipitation in many areas while the air concentration was high. The deposition mapping was done by measurements of the direct gamma radiation from the ground, and also by soil sampling where four samples from each municipality should represent a mean value of the deposition in that municipality. In some municipalities the mean deposition was higher than 80 kBq/m^2, and certain areas exceeded 200 kBq/m^2. It has been calculated that Norway received about 1200 TBq of Cs-134 and 2300 TBq of Cs-137, corresponding to 6 % of the amount released from the reactor.

ACTION LEVELS

The only action levels put into force in Norway applied to radioactivity in food, concerning food sales only and not consumption of food supplied of one's own. At the earliest stage the levels for all types of food were 300 Bq Cs-137/kg and 1000 Bq I-131/kg. The iodine level remained unchanged, but the levels for radiocaesium were revised in June 1986 to be 370 Bq/kg for milk and baby-food and 600 Bq/kg for all other food, now including both Cs-134 and Cs-137. An increased level of 6000 Bq/kg was applied to reindeer and game meat in November 1986 and to fresh water fish in July 1987. The justification of the increase was that it would only lead to a modest incremental dose to the normal consumer.

FOOD MEASUREMENTS AND ACTIONS TAKEN

The radionuclides of concern in food were Cs-134 and Cs-137, only. All concentrations mentioned below refer to total radiocaesium in fresh food. The concentrations in dairy milk increased from the end of May until the beginning of August 1986 when they started to decrease. The monthly mean value of radiocaesium in milk from any local dairy did not exceed 100 Bq/litre in 1986, and all types of white cheese, butter and casein had negligible contents. Meat from cattle contained in 1986 generally less than 600 Bq/kg, and the content of pork was close to zero.

The number of sheep in Norway is about 2 millions, and during the summer these animals are normally grazing in the mountain fields of which many received considerable amounts of fallout. By their return to the villages in the autumn 1986, only 70 % of the sheep had a body content of radiocaesium below 600 Bq/kg, and 3 % had more than 2000 Bq/kg, some even above 15000 Bq/kg. The others were given a special diet of low activity fodder with certain additives. Within 4-6 weeks, the body content of most of these animals was less than 600 Bq/kg. Similar treatment was given in 1987.

Most of the domesticated and wild reindeer areas in Norway south of the polar circle were heavily contaminated. Some animals had a body content of nearly 90000 Bq/kg in the spring 1987, at the end of the lichen feeding period. The meat from domesticated reindeer from this part of the country will exceed 6000 Bq/kg for several years. Although Northern Norway had very little Chernobyl fallout, the radiocaesium content of its domesticated reindeer nevertheless exceeded 600 Bq/kg already in the autumn 1986. As a consequence, about 85 % of the country's annual production of domesticated reindeer meat would not have been allowed sold, probably for several years, if the action level had not been increased to 6000 Bq/kg. This might have had serious social consequences for the Lapp population and culture.

Cultivated Atlantic salmon and other marine products were very little affected by the Chernobyl fallout. The radiocaesium content of the salmon was below 10 Bq/kg in 1986. Also the content of cultivated fresh water fish was low all the time. The

situation for wild fresh water fish was on the other hand less satisfactory, and in July 1986 all sales of such fish from nearly 50 municipalities were forbidden. Trout or char from some lakes in the most heavily contaminated areas exceeded even 50000 Bq/kg at the end of the summer 1986. Great variations were however observed, both among individual fishes, species and lakes. About 25 percent of the Norwegians are engaged in gamefishing in some way during the summer season, and the high radioactivity content in the fish caused a considerable public concern.

The highest radiocaesium concentration found in human milk was 20 Bq/litre in a sample from a woman who lived in the worst contaminated municipality, but most of the samples contained less than 5 Bq/litre. Except from very early lettuce and parsley, the radioactivity in vegetables has been low all over the contry, and the content of cereals has not exceeded 5 Bq/kg. Most of the lake and river waters measured in June 1986 had concentrati ons below 5 Bq/litre. Cistern water could contain as high as 10 kBq/litre in some areas, mainly iodine-131, just after the fallout. The population were recommended not to drink such water for some weeks.

TOTAL EXPOSURE OF THE POPULATION

The estimated individual and collective dose equivalents to the Norwegian population (4 million) the first year after the Chernobyl accident are given in the table below and compared with the background radiation doses. The total collective dose for all years after the accident is expected to be 2900 manSv, compared to 6200 manSv if no action levels had been established.

TABLE. Estimated dose equivalents to the Norwegian population the first year after the Chernobyl accident

Radiation source	Individual dose (mSv) Mean	Range	Collective dose (manSv)
Natural background	1.2		4990
Radon	4.0		16600
Medical X-ray	1.0		4150
Nat. and med. rad. total	6.2	1.0 - 15.0	25740
Inhalation	0.01	0.01 - 0.06	43
External gamma	0.10	0 - 2.3	414
Food	0.16	0 - 31	663
Chernobyl fallout total	0.27		1120

In October 1986 the Directorate of Public Health issued some general dietary guidelines concerning consumption of food of high radiocaesium content. Some preliminary studies indicate that the guidelines have largely been followed. The real individual doses to members of population groups like hunters, anglers, and sheep- and reindeer-breeders are therefore much smaller than those expected if their normal food consumption habits had not been altered. The 31 mSv dose in the table is thus probably too high.

DIFFERENT OTHER IMPACTS

The lack of an efficient governmental information service which could meet the very large demand from the media and the population in the early stage, placed a heavy additional burden on the employees of the involved institutions. Inevitably some contradictory statements were given and obtained major headlines in the media. The rather low action levels, being not only based on radiological, but also on commercial and psychological arguments, were in news media often erroneously referred to as risk levels. It also happened that, on radiological grounds, some experts argued for higher action levels while other experts wanted still lower levels. This lead to a widespread confusion and fear among the population, who developed a serious lack of confidence in the official statements and actions.

People were not adviced to stay indoors during the period of high air concentrations of radioactivity, and the cattle was let out as usual at the end of May without any restrictions. This was also criticized.

The exportation of cultivated Atlantic salmon was stopped by some of the importing countries, until it had been proven that the salmon was not affected by the Chernobyl fallout. 2300 tons of mutton and 550 tons of reindeer meat had to be discarded in 1986, as the content exceeded the action levels. The cost of the special feeding treatment plus the value of the discarded meat was about 150 million NOK or 23 million USD. In 1987 it is expected that such actions will cost about 70 million NOK. The reduction is partly due to the results of intensive research, including the use of an in vivo animal counting technique.

The tourist industry faced a reduction of visitors from certain countries as high as 42 % during the summer of 1986. In 1987 the number of foreign visitors was more like normal again.

CONTINGENCY PREPAREDNESS

The radiation contingency preparedness in Norway was before the accident mainly concentrated on emergency plans and monitoring systems in connection with the two research reactors in Kjeller and Halden. As a part of a nationwide early warning system, a network of ionization chambers has now been connected to the environmental data network system of the Norwegian Institute for Air Research. In September 1986 all main food control laboratories were furnished with modern equipment for gamma spectrometric analysis.

The committees having studied a future contingency preparedness system in Norway in light of the lessons learned from Chernobyl and other possible nuclear accidents, have presented their recommendations. These are now being assessed by the different governmental bodies involved.

NATURALLY OCCURRING RADIONUCLIDES AND FISSION PRODUCTS IN MEAT AFTER THE CHERNOBYL ACCIDENT

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ABSTRACT

The content of naturally occuring radionuclides and fission products in different anatomic regions of meat from cattle, pigs and chickens were examined. Results obtained for ^{40}K, ^{210}Pb, ^{210}Po, natural U, ^{134}Cs, ^{137}Cs, ^{103}Ru, ^{106}Ru, ^{90}Sr, ^{241}Am and others radionuclides in different meat spaces and anatomic regions of meat were compared.

Obtained results of examination showed that content of investigated radionuclides in meat depends on anatomic region and spaces of meat, as well as on the age, feeding condition, geografical area and season.

THE RADIONUCLIDES LEVELS IN HUMAN BODY AFTER THE CHERNOBYL ACCIDENT

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ABSTRACT

Population exposed to fallout from the Chernobyl nuclear power plant were selected by duration of exposure from differnt fallout area, begining from 150 km ap to 1000 km of Chernobyl accident point. The retained radionuclides activity in selected population were studied by whole body counting and radiochemical analysis of urine. Obtained results showed, at the begining a significant concentration of ^{131}I, but later dominants radionuclides were ^{134}Cs, ^{137}Cs, ^{103}Ru, ^{106}Ru, ^{90}Sr and some others. Results demonstrated that the internaly deposited radionuclides are maintained at a relatively high levels by the ingestion of foodstuffs containing radionuclides from fallout.

HEALTH EFFECTS OF CHERNOBYL ACCIDENT EXPOSURE

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INTRODUCTION

During the accident at the fourth reactor unit of the nuclear power plant in Chernobyl there was released a large amount of radioactive material and a vast area around the plant was contaminated. Most of inhabitants living in this area have traces of radiological contamination (1).

The aim of the present study was to analyse a group of persons, who were present in a circle of 60 miles range from the relevant point at the moment of accident, in a following ways:

- to determine the level of radioactive contamination, using radiotoxicologycal analyses of 24 hours urine samples.
- each subject had to pass through the extensive clinical examination and haematological analyses
- to analyse structural chromosome aberrations in the peripheral blood lymphocytes.

SUBJECTS AND METHODS

The subjects in this study were students: two female and four male, age 20 to 21, non-smokers. They hawe been provided with food in a public restaurant.

First days of May, 1986. they were walking through the city, attending the May Day Parade, and watching the bicycle race. These days subjects 1,2 and 5 got wet through from the rain.

After their arrival back home to Yugoslavia, they asked for a medical examination or just consultations.

All 6 subjects underwent a medical check-up, haematological tests, conventional structural chromosome aberration analyses (2) and bioasay (24 hours urine samples) (3).

RESULTS AND DISCUSSION

During a medical examination non of 6 subjects felt any healthy problems except slight auxiety and fear.

Haematological analyse results are presented in table 1.

TABLE 1 Haematological analyses

Subject	Leucocyte G/l	Thrombocyte G/l	Reticulocyte
1	4.8	300	0.009
2	4.9	254	0.004
3	3.8	207	0.007
4	6.9	300	0.004
5	5.9	226	0.008
6	4.4	155	0.006

White blood cells, including lymphocytes didn´t show significant deviation from the normal values. The same is with the thrombocytes, excluding subject 6 whose thrombocyte number is lower. Immature cell forms - reticulocytes were present in all subjects but particulary in persons 1, 3 and 5.

Radiotoxicological analyses of 24 hours urine samples showed the presence of technologically produced radionuclides. In two subjects checked during the first days of June, 1986, we found ^{131}J, ^{134}Cs, ^{137}Cs, ^{103}Ru, ^{106}Ru. ^{134}Cs and ^{137}Cs were dominate in analysed 24 hours urine samples of all 6 subjects. There were prooved traces of the following elements: ^{241}Am, ^{226}Ra, ^{232}Th, ^{140}La, ^{144}Ce, ^{95}Li and ^{90}Sr for subjects 3, 4, 6. (Table 2).

Table 2 Radiotoxicologycal analyses of 24 hours urine samples

Subject	RADIONUCLIDE Bq				
	^{131}J	^{134}Cs	^{137}Cs	^{103}Ru	^{106}Ru
1	28.3	35.7	74.8	4.9	15.7
2	30.24	36.4	75.0	5.07	14.62
3		24.6	50.2		
4		38.3	70.6		
5		18.6	40.22	4.89	15.68
6		20.46	38.86		

Chromosome aberrations frequency analyses were done on first in vitro methaphases. These results are shown in table 3.

Table 3 Chromosome aberration analyses

Subject	No scor.	CHROMOSOME ABERRATIONS - PER CELL			
	Cells	Break	Acentric fr.	Dicentric	Ring
1	200	0.07	0.04	0.02	
2	410	0.02	0.02	0.01	0.002
3	220	0.03	0.04	0.006	
4	200	0.04	0.03	0.005	
5	500	0.02	0.02	0.008	0.002
6	420	0.03	0.02	0.01	

In all subjects there's evident presence of chromosome breaks, acentric fragments and dicentric and ring chromosomes. There are some interesting cells with multiple chromosome aberrations, such as two dicentrics and more acentric fragments in one methaphase (Table 4). Considering the estabiliched background in laboratory of 1‰ dicentrics per cell for population unexposed to ionising radiation, such results are pointing to the significant raising of these specific aberrations (4).

Table 4 Distribution of chromosome aberrations

Subject	Cell scored	DISTRIBUTION ABERRATIONS PER CELL				
1	200	181	10	5	3	1
2	410	385	18	6	0	1
3	220	206	8	4	2	0
4	200	183	9	4	3	1
5	500	467	21	6	4	2
6	420	401	14	3	2	0

Assuming all given results we suppose that subjects had been seriously jodine contaminated, but unfortunately these measurings haven't been done for all of them. Besides, there could have been some of external irradiation too, which might have contributed the frequency of somatic cell mutations.

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This work was supported from Self Managing Community of Interest for Science of SRH-SRS and IAEA - Contract No 3165/R4/RB.

INVESTIGATION OF INTERNAL DOSE RECEIVED BY THE HUNGARIAN POPULATION DUE TO CESIUM RADIONUCLIDES AS A CONSEQUENCE OF THE CHERNOBYL ACCIDENT

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INTRODUCTION

Hungary was one of the countries affected by radioactive environmental contamination due to the reactor accident at Chernobyl. The total ground deposition of radionuclides showed a non-uniform distribution over the country. According to a nationwide survey the activity surface concentration varied within about an order of magnitude from place to place. The most characteristic local averages for ^{137}Cs ranged from 1 to 5 $kBq.m^{-2}$. An extended internal contamination monitoring programme was introduced by whole body counting very soon after the accident. The subjects to be measured systematically were selected from Budapest's residents representing both sexes and different age groups. The total number of subjects monitored regularly (control group) was 44. The representativity of the control group for adults was checked by comparing the results with those obtained for large number of subjects measured occasionally. The programme was later extended also to other parts of Hungary by single reference measurements. Six weeks average monitoring interval was chosen for following the whole body activities due to ^{134}Cs and ^{137}Cs radionuclides, which are responsible for the long term internal dose received by the population. The results obtained for adults are presented here.

MEASUREMENTS

The measurements were carried out in the low background whole body counter of the institute using single detector scanning arrangement. Multi-element bottle phantoms were applied for activity calibration simulating the different shape and body sizes of the subjects to be monitored. The gamma spectrometric measurements were evaluated by least-square fitting decomposition procedure providing the whole body activities of ^{134}Cs, ^{137}Cs and ^{40}K radionuclides. The reliability of the measuring, calibration and evaluation methods applied has been checked by international intercomparison measurements and the results were found to be very satisfactory.

RESULTS

The statistical investigation of the whole body activities showed log-normal frequency distribution. For this reason the value of the median was used in the interpretation of the measurements. The ratio of the mean to median was found to be about 1.10. The individual ^{137}Cs body contents in Budapest for adult males and females against the time after the accident are presented in Fig.1. Considerably large differences can be observed between the individual activity data. The solid lines in the figure represent the medians for both adult sex groups. The feature of the line course of ^{134}Cs activities is very similar to that shown for ^{137}Cs but influenced also by the shorter physical half life. The ratio of whole body ^{137}Cs to ^{134}Cs activities was found to be 1.83 in May 1986.

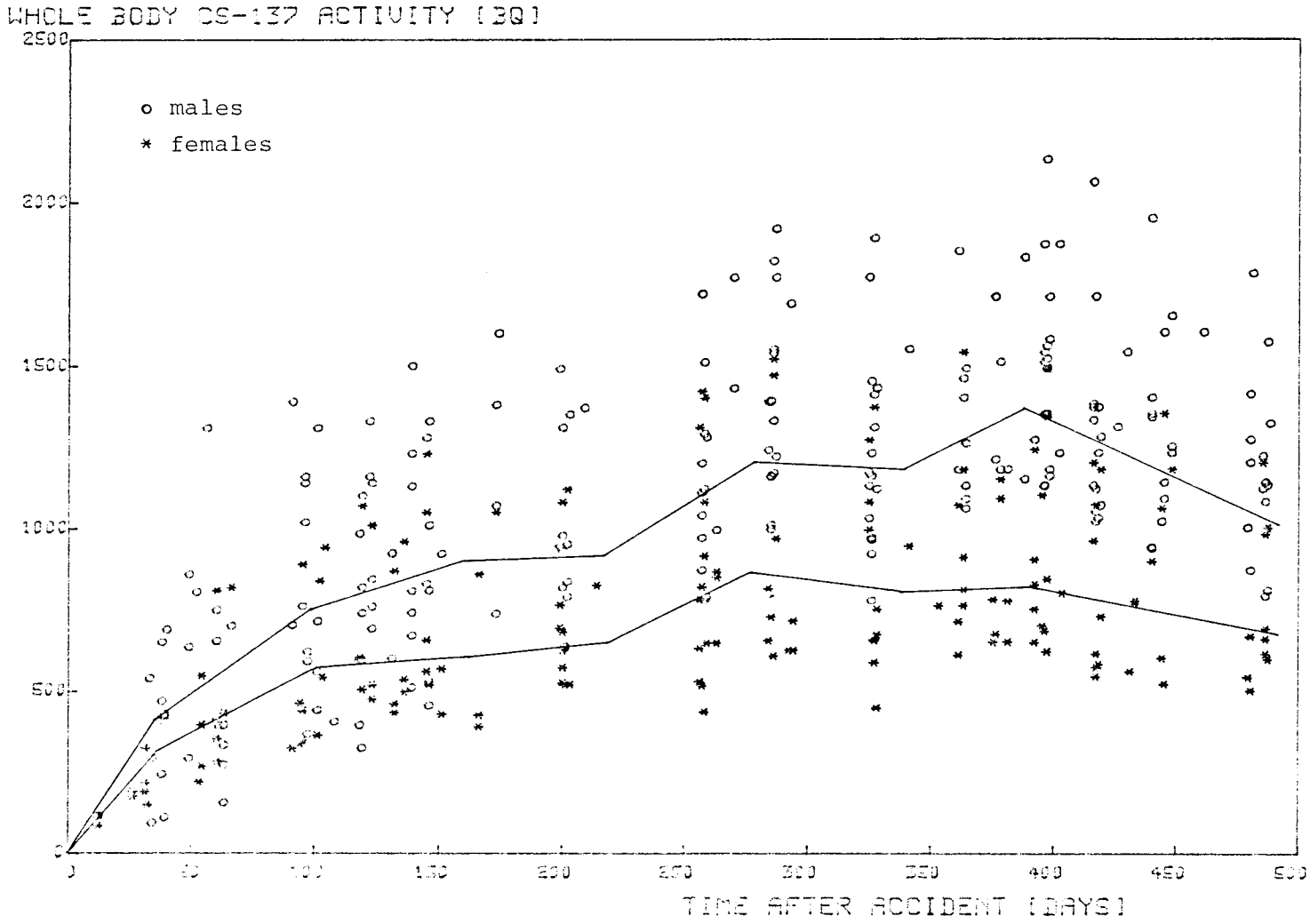

Figure 1. Whole body ^{137}Cs activity in Budapest's residents against the time

The dose rate due to the cesium radionuclides has been calculated for each individual by the formula

$$H(t) = \frac{q(t)}{M} \sum_i Y_i \cdot E_i \cdot \phi_i(M)$$

where $H(t)$ is the dose equivalent rate at time t,
$q(t)$ is the whole body activity for the given radionuclide at time t,
Y_i is the yield per transformation of the i-th type of radiation,
E_i is the average energy of the i-th radiation emitted,
$\phi_i(M)$ is the absorbed fraction of E_i energy for a given body mass M.

For cesium radionuclides both the source and target organs were assumed to be the whole body. The absorbed fraction of gamma radiation varies considerably with the total body mass. The mass dependence was taken into account individually.

CONCLUSIONS

The time variation of the medians of dose equivalent rates for males and females due to both ^{137}Cs and ^{134}Cs are shown in Fig.2. The whole body internal dose equivalents calculated for the first year after the accident are summarized in Table 1. where the values associated with the 5-95% interval of the frequency distribution are also indicated in parentheses. The local variation of the internal dose over the country was estimated by single reference measurements on 5-10 subjects from each place. The results are shown in Table 2. The differences which were found to be within a factor of 4, are in good correlation with those obtained for the local average ground deposition of these radionuclides over the country. The roughly predicted whole body 50 years dose due to the internally deposited cesium radionuclides will not exceed 100 μSv as an average in Budapest area. These follow up studies will be continued and extended also for children.

Table 1. Internal dose of Budapest's adult residents in the first year after the accident due to cesium radionuclides

	WHOLE BODY DOSE EQUIVALENT [μSv]	
	males	females
^{137}Cs	27.0 (12-65)	21.6 (10-40)
^{134}Cs	16.4 (7-38)	14.1 (7-24)
SUM	43.4 (19-103)	35.7 (17-64)

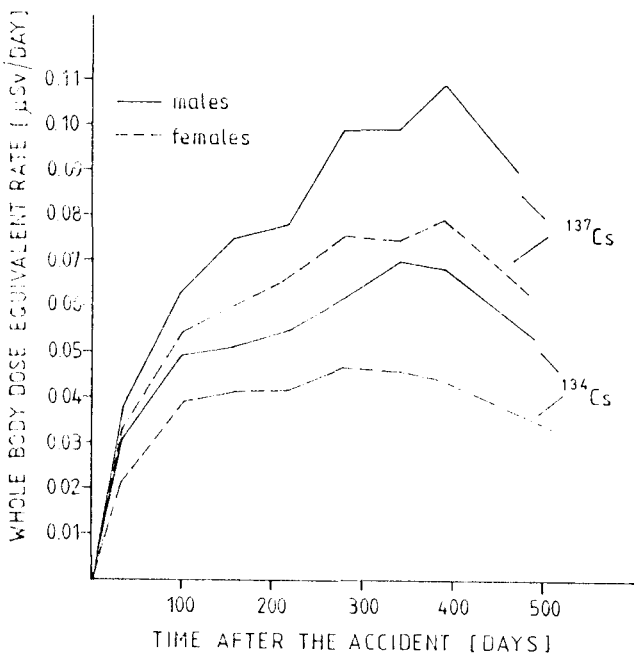

Figure 2. Internal dose equivalent rate against the time for Budapest's adult residents due to cesium isotopes

Table 2. Comparison of the total internal dose received by males living in different counties of Hungary due to cesium isotopes

COUNTY	WHOLE BODY DOSE EQUIVALENT [μSv]
Budapest	43
Szabolcs-Szatmár	26
Zala	63
Heves	54
Csongrád	35
Győr-Sopron	104
Tolna	32

THE CONTENTS OF RADIONUCLIDES Ru-103 AND Ru-106 IN FOOD

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The paper presents the results of radionuclides Ru-103 and Ru-106 contents evaluation in different foodstuff : milk and dairy products, meat, vegetables, fruit and honey. Samples came from some European countries after the nuclear accident in Chernobyl, from May to December 1986. The contents of the radionuclides in the fresh samples of food was determined on a Ge(Li) detector by the means of standard gamma spectroscopy.

The results indicate the activity of Ru-103 and Ru-106 in the investigated samples of food sampled during May and June 1986, to range up to 10^2 Bq/kg , except for some vegetable - the contents found to be 10 - 10^2 times larger. In food sampled in November and December 1986, the contents of Ru-103 was no longer within the measurable range, whereas the activity of Ru-106 was between $2x10^1$ and $5-8x10^1$ Bq/kg.

Considering only a few data about the metabolic ways of ru - tenium inside body , one can see that there is a need for sistematic and continuous study of the contents of these radionuclides, especialy Ru-106 , in food.

INTRODUCTION

The time dinamics of the Chernobyl accident in 1986 , followed the similar nuclear accidents over the world : on the first, at the end of April and during May 1986, the magnitude of contamination of the biosphere was basically determined by short-lived radionuclides , J-131 above all. Later on, as J-131 decayed, the comparatively short-lived radionuclides (Sr-89 ,

Zr-95, Nb-95, Ce-141, Ru-103, Ba-140) made an appreciable contribution to the radioactivity on the whole and finally, the main role in the process of contamination overtook the long-lived radionuclides : Sr-90, Cs-134,137 and partialy , Ru-106. But the composition of the radionuclide release was specific to some point [2]: about 3-5% of the relatively refractory elements (Sr,Ru,Pu) escaped from the reactor, that was much more than would be expected in a light-water reactor core melt. Besides, although ruthenium is less volatile then strontium , it appeared in appreciable quantities in filter samples throughout Europe, while strontium did not.

So far , ruthenium was not considered as a biologically significant radionuclide and the data on the mechanisms of it metabolism are only few, even though there are indications of the biochemistry similar to one of iron. All that points to the need of determination of contents of Ru-103,106 in food , so as the level of ruthenium intake in the human body could be evaluated.

MATERIAL AND METHOD

Sampled originated from different localities over Europe: from a period imediately after the accident (May-June, 1986) and some time after, at the end of 1986. The contents of the radionuclides Ru-103,106 in the fresh samples of food was determined on a Ge(Li) detector (ORTEC,USA) and a 4096 channel analyzer (NUCLEAR DATA ND-100).

Calibration was performed with a point etalon source Eu-152 (ET-75220 EGMA3) for the energy range of 121.78 - 1403.08 keV. The efficiencies for the measuring geometries (nonstandard Marinelli bottle 0.6 l and 200g PVC cilindric box) and different foodstuff were determined by "secondary standards" for fluid and solid state radionuclide carriers[1].

The activities of Ru-103 and Ru-106 were determined for the energies of 497.5 keV and 621.8 keV, respectively.

Table I . The contents of radionuclides Ru-103 and Ru-106 in food

Sample	Number of samples	Period of sampling	Activity* (Bq/kg) Ru - 103	Ru - 106
milk	27	May - June'86	1.7-11.7	2.4-10.4
cheese	22	May - June'86	1.4-93.0	8.0-82.1
	4	Nov.-Decem'86	-	19.0-59.3
milk powder	17	May - June'86	6.8-90.2	31.0-52.4
	2	Nov.-Decem'86	-	-78.2
yogurt	3	May '86	5.0-11.4	4.5-23.4
fruit and vegetable	12	May '86	3.5-11200	24.5-3570
honey	4	May - Sept'86	38.0-67.3	24.2-50.0
Meat:				
lamb	16	May - June'86	12.5-33.0	24.2-32.0
pork	21	May - June'86	3.5-9.8	15.2-20.5
	45	Nov. - Decem'86	-	11.0-16.4
beef	23	May - June'86	0.6-1.0	15.4-27.0
	6	Nov.-Decem'86	-	20.2-25.3
rabbit	8	July-Avg.'86	2.4-5.0	19.0-91.2
	6	Nov.-Decem'86	-	10.5-18.8
fish	5	May '86	9.4-26.5	16.0-32.8

* activities are presented in Bq/kg except for milk - in Bq/l; the table presents ranges of activities of Ru-103,106 in different foodstuff in the period of sampling ; the mean error is $\pm$ 15% , on the average

RESULTS AND DISCUSSION

The results of the radionuclides Ru-103 and Ru-106 contents evaluation in different foodstuff after the nuclear accident at Chernobyl in 1986, are presented in Table I. In the table there are presented the ranges of the activities of Ru-103 and Ru-106 for different samples of food in the two sampling periods.

The maximum contents of radionuclides Ru-103 and Ru-106 was found in rabbit, lamb, mutton milk and mutton milk dairies, as it could be expected considering the cattle feeding in that period of year. The extrimely high activities of ruthenium in some vegetable , green latice above all, immediately after the accident were due to surface contamination. At the end of the year, in a shortly established steady state, the contents of Ru-106 was found to be in a relatively narrow range, no matter the food. That indicates a specific mehanism of metabolism and distribution of this radionuclide inside the body and over the food chain.

The obtained results are in good correlation with the contents of radionuclides Cs-134,137 in food sampled in the same period.

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CHERNOBYL - A RETROSPECTIVE REVIEW TWO YEARS AFTER THE ACCIDENT

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SUMMARY The world's worst nuclear disaster - the only one which caused immediate loss of life and may, possibly have a small but significant biostastical health effects over a period of some thirty to forty years - was initiated during the early morning hours of 25 April, 1986. Its cause according to remarkably frank reporting by a Soviet delegation to the International Atomic Energy Agency at a meeting in Vienna in August 1986 was criminal negligence on the part of the operating staff, the failure of the plant supervisors to make responsible emergency decisions and the neglect to report the gravity of the incident to the headquarters of the Soviet Atomic Energy Ministry for some thirty hours after commencement.

1. INTRODUCTION

The author had the privilege, through the offices of the I.A.E.A. to be the first Western nuclear scientist to visit the Chernobyl district in July 1986 to make an assessment of the cause and consequences of the accident. Since his original reports were filed much has been published in both the technical and popular press, including a number of papers by the author. In the ensuing discussion we propose to consider four major aspects of the incident -

(1) A brief summary of the possible causes of the accident, including a description of the nuclear plants of the R.B.M.K. type.

(2) The human errors and man-machine interface deficiencies which had such disastrous consequences.

(3) The resulting short term and long term consequences of the chemical explosions in the nuclear plant and their impact on the residents of surrounding districts, more distant communities and the long term effects on agriculture and human life in Eastern bloc and Western European countries.

(4) The effects of Chernobyl on the public perception of nuclear energy and the impact of the accident on the long term future of the nuclear industry, world wide.

2. CAUSES OF ACCIDENT (Ref 2)

The design of the RBMK reactor was initially attractive to the U.S.S.R. for a number of reasons. As stated in the translation of the Soviet report, these include

> ...the absence of cumbersome pressure vessels which are difficult to manufacture and limit the reactor's unit power and production base; absence of a complex and costly steam generator; the possibility of continuous refueling and a

Table 1 Physical Characteristics

of Chernobyl Unit 4

Thermal power, MW	3200
Fuel enrichment, %	2
Mass of uranium per fuel assembly, kg	114.7
Number of fuel channels	1661
Total mass of graphite moderator, kg	1.7×10^6
Core diameter, m	11.8
Core height, m	7
Circumferential graphite reflector thickness, m	1
End graphite relector thickness, m	0.5
Steam pressure in separator, MPa	6.86
Coolant average core inlet temperature, ^oC	270
Coolant average core exit temperature, ^oC	284
Coolant total flow at full power, kg/h	3.76×10^7
Flow per pump, kg/h	6.27×10^6
Average full power steam quality at core exit, %	14.5
Maximum channel steam quality at exit, %	20.1

good neutron balance; a flexible fuel cycle easily adapted to the fluctuations of the fuel market; the possibility of nuclear steam superheating; high thermal reliability and durability of coolant flow, channel failure detection, monitoring of the parameters and coolant activity in each channel and on-load replacement of leaking assemblies.

Three major shortcomings of the design are: (1) the sensitivity of the neutron field to reactivity perturbations leading to control difficulties and requiring complicated control systems, (2) a relatively small and weak containment around the core itself, and (3) a positive void coefficient of reactivity that increases as power value of 0.02 δ k/k (where δk is the change in void fraction). Because of the enhanced positive void coefficient at low power and because of control difficulties associated with lack of accurate power measurement capability at the lower levels, continuous operation of the RBMK-1000 reactors is prohibited (by the Technical Specifications) at power levels less than 700 MW(t).

For Western experts in the fields of reactor dynamics and control, it is difficult to understand why staff with responsibility for such tests have not had the opportunity to simulate them on computing apparatus. It has however been made clear by the Soviet Union - even in general publications in

Pravda - that the first nuclear plant simulators were not available in the Soviet nuclear industry until 1972. It is very unlikely that a simulator had ever been designed for the RBMK-1000.

Hence it appears that the tests described in the Appendix were a "disaster waiting to happen" as six fatal errors were perpetrated.

1. The emergency cooling system was turned off to conduct the test.
2. The reactor power output was inadvertently lowered too much, making it difficult to control.
3. All water circulation pumps were turned on, exceeding recommended flow rates.
4. The automatic signal that shuts down the reactor if the turbines stop was blocked.
5. The safety devices that shut down the reactor if steam pressure or water levels become abnormal were turned off.
6. Almost all control rods were pulled out of the core.

3. CHERNOBYL ACTIVITY RELEASE (Ref 3,4,3,5)

The destruction of the Chernobyl nuclear power station core led to the release of millions of curies of radioactive material. This should be compared to 15 curies for Three Mile Island (1 curie 3.7×10^{10} Becquerel) and about one million curies at Windscale in the United Kingdom in 1956. The consequences of this within the USSR were catastraphic.

1. Thirty one known deaths.
2. Two hundred and three cases of hospitalisation.
3. Over 100,000 people received elevated radiation doses.
4. Over 50,000 people were evacuated from homes which are still uninhabitable.
5. Some 2000 sq. km. of land were badly contaminated.
6. Economic distress was suffered by people within a radius of 1000 km from Chernobyl.

Sweden, in particular suffered heavy contamination causing many forms to be quarantined, food to be abandoned and remainder to be destroyed. Spikes of fallout contamination after rain caused distress in Poland, Finland and even Scotland. However it is most unlikely that the contamination of the food cycle in these countries would cause any detectable excess mortality or life shortening experiences due to cancer. With a natural cancer mortality of around 25% in most industrialised countries, the excess from Chernobyl fallout would be very small and hard to predict. However, the disaster on which probably more radioactivity was released than Hiroshima and Nagasaki combined,

will be monitored closely and will provide a unique control population for studying the biostatistical response of human and animal populations to a unique event.

4. GLOBAL RESPONSE TO CHERNOBYL

Approximately twelve months after the Chernobyl accident, the nuclear industry appears to have recovered from a serious setback. Public opinion in most European countries, including Sweden, recognises that the risk-consequence phenomena associated with the nuclear fuel cycle is still considerably less than other societally accepted risks.

It is also apparent that countries with established nuclear power programmes - such as the U.S.A., Great Britain, France, Belgium, Sweden, Canada and Japan - have assessed that the Chernobyl disaster would be most unlikely to occur in nuclear power plant of Western design. There appears to be a concerted move to induce the I.A.E.A. to keep a closer watch on the operation of Soviet designed R.B.M.K. and V.V.E.R. reactors and to formulate plans for international litigation and fiscal reparation in such cases of accident.

For Soviet industry, the shutting down of its R.B.M.K. reactors and retrofitting and redesigning the equipment is a financial impossibility. It would bring to a halt much of the industrial production of European Russia. The best one can anticipate from this country is probably improved operator training and hopefully better design and multiple containment for her new nuclear power stations. (Ref 6.)

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ACKNOWLEDGEMENTS

The author is deeply grateful to his colleagues at the International Atomic Energy Agency and specialists in nuclear power plant safety in the United Kingdom, the United States and Australia for useful discussion and the supply of unpublished documents.

DIRECT AND INDIRECT RADIOTOXICOLOGICAL INVESTIGATIONS ON PEOPLE COMING FROM EASTERN EUROPE AFTER THE CHERNOBYL ACCIDENT

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Starting from May 2,1986 many Italian tourists and some Italian workers living in Eastern Europe during the Chernobyl accident were checked for internal radiocontamination at the Medical and Radiotoxicological Service of C.R.E. Casaccia,ENEA,Rome.

As the local W.B.C. was overloaded with many measures to be done within a few days, a simple equipment was installed for a rapid detection of I-131 in thyroid. This equipment consisted of a collimated and shielded 3"x3" NaI(Tl) detector having a sensitivity of 37 Bq for a 10' counting time (Fig.1);a counting efficiency of 3.7 % was calculated with a calibrated I-131 source having the thyroid geometry.

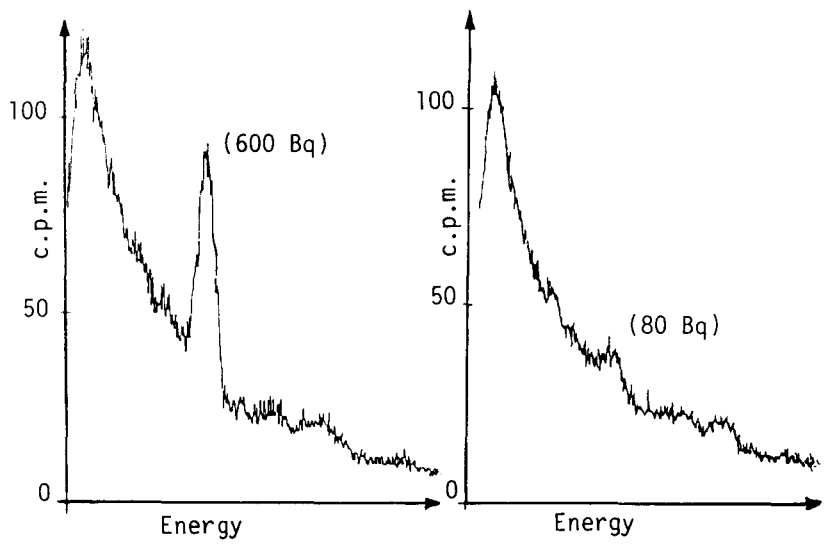

Fig. 1 Gamma spectra of I-131 in thyroid (600 and 80 Bq)

Only people showing a thyroid burden higer than 370 Bq were sent to the W.B.C. for longer and more accurate measurements. The thyroid burden ranged from 37 to 3330 Bq at the measurement time depending on the geographic region and on the winds direction.

DIRECT RADIOMETRIC RESULTS

As Table I shows, the checked people could be subdivided into four groups, namely : 1) Kiev region ; 2) Poland; 3) Hungary - Czechoslovakia - Austria - Rumania - Eastern Germany; 4) Moscow - Leningrad. To these groups a fifth group was added consisting of people living in Rome. Furthermore the I-131 burden at the moment of the probable acute contamination was calculated taking into account the cloud transit time on the region (1).

TABLE I
I-131 THYROID BURDEN (Bq) IN FIVE GROUPS OF PEOPLE COMING FROM DIFFERENT GEOGRAPHIC REGION

Gr n.	Region and number of checked people	Measure date interval (May 1986)	Mean value at the measure date	Radioactive cloud transit date	Mean corrected value	Ratio Gr.n / Gr.4
1	Kiev (11)	8-12	1594	27/4/86	4972	26.1
2	Poland (46)	3-21	548	30/4/86	1266	6.7
3	Other East Countries (77)	3-19	229	30/4/86	502	2.6
4	Moscow Leningrad (37)	4-12	81	30/4/86	190	1.0
5	Rome (35)	6-21	135	1/5/86	350	1.8

As the winds direction was North-West from Chernobyl in a first time and South-West in a second time, it is not surprising that people living in Moscow-Leningrad were less contaminated that the ones living in Rome.

The I-131 burdens calculated by this simple device were found in good agreement with the ones obtained by some Italian W.B.C. systems located in Rome,Bologna, Bosco Marengo, Ispra (2-5).

The thyroid committed doses calculated for the five group were respectively : 7.96 ; 2.03 ; 0.80 ; 0.30 ; 0.56 mSv.

INDIRECT RADIOTOXICOLOGICAL RESULTS

Some indirect radiotoxicological analyses (urine and faeces) were carried out on people showing a sensible thyroid contamination, i.e. gamma spectrometry with Ge(Li), Sr-90 and Pu-239 (Pu-240).As table II shows the following gamma emitters were detected in the urine: I-131, Cs-137, Cs-134, Te-132, Ru-103, while Sr-90 and plutonium concentrations were found to be lower than the detection limits. The above said radionuclides, more Ba-140 and La-140, were found in group 1 faeces, while plutonium was ever lower than the detection limit (Table III).

TABLE II
URINARY EXCRETION MEAN VALUES (Bq)
(number of analyses)

Gr	Date May '86	I-131	Cs-137	Cs-134	Te-132	Ru-103	Sr-90	Pu-239 (240)	Cs-137 / Cs-134
1	8-14	17.0 (10)	5.2 (10)	3.0 (2)	6.5 (10)	13.6 (3)	<4 E-2 (5)	<4 E-4 (4)	1.9 (2)
2	2-14	37.0 (29)	4.1 (29)	———	15.1 (29)	———	———	———	———
3	3-8	26.4 (11)	1.2 (11)	———	10.8 (11)	———	———	———	———

TABLE III

FAECAL EXCRETION MEAN VALUE (Bq) FOR GROUP 1 (DATE 9-14 MAY 1986)
(number of analyses)

I-131	Cs-137	Cs-134	Te-132	Ru-103	La-140	Ba-140	Pu-239 (240)
4.4 (2)	1.7 (4)	0.7 (4)	8.5 (2)	4.6 (4)	21.8 (1)	10.4 (1)	<2 E-3 (3)

RATIOS	Cs-137(F) / Cs-134(F)	I-131(U) / I-131(F)	Cs-137(U) / Cs-137(F)	Te-132(U) / Te-132(F)
	2.1 (4)	8.5 (1)	1.6 (1)	0.8 (2)

The ratio Cs-137 / Cs-134 in urines and faeces resulted to be about 2, as found in many other contaminated environmental samples. The ratios Urinary Excretion / Faecal Excretion for I-131, Cs-137 and Te-132, also reported in Table III, were found to be within the values reported in the literature (6).

In many cases it was possible to calculate the actual ratio : I-131 in thyroid / I-131 in the 24 hour urinary excretion (Table IV) obtaining a mean value equal to 62. However it was necessary to correct this value by subtracting the relevant activity found in thyroid and urine of the Roman population which also suffered a small internal contamination by inhalation and ingestion. As a matter of fact the direct and indirect checks were carried out after a some days staying in Rome. As an example Fig 2 shows the I-131 urinary excretion for the local population in the period 3-19 May 1986. The corrected ratio reaches the value of 116 which is quite similar to that (~150) reported in the literature (6).

TABLE IV

RATIO $\frac{\text{I-131 (THYROID)}}{\text{I-13I (URINE)}}$

Group	n.	Date May '86	Uncorrected value	Value corrected by subtracting the Rome blanks
1	10	8-12	89	104
2	12	3-17	57	143
3	7	3-6	41	100
Mean (29)			62	116
Weighted Mean			64	119

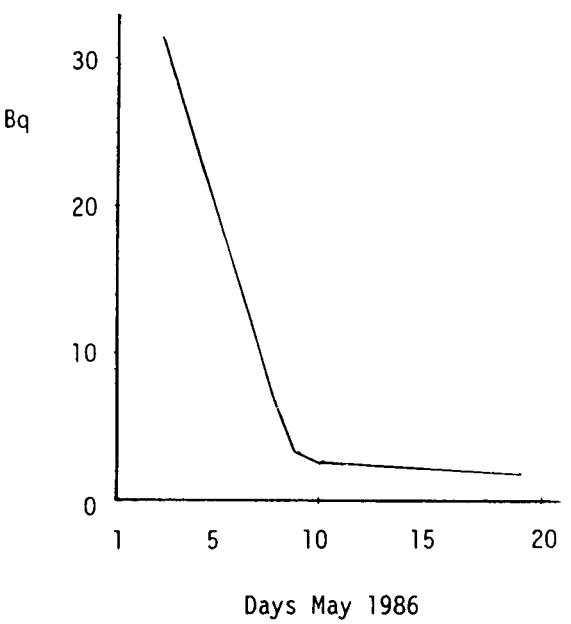

Fig.2 I-131 daily urinary excretion for people living in Rome

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MEASUREMENT OF INTERNAL CONTAMINATION WITH RADIOACTIVE CESIUM RELEASED FROM THE CHERNOBYL ACCIDENT AND ENHANCED ELIMINATION BY PRUSSIAN BLUE

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A serious accident was occurred in No.4 reactor of the Chernobyl Nuclear Power Station on April 26, as a result of which considerable quantities of radioactive materials were released to the environment. Fifteen members of a Chinese Foreign Trade Exhibition Group were working at Sofia and Profdef, Bulgaria from April 19 to May 23, 1986. During May 5-10 they occasionally found a rise in γ-background dose rate with a FD-301 dosimeter, which indicated that it might be contaminated with radioactive material released from the Chernobyl-4 reactor. Fifteen internally contaminated Chinese persons were monitored from June 12-21 in Beijing. This paper describes the results of measurements and of the effect of Prussian Blue on the rate of elimination of radiocesium.

The γ-activities of radionuclides in the 15 subjects were measured with the low background whole body counter developed in our laboratory [1] . The measurements were performed in two ways: a single crystal measurement at the sits of the thyroid, and a single, constant scanning speed detector moving along the length of the body. The counts were obtained by subtracting the counts in noncontaminated cases from the countes obtained from contaminated ones. The activities of radionuclides in the body was calculated by a matrix inversion method using calibration factors.

The results show that the activity of ^{131}I in the thyroid of these persons was lower than the minimun detectable level (10 Bq).

The activities of radiocesium at the moment of measurement are shown in Table 1. It express that radiocesium can be measured in vivo by a whole body counter for a relatively long time after contamination and may serve as a late marker of contamination of fission products released from a nuclear reactor accident. The initial intake was estimated on the supposition of: (1) A single intake at the middle of the exposure period i.e. May 11 (exposure period May 1-23), (2) The retention equation for cesium recommended by ICRP publication No.30 [2] was used in the estimation. Table 1 shows that the range of activity measured was 68-840 Bq (^{137}Cs) and 110-630 Bq (^{134}Cs). The range of the estimated initial intake was 95-1200 Bq (^{137}Cs) and 170-900 Bq (^{134}Cs).

The initial intake of ^{134}Cs and ^{137}Cs estimated here may be of significance in evaluating the levels of internal con-

tamination level of people who lived in Sofia and Profdef, Bulgaria, 6-30 days after the Chernobyl accident.

Table 1
Internal Contamination of 15 Chinese Subjects, Resident in Bulgaria at the Time of the Chernobyl Nuclear Reactor Accident

NAME	Date of Measurement	Activity in Body/Bq*		Intake Estimated/Bq	
		^{137}Cs	^{134}Cs	^{137}CS	^{134}Cs
ZN	86.6.12.	700	360	950	500
FU	6.19.	470	370	660	550
GO	6.17.	600	340	850	500
MU	6.18.	470	380	660	560
CE	6.20.	440	300	640	450
HO	6.21.	430	330	620	490
CN	6.19.	340	300	490	440
YU	6.18.	390	300	550	430
YN	6.20.	840	630	1200	930
HU	6.18.	280	170	390	260
EI	6.19.	350	240	500	350
SO	6.19.	440	250	620	270
WE	6.19.	320	210	460	300
JA	6.18.	370	260	530	380
LI	6.17.	68	110	95	170

* The relative error of all the measured values, except LI are in the range of 5-12%, while for LI the figures are 30% for ^{137}Cs and 31% for ^{134}Cs.

Activity of radiocesium was regularly measured in the 3 volunteers during a period of more than two months. The activities of ^{137}Cs and ^{134}Cs decreased with time after contamination with a single exponential function. The biological half-time of the radiocesium ranged from 42-71 days for ^{137}Cs and 42-51 days for ^{134}Cs.

In this study the biological half-times of radiocesium are shorter than the 110 days for the 0.9 of retention recommended by ICRP. As we know that age, sex, hormone and potassium content in food etc. may influence the turnover of radiocesium in body. So in the case of dose estimation, it is preferable to use the subject's own biological half-time instead of the recommended biological half-time of 2 days for 0.1 of retention, 110 days for 0.9 of retention.

The annual effective dose equivalent was estimated on the basis of the highest initial intake (YN), which was $1.1X10^{-2}$ mSv for ^{137}Cs, $1.2X10^{-2}$ mSv for ^{134}Cs. The above values were only around 2% of the ICRP annual dose equivalent limit for the public (1 mSv). For the other 14 cases the annual effective dose equivalent should be much lower.

Prussian blue(PB) was given to the 3 volunteers to investigate the effect on cesium excretion. 1.0g of Prussian Blue

was given 3 times per day, for a six days course. Three courses were given with a time interval of 6 days between each subsequent course. During the investigation period of 114-141 days after contamination, the biological half-time of radiocesium ranged from, 29 to 48 days for ^{137}Cs and 23 to 33 days for ^{134}Cs. From the biological half-time it can be seen that the body retention of radiocesiums declined more rapidly following Prussian Blue administration than in those of controls.

It is evident that shortening of biological half-time results in a more pronounced decrease of cumulative activity of radiocesium in the body. The percentage excess decrease of cumulative activity during the period of 114-141 days after internal contamination was calculated by using each subjects own biological half-time. The results in Table 2 indicate that the cumulative

Table 2
Diminution of Radiocesium Body-Burden in Man by Prussian Blue (PB)

Persons	Radiocesium	Cumulative Activity/Bq.d		Activity
		No PB	PB Admin.	Decreased(%)
YN	^{134}Cs	4400	3600	18
	^{137}Cs	9900	9100	8
ZN	^{134}Cs	2100	2000	5
	^{137}Cs	5700	5600	2
MU	^{137}Cs	4200	3500	17

activities decrease significantly ($P<0.05$), showing that Prussian Blue treatment can enhance the excretion of radiocesium.

Many studies have demonstrated that ferric ferrocyanide is an effective agent for accelerating the turnover of internally deposited radiocesium in animals and humans. Ferric ferrocyanide is not absorbed into the body, it preferentially binds cesium ions in the lumen of the gut, thereby interrupting the enteric circulation and preventing reabsorption and deposition in tissues. The shortening of biological half-time has been used as an indicator of effectiveness. Some works appears that the effectiveness of Prussian Blue is not strongly dependent on the time when treatment begins, but is proportional to the duration treatment[3-6].

Whole body ^{40}K measurements were also made on the same 3 volunteers. The results indicate that the difference of ^{40}K content between the control and PB administration period is not statistically significant. Body potassium and sodium burdens do not appear to be affected by treatment[5], so it is non-toxic and well tolerated.

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RADIOACTIVITY REMOVAL AND TRANSPORT PROCESSES FROM SOIL TO ADRIATIC SEA WATER DERIVING FROM CHERNOBYL FALL-OUT DEPOSITION

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INTRODUCTION

The radionuclide contamination of the Po river Valley, the largest italian area interested by the fall-out deposition due to the Chernobyl nuclear accident, may be defined taking into account the extent of radionuclide depositions, the soil sorption characteristics and the successive transfer processes from soil to the Po river system.

On the basis of three months investigations to evaluate the fall-out deposition (1-2), it was possible to work out the analysis of the radionuclide transfer from soil to surface waters and to verify the laws governing such a process.

Radionuclide transport is characterized by two main steps: a - the "direct runoff" of fresh deposited material from soil surface to receiving water body, promoted by rain water and b - the "delayed runoff" of accumulated radioactive material by desorption from soil and mechanical removal by rain water. In addition, being the Po river the final receiving water body for wet depositions, it was possible to obtain an evaluation of the purification power of the Po basin by considering the change with time of the concentration of some selected radionuclides (^{131}I, ^{103}Ru, ^{137}Cs). Of course, this evaluation holds only for the upstream part of the catch basin with respect to the investigated section. The latter was located at Castel San Giovanni, in the middle course of the Po river, with a catch basin of 42,000 Km^2 and an average river flow rate of 1,000 $m^3 s^{-1}$.

RESULTS AND DISCUSSION

According to Jacobi (3) the activity concentration, c_w, of a given radionuclide in river water, is expected to be the sum of the mean fall-out rate, a_F, and the mean accumulated soil activity, a_s:

$$c_w = k_F \cdot a_F + K_S \cdot a_s \qquad (1)$$

The correlation coefficients K_F and K_S are given by the following relationships:

$$K_F = F \cdot f_F / V_W \; ; \; K_s = F \cdot \lambda_s / V_w \qquad (2)$$

where F is the size of the drainage area (km^2), V_w the river water flow rate at the considered reference section (km^3/week), f_F^w is the relative fraction of fallout activity transferred to the river by direct runoff, λ_s is the rate constant for the elimination of activity from the soil to river water by delayed runoff. The maximum cumulated activities (KBq per m^2) were 10 for ^{137}Cs, 58 for ^{131}I and 9 for ^{103}Ru.

The correlation coefficients for the selected radionuclides were determined on the basis of eq. (1) inserting 16 measured weekly mean values of

a_F, a_S (corrected for physical decay) and c_W (Fig. 1 and 2). Radioactivity measurements were carried out in the time period April 28-August 12, 1986. Results are reported in Table 1. The quite high values of the linear correlation coefficients reported in the last row of Table 1 indicate that the adopted model exhibits an adequate reliability to describe the complex mechanisms of radionuclide transfer and release. The correlation coefficients for direct runoff, K_F, for the three radionuclides are similar for ^{137}Cs and ^{131}I whereas a greater mobility is assigned to ^{103}Ru. The correlation coefficients for the radionuclide delayed release from soil, which are representative of the strenght of their association to soil, indicate a stronger uptake for ^{137}Cs, intermediate for ^{103}Ru and lower for ^{131}I.

Once known the K_F and K_S values and the drainage surface it is possible to obtain the values of f_F and λ_s, as reported in Table 2. The f_F values indicate that only a small fraction (0.1 - 1%) of the deposited activity is removed from soil and directly transferred to the river. A different behaviour of the three radionuclides is evidentiated if the λ_S values are considered. Table 2 reports λ_S values evaluated on a weekly base and an a yearly base. The removal of ^{131}I and ^{103}Ru is much faster than the one for ^{137}Cs; however, taking into account the half-lives of the investigated radionuclides, only ^{137}Cs is expected to be found in soil for a long period. The delayed runoff of ^{137}Cs, with a rate constant of 0.3% per year, allows to work out an "effective" half-life for this radionuclide of 26 years. Of course ^{137}Cs removal from soil may be ascribed only to leaching processes. As a matter of fact the self-purification power of the river, which takes into account radionuclide migration into soil and bioaccumulation by vegetables, will act to keep lower the "effective" ^{137}Cs half-life. Evidence of the self-purification power of the Po river basin can be observed in Fig.2, which shows the variation with time of the radionuclide activity in water.

In addition the radionuclide interaction between aqueous and suspended matter phases was investigated. According to Onishi (4) the affinity of a species for suspended matter can be expressed by the K_d value, which in turn can be also interpreted as a delay term for a given radionuclide with respect to its transportation along the water body to the mouth . The Kd values for the investigated radionuclides are reported in Fig. 3. The Kd value sequence is $^{137}Cs > {}^{103}Ru > {}^{131}I$, thus indicating that ^{131}I transport is mainly correlated to aqueous phase mobility, whereas ^{137}Cs transport is delayed by strong interaction with the suspended matter.

In order to verify the transport characteristics, radioactivity measurements were carried out in bottom sediments collected at the mouth of the Po river in the Adriatic sea. ^{131}I and ^{103}Ru activities, normalized with respect to ^{137}Cs activity, are reported in Table 3. Normalized values are reported for total fall-out collected at 7 stations of the Po basin, and for bottom sediments. The latter activities corrected for the Kd ratios, are reported in the last column of Table 3.

The analysis of the ^{103}Ru normalized values indicates that, in the time period April 30 - June 12, 1986, the values are higher in sediments than in total fallout thus suggesting a greater mobility for ^{103}Ru. Later, upon the arrival of ^{137}Cs in the mouth area, normalized activities become similar.

In the case of ^{131}I, activity values are higher in depositions than in sediments, but when corrected for the ratio of Kd values, the trend is reversed, thus confirming a much higher ^{131}I mobility in the water body.

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Table 1 - Direct and delayed runoff coefficients for ^{103}Ru, ^{131}I and ^{137}Cs

	^{137}Cs	^{131}I	^{103}Ru
K_S (m^{-1})	3.9×10^{-3}	1.5×10^{-1}	2.7×10^{-2}
K_F (week/m)	2.0×10^{-1}	1.0×10^{-1}	7.9×10^{-1}
R (linear correlation coefficient)	0.95	0.90	0.94

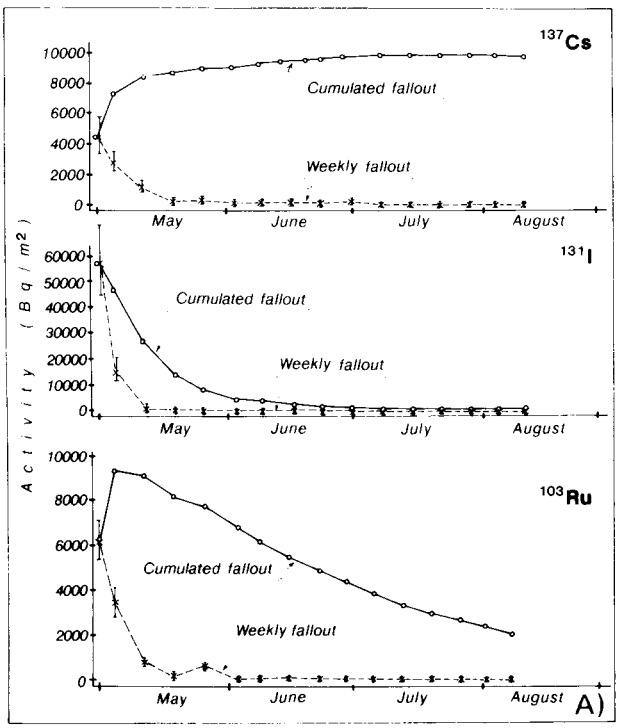
Table 2 - Evaluated values for f_F and λ_S

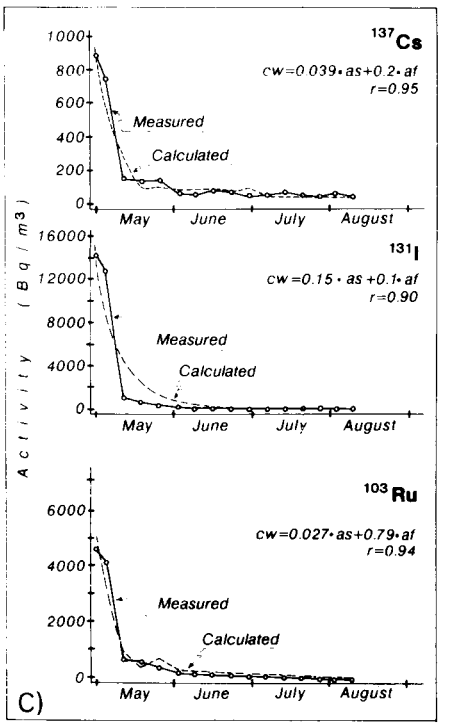
	^{137}Cs	^{131}I	^{103}Ru
f_F (%)	0.29	0.14	1.15
λ_{S1} (week base)	5.6×10^{-5}	2.1×10^{-3}	3.8×10^{-4}
λ_{S2} (year base)	2.9×10^{-3}	1.1×10^{-1}	2.0×10^{-2}

Table 3 - Normalized ^{103}Ru and ^{131}I activities in total fall-out and in Po river mouth sediments

Sampling period	^{103}Ru			^{137}Cs		
	fall-out	sediments	corrected sediments*	fallout	sediments	corrected sediments*
4-30/5-6	1.8	3.1	16	4.7	1.0	51
5-7 /5-29	1.6	2.7	13	0.8	0.2	10
6-5 /6-12	0.6	1.2	6	--	--	--
6-19/8-14	0.8	0.3	1.5	--	--	--

* Normalized sediment activity multiplied by the ratio of the Kd values for interested radionuclide and ^{137}Cs.





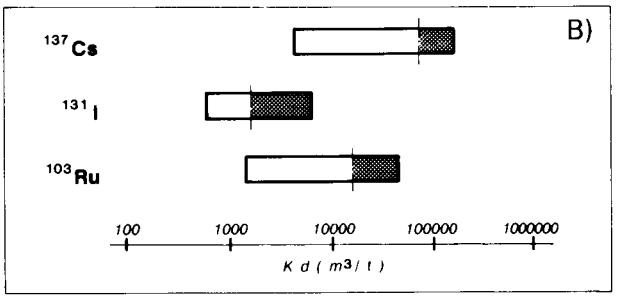


Fig.1:
A) Mean fallout-rate and cumulate fallout activity in the Po river Valley after Chernobyl
B) Minimum,mean and maximum distribution coefficient values (Kd) for the Po river
C) Comparison between measured and calculated activities in the Po river

THIRTY YEARS OF EXPERIENCE IN HEALTH PHYSICS EDUCATION AT PURDUE UNIVERSITY AND PLANS FOR THE FUTURE

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INTRODUCTION

The formal education and training of health physicists at Purdue University began thirty years ago with the introduction of a Ph.D. program followed later by M.S. and B.S. degree offerings. During that period 82 Ph.D., 52 M.S. and 131 B.S. students have received degrees in health physics and embarked on careers in radiation protection. Included in these numbers are graduates who have entered the medical physics field, but not those who entered nuclear medicine, although some of the latter have found themselves performing many health physics duties.

EMPLOYMENT TRENDS

One way of anticipating areas of emphasis which health physics education should stress in the future is to study the employment trends of graduates. This study has been carried out by evaluating employment trends in six categories: medical, university (academic and radiation safety), federal laboratories, regulatory agencies, nuclear power and waste management/remedial action.

A break-out of employment areas for Ph.D. recipients, as shown in Table 1, indicates that in the early years of the program almost all took academic positions. In the early 1970s there was a sudden opening of Ph.D. level positions with state and federal regulatory agencies such as the Nuclear Regulatory Commission, and at Department of Energy laboratories. During the mid-1970s many Ph.D. recipients were hired by the nuclear power industry, but this opportunity for employment has diminished. Also, when the federal program of Radiological Health Fellowships was discontinued in the mid-1970s, a considerable reduction occurred in the number of health physics graduate students that could be supported at Purdue, as illustrated by the last three columns in Table 1.

During the late 1960s and early 1970s most graduate students elected to continue on for a Ph.D. after their M.S. But in the mid-1970s, as illustrated in Table 2, many elected to take employment after completing their M.S. because of improved job opportunities. This option was facilitated during this period by

a decision to offer a non-thesis M.S. degree. By taking additional coursework a student could receive an M.S. without doing a research project, which was often not a requirement for the types of M.S. level positions that were available, such as with the nuclear power industry or regulatory agencies. This trend has now reversed in the mid-1980s and students are now electing to continue on toward a Ph.D. after finishing their M.S. degree. The large number of M.S. graduates hired by the nuclear power industry in the early 1980s was a direct result of the impact that the Three-Mile Island accident had upon the demand for advanced degree health physicists for ALARA engineering and emergency planning programs.

TABLE 1

Types of Employment Accepted by Ph.D. Graduates

	1958-60	61-63	64-66	67-69	70-72	73-75	76-78	79-81	82-84	85-87
Nuclear Power					3	4	2		1	
Regulatory				1	4	5	2	1	2	2
Federal Laboratory					4	4	5	2		
Academic	3	2	2	6	2	2	4	1	2	
Medical			1	4	3	3	1	1	1	2

Throughout its history the Purdue health physics program has encouraged graduate study by students from other countries. Approximately half of these students have taken permanent employment in the U.S. after completion of their graduate degrees. The remainder have returned to their native lands and are practicing health physicists in Canada (1 Ph.D.), Taiwan (3 M.S.), Iran (1 Ph.D.), Brazil (1 Ph.D.), Korea (1 M.S.), Philippines (1 Ph.D.), and Puerto Rico (2 Ph.D.).

TABLE 2

Types of Employment Accepted by M.S. Graduates

	1967-69	70-72	73-75	76-78	79-81	82-84	85-87
Nuclear Power			1	1	6	8	
Regulatory				4	1		
Federal Laboratory			2	6	3	1	
University Radiation Safety			1	3	3		
Medical	1	1	1	6	2	1	

In the early 1970s, the nuclear power industry became more active in seeking B.S. degree graduates in health physics. Consequently, an undergraduate program was developed at Purdue with a considerable amount of input from neighboring nuclear power utilities on the design of the curriculum. Courses such as reactor health physics and nuclear engineering were required in addition to environmental health physics, radiation biology and radiation dosimetry. Table 3 illustrates how rapidly this program grew at Purdue. The enrollment in the program peaked in 1983, but remains strong. Again, there was a significant increase in hiring by the nuclear industry for B.S. graduates shortly after the Three-Mile Island accident. The decreasing enrollment in the B.S. program may be related to the fact that most U.S. nuclear plants are now completed and operating. However, the demand for qualified B.S. students in the nuclear power industry continues to exist. Table 3 also shows, however, that there is a relatively even distribution of B.S. recipients in the other five areas of employment. Of particular importance is the development of a job market for B.S. recipients in the waste management/remedial action area.

TABLE 3

Types of Employment Accepted by B.S. Graduates

	1973-75	76-78	79-81	82-84	85-87
Nuclear Power	2	4	14	31	16
Regulatory	1	2	2	3	2
Federal Laboratory			2	5	4
University Radiation Safety	1	4	1	2	1
Medical		2	3	6	8
Waste Management/Remedial Action			1	5	3

The School of Health Sciences also has a strong undergraduate program in industrial hygiene. In several instances students have completed a dual major in health physics and industrial hygiene by taking the required courses in both areas. This type of degree has provided the student with the ability to perform health physics and industrial hygiene tasks at the same location.

Strong efforts have been made in the B.S. program to provide students with applied health physics experience through a summer internship program which takes place between the junior and senior year. Since its inception, 91 students have participated at sites such as Fermilab, Mayo Clinic, Brookhaven National Laboratory, Department of Energy remedial action sites and several nuclear power plants. This hands-on experience has been invaluable in helping students to determine which area of health physics they wish to pursue as a career.

CONSIDERATIONS FOR THE FUTURE

The demand for well-educated health physicists is expected to remain high well into the next century, particularly in the nuclear power industry. When this fact is coupled with the fact that many of the university health physics programs which existed in the 1970s are no longer active, there exists a potential for a serious shortage of health physicists in the U.S. in the near future. To complicate matters, with the declining enrollment in Ph.D. programs in health physics, there will be a dwindling pool of qualified new professors to perform teaching and research roles at universities.

To help meet this challenge, several developments in the Purdue health physics program will be made.

(a) Efforts will be made to strengthen course content in (1) radon hazard evaluation, (2) reactor decommissioning, (3) waste management/remedial action, (4) and regulatory policies. The potential revision of the radiation protection parts of the Code of Federal Regulations will require a considerable amount of modifications in course content, if not necessitate a separate course entirely devoted to regulatory affairs.

(b) Nationwide, only a few graduate students are being supported by governmental health physics fellowships, and little, if any, additional help is on the horizon. Consequently, efforts must be made in the academic sector and, hopefully by the Health Physics Society, to establish industry supported fellowships and scholarships. In like manner, federal funding for basic research in the health physics area is difficult to obtain. Only by increased efforts to obtain fellowships and funding for research will it be possible to carry out thesis research in vital areas such as radon emanation and measurement, and low-level and high-level waste management technology.

(c) The increasing interaction between the U.S. and other countries in radiation related matters, particularly in the area of low-level and high-level waste management, suggests the need for an increased amount of faculty and possibly graduate student exchange programs. This could take the form of sabbaticals for professors and practicums for graduate students, via bilateral agreements.

(d) Declining enrollments, particularly in the B.S. program, indicate the need for vigorous recruiting efforts to be instituted. Plans are currently in progress to distribute recruiting brochures to state high schools and, for graduate programs, to distribute recruiting posters to college chemistry and physics departments nationwide. It is hoped that recruiting efforts by existing university programs will halt the decreasing enrollment trend and provide an adequate pool of health physicists entering the job market.

NUCLEAR POWER PLANT CONSEQUENCES : PERSONNEL EDUCATION

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ABSTRACT

The 24,000 people working, to operate the french power plants were not well prepared to perform their own analysis allowing them to give a personal opinion and judgment regarding the information released by the medias shortly after the Chernobyl accident.

Consequently, EDF had launched a wide educational program for its personnel. Because the tremendous volume of work to inform this huge population, it was necessary to educate first one hundred instructors (about 2 people by plant), in charge to teach the executive staff of all power plants and technical departments; after what each executive staff having to teach its own troops.

The paper presents this approach, the pedagogic methods as well as the training means used to prevent any deviation of the program content.

The main difficulties and inapropriate solutions for implementing this program are also presented in this paper.

The conclusion points out the possible improvements of such approach and the prospective enlargement of information distribution to the entire population through the natural and trusting vectors such as professors physicians, chemists priests and pastors.

EDUCATION AND TRAINING FOR EMERGENCY PROCEDURES

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Mandatory emergency procedures are in force at all nuclear power stations and related sites using fissile material. Large numbers of people and enormous resources are committed to this purpose and so education and training is essential for the system to work at all. In this paper a proposal is made for an education and training scheme designed to maximise the efficiency and reliability of the system. Examples of courses quoted in this paper are drawn from the Department of Nuclear Science and Technology which is an autonomous unit in the Royal Naval College, Greenwich. The full range of training provided by the Department has been described by Lakey etal (1) (2) (3). In this paper courses are classified (using a letter code) in accordance with their objectives and recommendations made for the application of the courses to the people involved in emergency procedures.

COURSES

COMMAND AND CONTROL (C)

The most important courses for nuclear emergency personnel are for those in command and control. A common need in these courses is a foundation syllabus giving an understanding of the hazards involved which is essential if they are to control the process by evaluation or checking of information received. Command and Control training given at RNC mainly consists of an intensive one week Nuclear Accident Procedures Course (NAPC) which acquaints officers with the overall strategy for dealing with accidents and with their individual roles within the organisation.

PLANNING COURSES (P)

The School of Public Health at Harvard University provides a course entitled Planning for Nuclear Emergencies (PNE) (4) which is specifically designed for Planning Officers and local authority personnel, including Police, Fire and Health Services. In addition to a Foundation syllabus the course reviews USA mandatory requirements and specifically the design and conduct of exercises. A Table Top exercise (5) is used to consolidate this material.

STRATEGIC COURSES (S) AND MEDIA TRAINING (M)

Courses, of shorter duration than one week, are needed for headquarters personnel whose role is strategic and political - in this case the technical content is limited but acquaintance with current legislation and policy is essential. The very senior

staff at headquarters must be given media training and some practice in radio and TV interview procedures. Such courses are appropriate for all staff likely to be interviewed.

PROFESSIONAL ACADEMIC COURSES (A)

The first line of defence against nuclear accidents and the initial action rests with the operator. In the nuclear submarine this is the Marine Engineer Officer who must qualify in a series of professional courses which include the 6 month Nuclear Reactor Course which attracts a post graduate diploma of the Council for National Academic Awards. Training for staff appointed to nuclear dockyards is given on the Nuclear Dockyard Course (NDOC), a nine-week course. The PWR and its support systems are covered in sufficient depth to give an overall appreciation of the maintenance required. Officers appointed to Health Physics posts will go out to a period of on-job training at their parent establishment and on attachment at other establishments. The trainees return to DNST for the Nuclear Radiological Protection Course (NRPC). This is a three-month professional course which includes a relevant project, written report and oral presentation. Successful completion of the NRPC and the additional project qualifies the trainee for the award of a University of Surrey post graduate diploma in Radiological Protection.

INTRODUCTORY COURSES (I) AND GENERAL COURSES (G)

All new personnel must receive some introduction to emergency procedures and where possible they should become acquainted with the concepts of nuclear technology. For posts in the nuclear propulsion programme which do not directly involve work on the nuclear plant but which involves working on or near nuclear submarines, an introductory courses, giving an overview of nuclear submarine propulsion and associated hazards, is considered to be essential. To meet this need the DNST runs a two-week course called the Nuclear Introductory Course (NIC). This course covers in qualitative terms the essentials of nuclear and reactor physics, plant descriptions and operating procedures, health physics and radiological protection. Specialised skills are required for search and rescue, monitoring, fire fighting etc and General Courses giving training in the emergency plan and acquaintenance with the facilities on site are essential.

DRILLS

The skills required in emergency response are acquired by drills which extend from simple evacuation drills (ED), through individual drills (ID) or standard drills, on to live drills (LD) involving real equipment and possibly live radioactive exercises, which are replaced where feasible with simulation drills (SD) which avoid the risks of the live drill. In many cases staff must work as a team and this is consolidated through team drills (TD). For example, fire crews (6) must develop skills in operation of their equipment.

EXERCISES

Full exercises (FE) are carried out at least once per year on most nuclear sites. This involves all members of the site emergency team and off site organisations. Limited Exercise (LE) are useful to test part of an emergency procedure and to give those with relevant skills more frequent practice. Table Top exercises (TT) play an important role since events from a minor incident to a full scale emergency can be simulated (5). Paper exercises can be planned and procedures executed using a minimum of time with a few personnel and causing no disruption to plant staff.

PROPOSED EDUCATION AND TRAINING SCHEME

The above concepts are combined into a comprehensive education and training proposal summarised in Table 1, the symbols used to describe courses, drills and exercises are drawn from this text but the tasks and locations are linked to USA practice on power reactor sites (7). Evacuation drills (ED) or Full Exercises (FE) are omitted since these will affect all personnel. For any one reactor site the cost of the emergency training programme is formidable and the total effort can easily reach several man-years per year. But realistic and effective training is the key to successful performance of the emergency response organisation without which the reactor operation would be unacceptable to the authorities.

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TABLE 1 PROPOSED EDUCATION AND TRAINING SCHEMES

Job Title/ Location (1)	Duties	Courses (2)	Drills (3)	Exercises (3)
Shift Supervisor and crew (CR)	Recognise, Analyse and Report	A(NRC)	LD(i) SD(a) TD(m)	LE(m) TT(a)
Manager (TSC)	Command and Control	A(NRC) C(NAPC)	TD(m)	LE(i) TT(m)
Health Physics (TSC)	Advisory	A(NRPC) C(NAPC) M	TD(a) LD(a)	LE(i) TT(m)
Health Physics (OSC)	Survey	A(NRPC)	LD(i) TD(m)	LE(m)
Emergency Planner (EOF)	Advisory and Co-odination	P(PNE) M	TD(a)	TT(m)
Fire (OSC)	Damage Control, Search and Rescue	P(PNE) or G	LD(i) ID(w) TD(w)	TT(a)
Public Relations	Media Releases, Rumour Control	C(NAPC) M	TD(a)	TT(a)
General (OSC)	Specific Tasks	I(NIC) or G	ID(i) TD(a)	
Local Authority	Police and Medical	P(PNE)	TD(a)	
Headquarters	Strategic and Political	S M	-	TT(i)

(1) Terminology is similar to that in US power reactor sites; CR = Control Room, TSC = Technical Support Centre, EOF = Emergency Operations Facility, OSC = Operational Support Centre, JPIC = Joint Public Information Centre.

(2) Specific course titles in brackets refer to RNC or Harvard.

(3) Frequency indicated by, (i) initial, on joining, (w) weekly, (m) monthly, (a) annually.

THE CHANGING EMPHASES IN HEALTH PHYSICS

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This paper explores the changing emphases in health physics as evidenced by the subject matter of published papers in four primary English language journals of interest to health physicists. Articles from each journal were first grouped by subject and date of publication and were then compiled according to the list of professional domains practiced by health physicists. In 1983, the ABHP established these five Domains of Practice:

- Measurements (includes dosimetry and environmental monitoring)
- Regulations and Standards (includes ALARA and emergency response)
- Facilities and Equipment (includes shielding, ventilation, and instrumentation)
- Operations and Procedures (includes radiation control for current and proposed operations)
- Education and Training (includes workers and public)

Each of these domains includes a number of subcategories, some of which are shown parenthetically above. These domains are representative of the major areas in which a professional health physicist in the United States spends his working time. The ABHP uses these domains to administer a rigorous written examination designed to test the knowledge of prospective diplomates.

The Health Physics journal provided the largest data base for this study, encompassing three decades (1958 to present). The other journals (with the year of introduction shown in parentheses following the title) included the Journal of the Society for Radiological Protection (1981), Radiation Protection Dosimetry (1981), and Radiation Protection Management (1983). These latter three journals not only are indicative of the growth of the profession but also of changing or intensifying emphases.

We tabulated nine sets of articles from Health Physics, spanning all 30 years of publication. The first set consisted of volumes 1 to 5 (1958-1961), representing half of the volumes included in the first combined subject index (Volumes 1 to 10); the second set, volumes 11 to 13 (1965-1967), comprised those volumes included in the second Health Physics subject index. Each of the other seven sets consisted of two volumes, representing at least one full year of issues. The volumes reviewed (see Table 1 for specific volumes included) were selected at approximately 3-year intervals. Using the guidance provided by the first two Health Physics subject indexes

(volumes 1-10 and 11-13) and our own experience, we identified 18 broad subject categories: accelerators, accidents, airborne radioactivity, bioassay, dosimetry, education and training, environmental monitoring, fallout, instrumentation, medical, plutonium, power reactors, radium, radon and daughters, regulations and standards, uranium, waste disposal, and weapons tests. We then placed each article into one of the 18 categories and tallied the number of articles per category and volume set. An estimated 90% of all the articles published fell into one or more of these 18 categories. There was the possibility for overlap, which may have resulted in some articles being classified in more than one category. Similarly, some articles (<10%) were not included in the tally (for example, an article on manpower) as they simply did not fit into any of the categories selected. Because the absolute number of articles in each category and volume combination was small, we chose to group the data resulting from the 18 categories into the five domains of practice established by the ABHP.

Table 1 shows the relative percentages of the Health Physics articles categorized by domain. Although the results of the tabulation may be more representative of the Health Physics journal than health physics as practiced, the data suggest a decreasing emphasis in the Measurements domain and an increasing emphasis in the Education and Training domain. Over the years, dosimetry (a subcategory of the Measurements domain) has been the dominant area of interest, though the relative emphasis in dosimetry from the late 1970s has declined. However, some of the apparent decline is not real, for the journal Radiation Protection Dosimetry, almost exclusively devoted to this specific area, was initiated in 1981. Thus papers that would ordinarily have been published in Health Physics were diverted to this new journal.

TABLE 1. Percentage of Health Physics Journal Articles by ABHP Domains per Volume(s) and Year(s) of Issue

	Volume Number								
Domain	1-5 (pre-1962)	11-13 (1965-1967)	19-20 (1970-1971)	25-26 (1973-1974)	30-31 (1976)	36-37 (1979)	41-42 (1981-1982)	46-47 (1984)	51-52 (1986-1987)
Measurements	68	57	59	74	68	46	36	36	37
Regulations & Standards	5	4	2	4	3	3	3	5	6
Facilities & Equipment	21	29	29	13	17	35	41	41	31
Operations & Procedures	3	6	5	6	7	8	11	10	14
Education & Training	3	3	4	4	5	9	10	8	12

For the other three journals, all of which have been published for fewer than seven years, we used the sorting process described for the Health Physics journal, but included all articles published to date. The current emphases in the health physics profession are dramatically expressed in terms of these three new refereed English language journals. As shown in Table 2, the Measurements domain is clearly the most emphasized, followed by the Facilities and Equipment domain. Emphasis on the Measurements domain is even more pronounced when articles from Radiation Protection Dosimetry, which are more or less exclusively published in the Measurements domain, are included in the tally. The domains of Regulations and Standards and of Education and Training are clearly given more emphasis in the newer journals than in the flagship journal, Health Physics.

We grouped the Health Physics journal data for the last three sets (i.e., Volumes 41-41, 46-47, and 51-52) to cover a period of time comparable to the assembled data for the other three journals (Table 2). Because we had also recorded the data for books reviewed in the Health Physics journal by category (and then by domain), we have included that information in brackets beside the Health Physics article summary in Table 2. These data suggest that the books reviewed were more equally distributed within the five domains than were the journal articles. For comparison, the percentage weightings for the ABHP examinations are also shown in Table 2.

TABLE 2. Percentage of Published Articles in Health Physics Journals During the 1980s Compared to the ABHP Exam Weightings

Domain	Health Physics	RPM*	Journal of the SRP**	Weighted Average All 4 Journals+	ABHP Exam Weighting
Measurements	36 [31]++	42	36	44	30
Regulations & Standards	5 [12]	15	23	8	16
Facilities & Equipment	39 [15]	24	17	28	24
Operations & Procedures	12 [16]	4	11	10	18
Education & Training	9 [26]	16	13	9	12

* Radiation Protection Management (U.S.A.)
** Journal of the Society for Radiological Protection (U.K.)
\+ Includes Radiation Protection Dosimetry plus other journals shown
++ Bracketed numbers refer to domains assigned to Book Reviews only

Other less obvious changes in emphasis were also observed through the review of articles published over the years. Such changes would not necessarily be reflected in the domains, which are rather broad in their scope and hence would tend to mask even large changes in relatively specific areas. These changes included an increased emphasis on data processing, as evidenced by the addition of a software section in Health Physics during 1985; the change from ICRP 2 to ICRP 30 methodology; an increased emphasis on ALARA; and significant changes in the areas of environmental monitoring. For example, the emphasis on radon and daughters (from less than 1 or 2% of the articles in Health Physics in the 1960s to greater than 10% in the 1980s) has shown a dramatic increase compared to the subject area of environmental monitoring, which peaked at 25% in the mid-1970s and now hovers in the 10 to 15% range as it had done in the earlier years. Articles relating to ALARA were nonexistent 20-30 years ago, but today are reflected in the increased emphasis on the Regulations and Standards domain.

Perhaps somewhat surprising is the relatively low level of emphasis consistently given in the Health Physics journal to certain subject areas. In particular, this applies to power reactors and waste disposal, in which there has been an obvious and increasing interest in recent years by regulatory bodies and the media, as well as the general public. However, this low emphasis on power reactor health physics may be misleading, for this subject area is covered in numerous, other more recent publications, especially in the new refereed journal, Radiation Protection Management. Similarly, the Journal of the Society for Radiological Protection has provided additional emphasis on not only the area of power reactor health physics, but medical radiation, regulations and standards, and accidents. Clearly, the specific emphases given in these latter two journals differ from that of the older journal, Health Physics.

To some extent, the changes in emphasis on various domains may reflect the differences between the world of the operational or applied health physicist and his research-oriented counterpart. Currently, about three-fourths of the articles in the peer-reviewed health physics literature fall into one of two domains: Measurements (including dosimetry and environmental monitoring) and Facilities and Equipment (including shielding, ventilation, and instrumentation). The balance of the articles were approximately equally split among the remaining three domains. The distribution of articles within the domains is similar to the weighting given these same domains by the ABHP, which might be interpreted as an indication of the relevance of the examination process for ABHP diplomates.

This work was supported by the U. S. Department of Energy under Contract DE-AC06-76RLO 1830.

COMMUNICATION STRENGTHS AND WEAKNESSES OF RADIATION PROTECTION PROFESSIONALS IN THE UNITED STATES AND CANADA

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INTRODUCTION

Effective health risk communication may well determine the future of peaceful applications of nuclear technology and the social acceptance of risks from radiation in medicine, research, and industry. However, radiation protection professionals who know how to quantify risks and provide appropriate safeguards, have historically encountered great difficulties in communicating their risk perspectives to the concerned public. Since the early days of the Manhattan Engineering District, health physicists have struggled with the problems of communicating the benefits and risks of radiation technology to non-technical people.

In the U.S., organizations, such as the Health Physics Society and the American Nuclear Society, have traditionally attributed communication difficulties to the public's lack of technical understanding. This has led to the belief that if the public could be provided sufficient information or education, they would "understand" radiation issues and their concerns about radiation risks would be resolved. Consequently, these national organizations and their local chapters have established public information programs and speakers bureaus. These programs primarily focus on presentation of technically accurate data and attempt to foster understanding of radiation by analogies with background radiation or other sources of risks commonly accepted by society.

This paper will show that such public information programs can at their best reach only about 25% of the general public. These programs could greatly enhance their effectiveness by learning the different ways that radiation professionals and the general public prefer to gather data and make decisions. This insight was derived from a four year study of communication styles conducted by the Baltimore-Washington Chapter of the Health Physics Society. The study began in early 1983 following a meeting convened by the Chapter President, Dr. Allen Brodsky, to address what we knew about the communication strengths and weaknesses of health physicists. Recognizing that traditional communication approaches were not achieving desired results, the Chapter's Public Information Committee decided to host a one day workshop in early June 1983 to explore other approaches. The

authors, as both radiation professionals and communication specialists, conducted the workshop. This included an analysis of communication profiles using the Myers-Briggs Type Indicator (MBTI), a tool that has become the main basis of the ongoing communication study*.

MBTI COMMUNICATION STUDY

The MBTI provides a quantitative measure of how we prefer to gather information, make decisions, and relate to other people. The development of this tool was a lifelong effort by Katherine Briggs and her daughter, Isabel Briggs Myers, to measure concepts of psychological type originally defined by Carl Jung, a Swiss psychologist in the 1920's (My80)(Ju23). The MBTI is now used worldwide for management training, organizational development, team building, conflict resolution, and communication training.

The MBTI uses 166 multiple choice questions to determine our natural preferences in four categories:

1) Where we get our energy by EXTRAVERSION (E) -- from interacting with people, activities, and things, or by INTROVERSION (I) -- from inner reflection, ideas, and private time.
2) How we collect information by SENSING (S) -- using our five senses for specific, detailed, and practical data, or by INTUITION (N) -- seeing patterns, connections, meanings, and imaginative possibilities.
3) How we make decisions by THINKING (T) -- using logical analysis, principles, and laws to determine the truth, or by FEELING (F) -- using personal values, other's concerns, and sentiments for what is good.
4) How we relate to others by JUDGING (J) -- planning, organizing, making decisions, and striving to reach closure, or by PERCEIVING (P) -- enlarging our awareness, keeping options open, resisting closure, and striving for understanding.

Jung said that everyone will have a distinct preference for one side of each category in the same way that we naturally favor our right or left hand. With the current data base of over 800 profiles, we find that the preferences of radiation protection professionals in the U. S. and Canada are as follows:

(E) EXTRAVERSION -- 37%	(I) INTROVERSION -- 63%
(S) SENSING -------- 43%	(N) INTUITION ------ 57%
(T) THINKING ------- 80%	(F) FEELING -------- 20%
(J) JUDGING -------- 66%	(P) PERCEIVING ----- 34%

The overall preference is for I N T J. This is one of 16 possible combinations of Myers-Briggs type, however, this particular combination of preferences represents only one percent of the general population. On the other hand, it is not

* This study is funded by the Communication Sciences Institute, Advanced Communication Techniques, and the Baltimore-Washington Chapter of the Health Physics Society.

coincidental that people with the I N T J preference would choose the health physics profession. People will naturally choose occupations that allow them to most freely exercise their natural preferences. I N T J's have attributes that are vitally important to radiation protection. Their strengths are that they tend to be independent decision makers, self confident, high achievers, competent, efficient, good with theoretical models, creative, logical, and analytical. Their weaknesses arise when they are seen as too independent, unemotional, demanding, reserved, and critical. They may neglect social rituals and do not like to waste time in idle dialogue and play. Dealing with opposite types is their major problem in communications.

LANGUAGE DIFFICULTIES

This MBTI study has also revealed some insights that could revolutionize the ways we communicate with the public. Namely, even when we all use the same English words, there are actually four English languages in current use today where the meaning and context of the words represent lifestyles as totally different as those of foreign countries.

The languages are based on one of our data gathering preferences, SENSING or INTUITION, or on one of our decision making preferences, THINKING or FEELING (Ye82). The first preference of the majority of radiation protection professionals (60 to 80%) is to communicate in the THINKING language. In this language they will provide logical, rational, analytical conclusions to radiation issues. Since this approach is favored primarily by technical professionals, such as scientists, engineers, and managers it will miss 75 - 90% of any general audience. Realizing this discrepancy, radiation professionals could turn to their second preferred language which is INTUITION (25 - 30%). This language relies on inspiration, imagination, ingenuity, theories, and models. Unfortunately, this language will be heard by less than 10% of any general audience, mostly those who are writers, artists, theoreticians, or entrepreneurs.

The preferred language of 40% or more of the general population is SENSING. This language is based on the five senses for communication. It depends on first hand experience and focuses on what is said and done at the present moment. SENSING types want answers to radiation issues in terms of "Is is safe or not safe, right now?" They do not want to hear about concepts of linear non-threshold dose models or future probabilities of radiation risks. SENSING is third in preference for radiation professionals and less than 10% are inclined to speak this language.

The most difficult language of all for radiation professionals is the language of FEELING. This is fourth in preference and the language they would prefer to avoid as far as possible. Less than 10% of health physicists will use this language, even though it is favored by 25% or more of the general population. This language is the opposite of THINKING as a decision making lifestyle and relies upon subjective values, sentiments, and personal experience.

WHY COMMUNICATIONS FAIL

When we realize that, in our every day communications, people are using four different languages, we can begin to appreciate why we have so many misunderstandings. To have any understanding at all, we have to continuously translate into our frame of reference. Communications break down when we cannot (or will not) do the translating. Failures in communication arise when two people speak different languages and are not able to share a common value system or derive a common meaning.

PUBLIC COMMUNICATIONS

Most public information programs are oriented towards providing simplified information to give the public a basis for understanding nuclear technology. What is usually not stated (but we know deep in our hearts), and what we really want, is for the public to understand radiation "THE WAY WE DO" and for them to think "THE WAY WE DO," in order to arrive at the same logical conclusions "AS WE DO." Therefore, we engage our best technical experts, who primarily use only the THINKING language to present both verbal and written communication to the public. But, what is the primary language of our audience? Most general audiences with concerns for radiation, especially at public hearings, will prefer to communicate in the FEELING or SENSING languages.

CONCLUSION

The key to successful public information is to recognize that there are two (or more) languages in use. Then we have the option of communicating in the preferred language of our audience. We can do the translation into the language they will hear and appreciate as their own, rather than forcing our audience to deal with our foreign language. When we attempt to communicate in their language, audiences will be more comfortable and receptive, even if we are not perfect in their language. Radiation professionals can easily double their effectiveness for addressing public concerns about nuclear technology by learning to communicate in the preferred language of the audience (Jo84).

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EXPERIENCE IN TEACHING HEALTH PHYSICS AT "POST LAUREAM" LEVEL AT THE UNIVERSITY OF PISA

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We describe the experience achieved in teaching Health Physics at "post lauream" level at the Department of Physics of the University of Pisa in the past 13 years. An updated description of the overall situation at the italian Universities is also presented.

The teaching of Health Physics at "post lauream" level in Italy was initiated in 1961 at the University of Bologna with an one-year course in "Radioprotection and Radioisotope Techniques". More than ten years later a second initiative was established at the University of Pisa with an one-year course in Health Physics ("Corso di Cultura in Fisica Sanitaria"), which was open to students with a degree in Physics, Engineering and Chemistry. The course was partially devoted to Health Physics, but other subjects were taught which were more related to Medical Physics. This was due to the fact that the discipline "Fisica Sanitaria" is not intended in Italy as an exact translation of the english term "Health Physics", but it has a more general meaning;for instance, it covers all the applications of the Physics to Medicine, which are tasks of the Departments of "Fisica Sanitaria" of the Italian Hospitals, as provided by the Reform of National Health Service Act [1969]. Thus the term "Fisica Sanitaria"should be thought of with a more general meaning of Health and Medical Physics, all throughout the paper.

The "Corso di Cultura in Fisica Sanitaria" was run at the University of Pisa for six consecutive years from 1974 to 1980. The technical and scientific support was provided by the Department of Physics and by the Institute of Nuclear Engineering of the University of Pisa, by the Institutes of Radiology and Nuclear Medicine of the University of Florence, by the Pisa branch of the National Institute of Nuclear Physics (I.N.F.N.) and by ENEA (the Italian Agency for Nuclear and Alternative Energies). A statistical analysis of the students who attended the course is presented in table 1. In summary, about 50% of the enrolled students completed the course, and about 80% of htese already held or took after the completion of the course a job in Health and Medical Physics. From what was learned from our and from other experiences (similar courses were held at the Universities of Bologna and Milano) and taking into account the new University legislation (The Reform of University Act [1982] and the reform of the post lauream

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Table 1

Statistical analysis of the students of the
" Corso di Cultura in Fisica Sanitaria "

Academic year	1974-75	75-76	76-77	77-78	78-79	79-80	Total
Maximum number of students	no limit	no limit	12	15	15	10	-
Number of students enrolled	6	24	12	7	13	4	66
Number of students who have completed the course	5	12	8	3	6	0	34
(in percentage)	83%	50%	67%	43%	46%	0%	52%
Number of students who have completed the course and have a job in Health or Medical Physics	5	9	6	3	5	-	28

Table 2

List of the subjects taught at the
"Specialty School in Fisica Sanitaria" of the
University of PISA

First year

Mandatory subjects:
- Advances in Physics
- Elementary in Biology, Anatomy and Human Physiology
- Physics and Dosimetry of Radiation (I)
- Instrumentation and Biomedical Technologies (I)

One among the following subjects:
- Biological Effects of Radiation
- Biophysics
- Electronics and Nuclear Instrumentation

Second year

Mandatory subjects:
- Protection of the Environment and of the Human Being
- Computer Science and Statistics in Medical Applications
- Physics and Dosimetry of Radiation (II)
- Instrumentation and Biomedical Technologies (II)

One among the following subjects:
- Physics Methodologies in Radiotherapy
- Physics Methodologies in Diagnosis

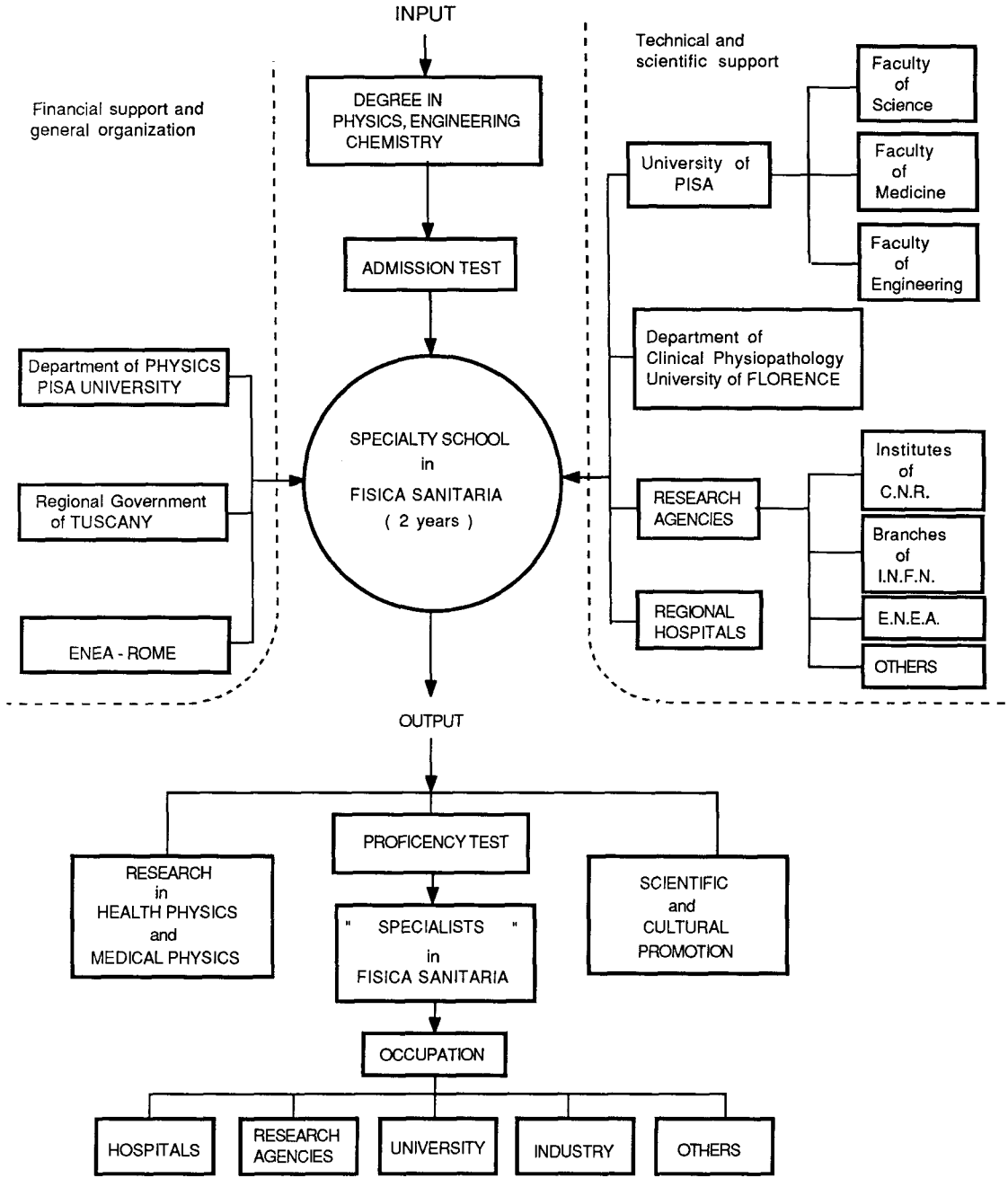

Fig. 1- General scheme of the Specialty School in "Fisica Sanitaria" of the University of Pisa

Teaching Act [1982]), there came evident that the existing one-year course should evolve into a two-year post graduate school. The Specialty School in "Fisica Sanitaria" was first established at the University of Pisa in 1983. It followed several modifications, also as a result of the general coordination with the other similar schools in Italy. Table 2 shows the subjects which will be taught as from 1988 and fig.1 shows the general scheme of the School.

There now exist five Specialty Schools in "Fisica Sanitaria": at the University of Bologna, Milano, Pisa, Roma "La Sapienza" and Roma "Tor Vergata" (see table 3): 52 students in total were enrolled in 1987 in the first year of these schools; this number almost fills the overall capability. New initiatives are under consideration among other universities, especially in the South of Italy. Over the last three years 50% of the students on average completed the two-year school and took the final exam to become "Specialists in Fisica Sanitaria". The various schools receive scientific and teaching support from many Departments of their own University (mainly related to the fields of Science, Medicine and Engineering), from Research Agencies (INFN,CNR,ENEA,..) and from Regional Hospitals (see table 4).

Table 3

Students enrolled in the first year of the existing italian Specialty Schools in "Fisica Sanitaria"

Academic year	1982-83	83-84	84-85	85-86	86-87
University					
BOLOGNA	-	-	-	3	7
MILANO	15	15	20	10	17
PISA	-	3	3	2	5
ROMA "La Sapienza"	-	-	-	7	8
ROMA "Tor Vergata"	-	9	11	14	15

Table 4

Affiliation of the members of the Faculty of the School

	BOLOGNA	MILANO	PISA	ROMA "La Sapienza"	ROMA "Tor Vergata"
University Departments					
Science	10	4	3	8	4
Medicine	5	2	2	-	5
Engineering	1	-	1	-	9
Others	2	-	1	-	-
Research agencies	16	2	4	2	17
Hospitals	7	3	-	-	3
Total	41	11	11	10	38

Acknowledgments are due to Professors G. Maltoni, E. Fiorini, B. Maraviglia and C. Franconi for providing the data relative to the Specialty Schools of Bologna, Milano and Roma.

TRAINING IN THE REMOTE-PILOTING OF VEHICULES FOR INTERVENING IN HOSTILE ENVIRONMENTS

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ABSTRACT

Benefiting from its long and rare experience accumulated over more than twenty years, the Groupe for Remote Intervention of the French Atomic Energy Commission (IPSN-DPT) now offers a highly specialized training in the remote- piloting of vehicules for intervening in environments hostile to man.

The aim of this training, which is unique in Europe, is to teach individuals how to pilot various types of remote-controlled vehicules, the purpose of which is to carry out remote controlled operations inside or outside installations.

Depending on their specific nature and equipment, these vehicules, which are remote controlled by cables or electromagnetic waves, can assure many different functions in various hostile environments :

- direct audio-visual recognition,
- physicochemical measurements in real time or acquisition of data for batch treatment,
- sampling of solids, liquids and or gases,
- various remote operations on different materials equipment and installations in the presence of great potential dangers,
- remote handling and or transport of dangerous loads (presenting radiation or explosion risks),
- providing assistance to personnel during interventions,
- rescue of persons in difficulty.

Remote controlled equipment can be very precious in circumstances resulting from natural catastrophies (earthquakes, floods, avalanches, forest fires, volcanic eruptions) in addition to being indispensable for certain industrial applications (nuclear, chemical and biological). It is, of course, extremely advantageous to know how such equipment is used and what the limits are. One of the objectives of this training is to communicate this knowledge.

TRAINING AND RADIATION PROTECTION AT EDF

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ABSTRACT

E.D.F which is the only French Nuclear Electricity utility has decided in 1974 a large nuclear program which has been realized at the end of the seventies and at the beginning of the eighties.

In 1986, 42 nuclear units are in operation for a total of 45 000 MW and produced, since their first start-up, one thousand billions of Kwh, representing 186 years-reactor.

To reach this result E.D.F had to developp a strong training program for its personnel and enterprises staff employed for its own account.

This paper will talk about :

- *training devoted to the employees working in nuclear plants, and reception offered to outside workers,*
- *actions led to well know the field where workers perform,*
- *implementation means to monitor the personnel and to reduce his exposure at a low level,*
- *results obtained, after 10 years of reactors in operation.*

SCIENTIFIC GROUNDS FOR X-RAY SPECIALISTS INSTRUCTION IN RADIATION HYGIENE

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ABSTRACT

X-ray examination is known to be most contributary to the population irradiation dose. This dose significantly depends on the level of radiation safety competance inhering in the roengenologist dealing with the examination.

By means of analysis of the radiation safety fundamentals knowledge carried out prior to the course of training at the chair, by questionnaire and observation of 510 X-ray doctors' professional activities in 53 X-ray departments, a set of occupational and functional requirements on radiation safety has been elaborated.

These requirements determine the objectives of the training, the range of problems to be resolved by X-ray practice as well as the scope of the specialist's knowledge and techniques in regard to radiation protection measures. The requirements are primarily aimed at assuring radiation protection of the staff, patients and public at large in the process of medical X-ray studies. This aim is achieved by deciding two principal issues: firstly staff and patients safegard within X-ray examination, secondly organization and methods of realization of the above-mentioned requirements. For either of issues individual forms of the X-ray specialist's activities as well as the scope of necessary instruction and ability on his part are determined.

On the basis of the occupational and functional requirements an advanced system of X-ray doctors training has been worked out. Under present conditions it proves more progressive than the traditional one.

RADIATION PROTECTION AND NUCLEAR SAFETY TRAINING

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ABSTRACT

Radiation protection and nuclear safety training main aim is its criteria and philosophy understanding including the technical means applied. The training program adopted by the Argentine Radiological Protection and Nuclear Safety management is presented in the present work, which includes:

a) Professional level training: its main aim is to prepare people to face the problems of radiological protection and nuclear safety in the whole nuclear fuel cycle, a description is given of the itemized subjects, duration, and deepness. This training course is sponsored by the IAEA since 1981. From 1980, 232 professionals havebeing trained, 105 of them foreigners. A deep statistical analysis has being performed, classifing: professional orientation, country of origin and final insertion in to the argentine or foreign nuclear activity.

b) Technical level training: it is prepared to give elementary knowledges enough to identify the radiological risks inherent to its own tasks. Also a description is given of the itemizad subjects, duration and deepness.

c) Subsidiary training programs: a description is given of training aid given to other CNEA and external CNEA groups given to them teaching and scientific collaboration, as requested on the subjects of its competence.

INTRODUCTION

The main objective of Radiation Protection and Nuclear Safety training is to understand the criteria, philosophy and technical means applied in the specific aspects of modern science and technology.
With this objective in mind, the Radiological Protection and Nuclear Safety branch (Gerencia de Protección Radiológica y Seguridad - G.P.R.S.) of the Argentine National Atomic Energy Commission (C.N.E.A.) has implemented a Training Plan with special emphasis in the aforementioned subjects. This was intensely pushed in the last years (the professional course outlined in the following paragraph is one main example).
In general the training is directed to different levels of education and fields of appliccations as follows.

PROFESSIONAL LEVEL TRAINING

The objective of this level of training is directed to the professionals, of diferent fields of knowledge, directly engaged in the production, application, control and regulation of radioactive materials, radiation, nuclear energy, etc.. In this last field the idea is to cover the full nuclear fuel cycle.
To fulfill this objective the CNEA (created in 1950) through the G.P.R.S. has implemented since 1980 an annual, full dedication, Radiological Protection and Nuclear Safety Post-Graduate Course in coordination with the University of Buenos Aires (School of

Engineering) and the Health and Social Security (National Direction of Environmental Quality), agreement signed among the three institutions in 1980. The course is also sponsored since 1981 by International Atomic Energy Agency - IAEA.
It is directed to the training of professional people involved in different activities around the nuclear field e.g.:licensing authority personnel and its advisors, GPRS and outside advisors as: Federal Police Fire Brigade, Borders Control Brigade, etc.; utilities people responsible for Radiological Protection and Nuclear Safety, other groups involved in the use and handling of fissionable and radioactive materials and ionizing radiations. In general are professionals in or close to the decision level.
It is limited to 30 participants per academic year: 50 % offered by IAEA to candidates designed by the member states and the remaining 50 % are argentine or from other countries with which our country has special agreements.
In the following two charts we give a classification of the assistants taking into account in the first the profession and in the second the nationality.

CHART Nº 1

Profession	1980	1981	1982	1983	1984	1985	1986	1987
Engineers	20	16	21	14	11	13	9	18
Biologists	1	2	1	2	2	2	2	-
Physicians	3	2	2	1	3	4	2	-
Chemists	1	4	1	2	2	1	2	1
Biochemists	2	2	-	3	3	1	-	-
Physicists	-	-	3	4	5	5	6	2
Others	2	4	4	5	5	3	4	4
TOTAL	29	30	32	31	31	29	25	25

CHART Nº 2

Algeria	1	Morocco	1
Bolivia	9	Mexico	5
Brazil	4	Panama	1
Chile	10	Paraguay	4
Colombia	9	Peru	23
Costa Rica	1	Poland	1
Cuba	6	R.Dominican	1
Ecuador	8	Romania	1
El Salvador	1	Uruguay	6
Philippines	3	Venezuela	11
Guatemala	2		

	1980	1981	1982	1983	1984	1985	1986	1987
Argentines	26	14	19	12	15	14	14	10
Foreigners	3	16	13	19	16	15	11	15

Two clarify the first chart we must mention than in "others" we include third grade education people but not neccessary with universitary degree (e.g.: Fire Brigade officials, Science proffessors of chemistry and physics, etc.).
Also in this chart we must mention that, as will be reviewed latter, the curricula includes as a second part a full Nuclear Safety study plan for which the neccessary base is the first part on Radiation Protection and in general the working expectatives (after of the completion of the course) is better for engineers than for

other professionals, that is the reason why the number of engineers applying is higher.
In the second chart we show the number of Argentine and foreign people (also from 1980 until 1987) and the country of origin of the foreigners attendants. It must be pointed out that as since 1981 this course is also sponsored by the IAEA and also since 1983 several latinamerican countries make use of provisions made in bilateral agreements with Argentina the number of applicants from foreign countries has incresed. The highest number of foreign attendants usually cames from Peru, due to the fact that Argentina has under construction a Nuclear Research Center (which includes a Nuclear Research Pool type Reactor, a Radioisotope Production facilities, a Radiation Protection and Nuclear Safety National Center, etc.).

Organization of the Post-Graduate Course

The total duration is extended to 8 full months, with full dedication; this means a total of 1400 hours, from which 1000 are devoted to class room lectures including theoretical classes and problems and exercises.
Also a total of 250 hours are devoted to field experiences in several laboratories of the Ezeiza Nuclear Research Center, primarily at the G.P.R.S.
During the rest of the time (approx. 150 hours) the students, accompanied by professionals and technicians of the different specialities involved, make several technical visits to different nuclear instalations of the country, e.g.:

- Research Reactors, RA-3 (10 MW pool reactor for radioisotopes production) C.A.Ezeiza - Buenos Aires.; RA-6 (0.5 MW pool reactor training and research reactor) C.A.Bariloche -
- Radioisotopes production plant (I-131, Mo-99/Tc-99m, Cr-51, etc) C.A.Ezeiza - Buenos Aires.
- San Rafael - Mendoza, Uranium extraction and concentration; radiological aspects evaluation.
- Nuclear Power Plant Atucha I (370 MW - PHWR in operation since 1974) Buenos Aires.
- Nuclear Power Plant Atucha II (750 MW - PHWR in construction) Buenos Aires.
- Nuclear Power Plant Embalse (600 MW Candu in operation since 1984) Córdoba.
- I.M.P.S.A. - Mendoza - Mechanical Industries Pescarmona.

During the 8 months time they are covered theoretically and practically all the aspects, starting from the begining, of radiological protection and nuclear safety.
Previously to start with the specializad aspects, there is a four weeks length levelization course to take care of the different professions involved in it, with talks on : mathematics, physics, chemistry and biology.
The program includes the following subjects:

1st.Part. (130 hours duration) Basic Knowledges: Radioactivity; Nuclear reactions; Radiation and matter interactions; Neutronics; Radiation measurements.
2nd.Part. (520 hours duration) Radiation Protection: Radiodosimetry; Biological Effects; Radiation Protection Basic Criteria; Workers Radiological Protection; Public Ra-

diological Protection; External Radiation Protection System; Radioactive wastes management; Radioactive materials Transport; Radiological Protection in X-Rays Applications.

3rd.Part. (33 hours duration) Criticality: Criticality accidents Prevention.

4th.Part. (230 hours duration) Nuclear Safety: General Aspects; Safety in the Siting of N.P.P.; Safety in the Design of N.P.P.; Safety in the Operation of N.P.P.; Quality Assurance; Nuclear Risk in the Near Future.

Moreover, at the end of the course each participant must prepare a seminar, individual field of specialization oriented, with one week preparation time under the supervision of an specialist in the corresponding area and a formal presentation before a panel of professors.

They are also several lectures in very special topics given by argentine and foreign professors (e.g.: Dr. Jammet; Dr. Beninson; Prof. Bengstron; Mr. Lederman; etc.).

Finally, in accordance with the organization that grants the fellowships, IAEA, the students must fulfill all the programed activities and past partial and final examinations: the CNEA has the responsability to keep informed the IAEA about the achievements of every attendant and the general development of the course for every academic year.

TECHNICIANS LEVEL TRAINING

Technicians level in Argentina is equivalent to a High School Level in USA, but of 6 years duration, the first 3 years devoted to general knowledge in mathematics, physics, drawing, humanities, etc., and the last 3 years with specialization e.g.: in Chemistry, Electronics, Mechanics, Electricity, etc.

The technical level training course is organized in a way to allow the technicians (that work in the laboratories as supervisors with the professionals) to identify the radiological risks associated with his job with ionizing radiation.

This course started in 1980 in the G.P.R.S. for his own technicians and has being extended latter to technicians of other areas of CNEA and other institutions.

It is given once a year with a duration of 9 weeks, with full dedication, this amounts to a total of 360 hours, 270 of which are class room lectures, problems and practical applications. The rest of the time is dedicated to technical guided visits to nuclear and radioactive instalations with special emphasis of the tasks performed in them in relation with Radiological Protection.

The program covers the following subjects: Introduction to nuclear physics; Radioactivity, units; Radiation and matter interaction; Nuclear reactions; Radiation measurements; Nuclear Instruments; Introduction to radiodosimetry; External Irradiation; Biological Effects of radiation (stochastic or non stochastic); Radiological Protection, objectives, justificability, optimization, dose limitation; Operative Aspects of the Radiological Protection (workers and people protection); Radioprotection Technology, protection against external irradiation, protection against contamination; Radioactive Waste Management; Radioactive Materials Transportation; Some general ideas on Nuclear Safety.

As was explained at the very begining of this section, the appli-

cants must hold a degree of technician and up to now we take up to 20 persons per year and 150 got the approval certificate extended by CNEA - GPRS.

ADDITIONAL TASKS

Several times the GPRS has being requested to help other Institutions or CNEA groups with Radiological Protection and Nuclear Safety training short courses as a complement to different courses in related areas, for which it is convenient or imperative to have some knowledges on the subject, at least to be able to evaluate the risk associated with those activities.
Since 1981 we have prepared radiological protection training programs (and provided the neccessary professors) that have being use and are used in e.g.: Radioisotopes methodology and applications (radioinmunoanalysis oriented) offered by the CNEA Radioisotopes and Radiations Branch and a similar one by the University of Buenos Aires (Pharmacy and Biochemistry School). There is a general elementary program about Radiological Protection that covers such aspects as: Radiological Protection Basic Criteria; Operational Aspects; Technological Aspects; Radioactive Materials Transport; Radiations Biological Effects; Legal Aspects and National Regulations on Radioisotopes and Ionizing Radiations. It takes 30 hours of theoretical and practical lectures. Nuclear Engineering Post-Graduate Course, organized by the CNEA Research and Development Branch in agreement with the University of Buenos Aires (School of Engineering). In this case the program also includes some aspects of Nuclear Safety to cover knowledges in siting, design, operation and quality assurance of Nuclear Instalations and it takes 25 days (more or less 150 hours) of theoretical and practical lectures.
They are offered also several short duration courses (20 hours approximately) to collaborate with other Institutions related to ionizing radiations e.g.: Health and Social Security Ministery (12 courses per year); National Technological University (10 courses per year); Civil Defense Direction (Buenos Aires and provinces - 10 courses per year).

CONCLUSIONS

After many years of work we are prepared and able to organize courses in differen levels to lead this discipline in our country and to help other countries to organize their own programs. Moreover all the courses are always changing in their content to keep up to date information and knowledges in accordance wich the changing fields of the Radiation Protection and Nuclear Safety, this also include extra lectures about e.g.: TMI 2 and Chernobyl accidents, Mexico (Juarez city), Brazil (Goiania city), and Morocco accidents.

COLOR GRAPHICS DISPLAY OF LOW-HIGH ENERGY ELECTRON-PHOTON TRANSPORT USING EGS4†

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INTRODUCTION

Visual presentations are widely accepted and are a dynamic tool in education. Aptly stated by the often quoted phrase, *one picture is worth ten-thousand words.* The technique of visual presentations of Monte Carlo simulated interaction and transport using the EGS4* Code System[1] via SLAC Unified Graphics[2], connected to an IBM-5080 High Resolution Color Monitor, has been previously demonstrated[3,4].

EGS was originally developed by the Stanford Linear Accelerator Center (SLAC) and the High Energy Physics Laboratory (HEPL) at Stanford University as an analytical tool for engineering and physics. The EGS4 version released in 1985 is capable of simulating radiation transport from several TeV down to 1 KeV (photons) and 10 KeV (electrons and positrons). EGS can not only transport monoenergetic particles, but also particles generated from an energy distribution (*e.g.*, a β-decay spectrum). Although the purpose of this paper is to demonstrate the graphics capability as a teaching tool, the EGS code has been thoroughly verified by comparison with experiment and theory. The reader is referred to SLAC-265 for these comparative studies[1].

THE EGS CODE AND UNIFIED GRAPHICS: GENERAL CONSIDERATIONS

The EGS4 code is a general purpose computer program for the coupled transport of electrons, positrons and photons. The code is composed of 13 subroutines, block data and user interface. Two subroutines of interest are `HATCH`, which brings in media data created beforehand by a preprocessor code called PEGS4, and `SHOWER`, which enters the initial particle data and starts the simulation. The EGS code itself is "driven" by a *user-defined* code written in the Mortran3 language[5]. This User Code defines a specific geometry by means of subroutine `HOWFAR`. Another *user-written* subroutine, `AUSGAB`, scores the results of interest, such as particle fluence, dE/dx, etc. Each particle being transported by EGS carries specific information about its current location, energy, direction, etc.—in other words, a *vector* is associated with each transport that takes place. An auxiliary subprogram package called SHOWGRAF has recently been created[6] in order to plot these vectors. The various subroutines of SHOWGRAF make multiple calls to the SLAC Unified Graphics package[2], which has been interfaced to an IBM-5080 High Resolution Color Monitor by means of an IBM-3033 mainframe computer.

SIMULATION OF THE β-DECAY SPECTRUM OF ^{32}P

For the example to be demonstrated, a three dimensional geometry of a finger was simulated by means of concentric cylinders of finite length (8 cm). The geometric parameters

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* Electron-Gamma Shower (Version 4).

were based on measurements taken from a hand x-ray. The simulated bone diameter (1.0 cm) represents the average diameter of the middle, joint and proximid of the middle phalanges. The thickness of the dermis and epidermis was taken as 0.3 cm and 0.04 cm, respectively. The ^{32}P distributions considered were 1) a 4 cm isotropic line source positioned at the outer edge of the epidermis and 2) a 4 cm long isotropic volume source distributed uniformly throughout the bone. The bone was assumed to be of uniform composition and density (1.85 g/cm^3), as was the tissue (1.00 g/cm^3). All particles were followed down to 10 KeV.

The form used for the theoretical beta spectrum was that described by Konopiniski and Rose[7]. The probability density function (PDF) for ^{32}P was generated using the Fermi function tabulation of Fano[8] corrected for the screening effect. The spectrum was introduced into EGS in the form of a cumulative distribution function (CDF) look-up table.

To verify if the beta spectrum in EGS was correct, a comparison was made between the histogram obtained by sampling from the CDF and the theoretical distribution represented by the normalized PDF. As shown in Fig. 1, the histogram compares quite well with the ^{32}P spectrum.

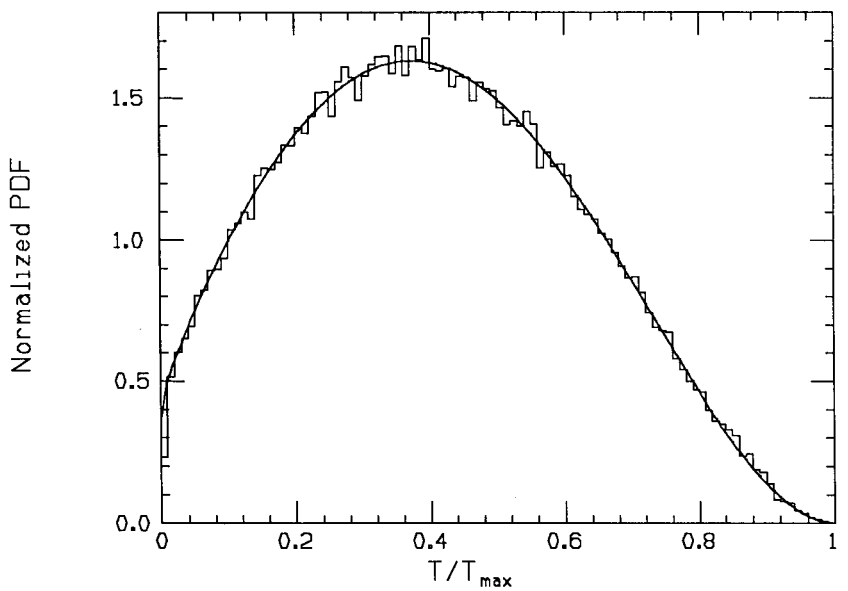

Figure 1. The β-decay spectrum of ^{32}P: sampled distribution (histogram), theoretical PDF (solid curve).

Figure 2 represents a view normal to the cylindrical axis of the simulated finger, where the ^{32}P line source is quite apparent at the top edge of the epidermis (*i.e.*, the outer cylinder). For purposes of clarity, all isotropically sampled betas emanating upwards were redirected 180° back into the finger. Some of the betas are observed to transport to the bone region (T_{max} ≈1.7 MeV corresponds to a CSDA range of 0.81 cm in water). A few bremsstrahlung photons are also shown (dotted lines). A slightly expanded end view of the same set of events is shown in Fig. 3. In all cases 200 events are shown.

The stochastic processes encountered in microdosimetry are nicely demonstrated in Fig. 4, where ^{32}P radioactivity is uniformly distributed within the bone. An expanded end view is show in Fig. 5.

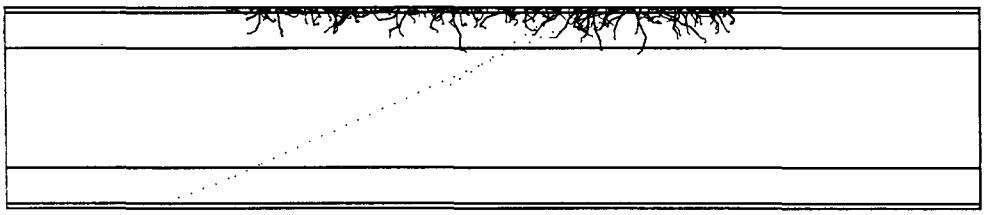

Figure 2. Side view of the finger with a ^{32}P line source positioned on the epidermis.

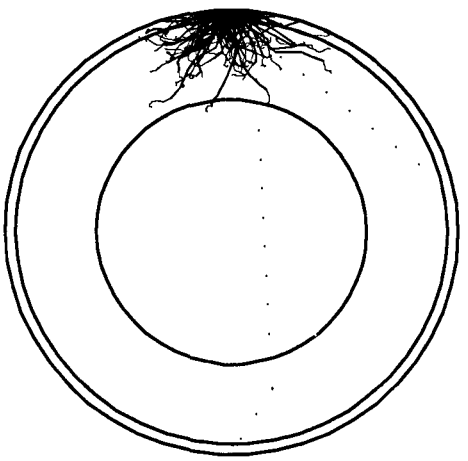

Figure 3. End view of the finger with a ^{32}P line source positioned on the epidermis.

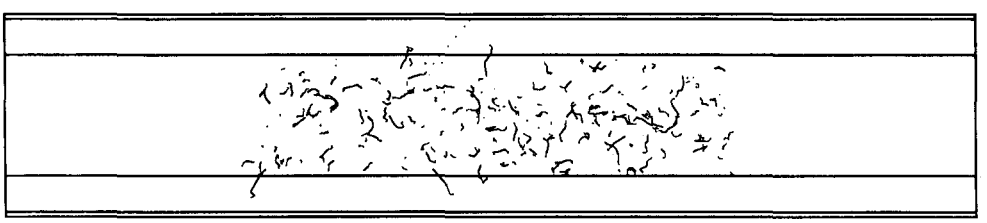

Figure 4. Side view of the finger with ^{32}P radioactivity uniformly distributed over a 4 cm length of the volume.

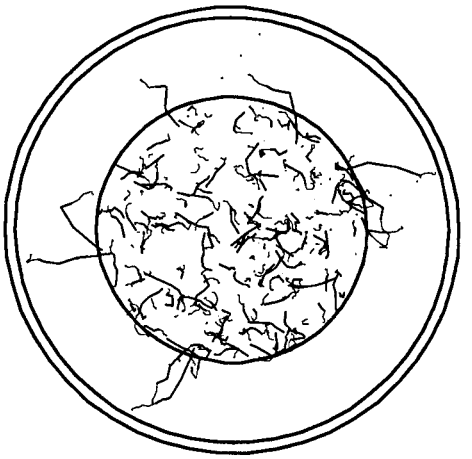

Figure 5. End view of the finger with ^{32}P radioactivity uniformly distributed over a 4 cm length of the volume.

DISCUSSION

The radiation transport and stochastic processes are clearly shown in the EGS4 generated 2-D black and white figures shown above. However, the full impact of the EGS4/Unified Graphics coupling can only begun to be appreciated as a dynamic teaching tool in the full 3-D color graphics that will be presented at the Poster Session of this meeting. In addition to the ^{32}P example above, additional examples will be presented at this session.

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† Work supported by the U.S.D.O.E. under contract DE-AC03-76SF00515.

TEACHING RADIATION PHYSICS AND RADIATION PROTECTION TO MEDICAL STUDENTS

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INTRODUCTION

Before even beginning, the good teacher needs to formulate answers to the following questions: Why is it necessary to teach a certain subject or to deliver a particular course? What should the content of the course be? How shall teaching take place? The goal of the course should be clear to both pupils and teachers right from the start. By commencing the course with a definition of goals, one at the same time indicates the content and provides the students with the necessary motivation. Further, if the content of the course is presented in a positive and appealing way, the students are likely to listen and actively participate in the work. In the following, there is discussed the teaching of a course in radiation physics for medical students in the context of the three questions stated above.

THE NEED FOR MEDICAL RADIATION KNOWLEDGE

Man is continuously exposed to ionizing radiation from natural sources, over which he has little control. This so-called background radiation is either "external" or "internal" depending on where the source is situated. Artificial sources also contribute to the total radiation dose. Figure 1 depicts the main sources (1). The natural radiation reaching man from outside consists of cosmic rays (13%) and radiation originating inside the Earth (16%). The internal radiation comes from the radioactive isotope ^{40}K (16%). The decay of radon also contribute to the annual radiation dose (33%). Artificially produced radiation, whose source is Man himself, includes the medical use of radiation (21%) and minor contributions from fallout, radioactive wastes from nuclear power plants and nuclear fuel (1%).

Although the annual radiation dose varies widely with geographical location, we have reason to believe that the values given are representative averages for industrialized countries. Man's own contribution to the annual radiation load is therefore slightly more than 20%. Among the artificial sources the medical contribution is overwhelming ; under normal conditions other artificial radiation sources are so small that they can be neglected.

On average, one fifth of the total annual dose received by the population in an industrialized country comes from medical radiation sources. The medical radiation dose per person is about 500 μSv/annum. It must be emphasized, however, that this is an average value and large individual fluctuations around this average may occur.

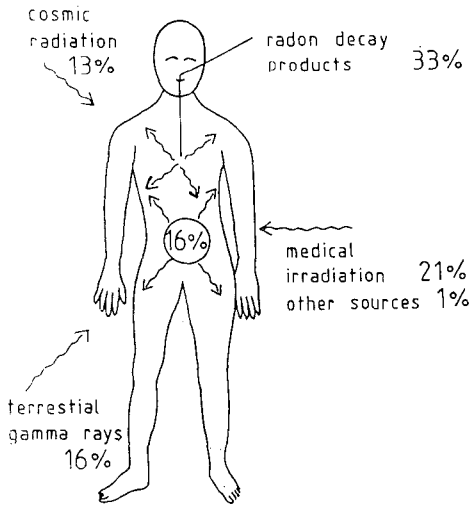

Figure 1 The main sources of background radiation including radiation from artificial sources.

The medical staff need to know how to minimize their own and their patients' radiation load. In many cases the radiologist will have to choose between different strategies of diagnosis or treatment. The positive medical benefit, i.e. a reliable diagnosis with visual material of good quality, will have to be weighed against the radiation dose the patient receives and to the possible complications.

There is one further aspect that is relevant in motivating students toward a course in radiation physics: That is the very fast development of radiological equipment and techniques. Present trends indicate that increasingly sophisticated methods will be used in diagnosis and therapy in the future: The rapid pace of development makes it essential that physicians have adequate scientific and technical knowledge of the equipment they are using so that it can be used in the most effective and safest way.

From the arguments given above it is self-evident that the medical staff of a radiological department should have a sound education in radiation physics and radiology. But it is also important that the ordinary physician have a thorough knowledge of the different methods he can apply, how they are carried out and what effects the complete procedure may have on patients and staff.

The aim of the course in radiation physics is to prepare the students for further studies in radiology, to show how physics can be applied to solve present medical problems and to create a sound basis for the understanding of phenomena and apparatus to be encountered in the future.

CONTENTS OF A COURSE IN RADIATION PHYSICS

The medical faculty of the University of Helsinki gives a compulsory course in medical physics and radiation physics. Figure 2 shows the volume of the course, how it fits into the gene-

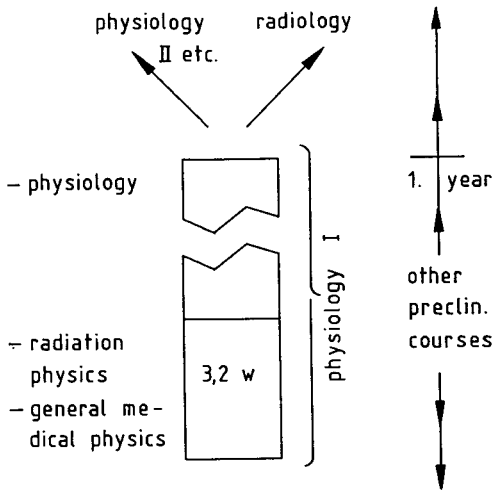

Figure 2 Graph showing the placement of medical physics and radiation physics in the program of medical studies. Also depicted are the connections with other disciplines.

ral course of studies, and its relations with other medical subjects. The course is given during the first year of studies (second semester) and the students are pre-clinical students still without a deep understanding of medicine. The course consists of lectures, exercises, demonstrations and laboratory work and is equivalent to a total of 3.2 working weeks (1 week = 40 hours). Table 1 shows the content of the course.

TEACHING FORMAT FOR RADIATION PHYSICS

The motivation of students towards the course varies during the term. The first part (general, medical physics) is not received as enthusiastically as the part dealing with radiation physics. From the outset the students have a fairly good idea as to how they can benefit from the course in radiation physics (everyone is aware of the usefulness of X-rays). But it is important that the students are motivated to study medical physics in all its forms. The complete course in medical and radiation physics consists of smaller parts (modules) that can easily be transported from one semester to another, if necessary.

Figure 3 shows different course formats. As the course proceeds from theory (lectures) to practice (clinical routines) the motivation of the students increases rapidly.

There are different ways of integrating the more theoretical aspects and clinical routines. One solution is to take the students to the clinics to become familiar with apparatus and see how it functions. However, it is often difficult to coordinate clinical visits with lectures and, as one alternative, slides can be shown describing the functioning of the apparatus. Also data from clinical routines can be analyzed. Video programs are included in the course from the start, to make the demonstrations more vivid. This audio-visual method is not excessively expensive: all that is required is a portable video camera and monitor. Since it makes possible a demonstration of the functioning principles of even large and expensive apparatus, the method is invaluable for pre-clinical courses.

Table 1 Content of the course in radiation physics

Atomic and nuclear physics :
- a basic knowledge

Radioactivity :
- radioactive decay
- half-life (phys. biol. eff.)
- series decay
- Tc-generator

Interaction between radiation and matter :
- different types of radiation
- modes of interaction
- detection of radiation
- scintigraph and gammacamera

X-rays :
- X-ray tubes
- X-ray spectra
- kVp, mA, s

Diagnostic radiology :
- radiological picture
- tomography
- CT

Nuclear medicine :
- isotopes
- statical investigations
- dynamical investigations
- radiation dose

Radiation biology :
- biological effects
- cell curvival curves
- radiation therapy
- radiation protection

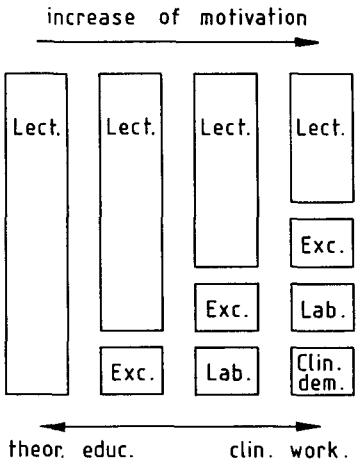

Figure 3 Different course formats. As the course proceeds from theoretical topics to clinical work the motivation increases.

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TRAINING OF NON-CREDENTIALED X-RAY OPERATORS IN RADIATION SAFETY

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The State of Washington, in common with approximately thirty other States in the United States, has no requirements for minimum training for operators of x-ray machines on human patients. In an attempt to correct the educational deficiency in non-credentialed operators, a training course has been developed by the State of Washington in cooperation with the Food and Drug Administration. Course topics, results of a post-class questionnaire, and changes in observations by State inspectors are discussed.

COURSE TOPICS

Subjects covered in the day-long course include Basic Radiation Perspectives, Radiation Production, Radiation Interaction with Matter, Biological Effects, Risks of Radiation Exposure, Radiation Protection, Image Quality, and Quality Assurance Techniques. In all cases, of course, the intent is to supply attendees with enough information so that they may be aware of the reasons for certain actions and concerns. The effect of changes of kVp upon resulting radiographs and upon scatter radiation indicate to students the need for kVp selection appropriate for a particular examination as well as the State requirement that no one be in the room during radiography unless that individual is necessary for the procedure.

Early attendee critiques requested answers to patients' questions concerning radiographic examinations. Therefore, a session entitled "Patients' Concerns" was added to the standard course.

POST-CLASS QUESTIONNAIRE

Originally, pre- and post-class written examinations were given to students attending the class. However, written examinations did not measure the behavior modification expected from the class. The intent was not to turn non-credentialed technologists into physicists; the course was to motivate attendees to change their working conditions in such a way so as to obtain better radiographs with less radiation exposure to themselves and their patients.

In 1986 and 1987, questionnaires were mailed to attendees of the nine separate courses held in 1985 and 1986, respectively, a total of 213 x-ray facilities throughout the State. Responses were received from 70 facilities. The questions were designed to identify technical, administrative, and engineering changes, and work performance changes observable by both managment and staff. The questions also noted relationships between attendees and patients and fellow employees regarding concerns of radiation protection.

The following equipment changes were found to be a direct result of course attendance:

25% had replaced their screens;
20% had purchased thermometers, timers, and other darkroom supplies;
16% had upgraded major equipment or components (collimator, processors)
13% replaced the safelight;
10% began a commercial processor quality assurance contract;
7% obtained a commercial gonad shield;
7% bought patient-positioning aids;
7% replaced or bought lead aprons and/or gloves;
4% had introduced the use of a grid for chest examinations.

Regarding changes in operators' technical procedures:

30% had reduced exposure time;
18% began to follow an established film development guide;
18% began film badge monitoring;
16% performed a film fog test and made some corrections;
13% began processor sensitometry/densitometry testing;
10% began to refer to technique charts routinely.

In terms of observable changes in the quality of patient care:

50% reported "some" to "considerable" exposure time reduction;
36% improved collimation practice;
25% felt that occupational exposure was also reduced;
20% reported improving patient gonad shielding practices.

Regarding changes in radiographic quality:

70% experienced image enhancement;
50% reported "some" or "marked" reduction in film retakes;
20% indicated improved patient positioning.

INSPECTIONAL OBSERVATIONS

Routine post-course inspections were made of 49 facilities with operators that attended the course and compared to observations at 59 facilities without operator attendance. The 49 inspected facilities in the study group were not selected simply because of course attendance; these were facilities which were already due to be inspected according to the State's routine priority scheduling. The 59 control facilities were randomly selected to match the same proportion of types of facility (dental, medical, chiropractic, etc.) and geographic locations as those in the study group.

For dental facilities, those without training showed an average decrease of patient exposure of 6% (320 to 300 mR) between inspections performed in 1983-84 and those performed in 1986. Facilities with training showed a 14% average drop (280 to 240 mR) during the same period of time. Washington State NEXT data for bitewing examinations in 1984-85 was 284 mR. Seemingly, facilities willing to send technologists to the course already are motivated toward lower patient exposures, and attendance at the course induces further exposure reduction.

For PA chest examinations, facilities without training reduced patient exposure by 23% average (22 to 17 mR) within the same time period. Facilities with training decreased patient exposure by 37% (19 to 12 mR). Again, facilities motivated to training seemingly began at a lower patient exposure and decreased the patient exposure even more after the training.

Similar data occur with studies of other examinations.

Data was also collected to compare facilities' noncompliance rates as a function of operator training. During the same period of time, 57.3% of all Washington State facilities were found to be in compliance with State regulations. Those facilities which sent operators to the course had a pre-training compliant inspection rate of 61% which increased to 74% during the first inspection after the training course.

The types of noncompliance also change with training. Pre-training inspections noted that most noncompliances were associated with lack of adequate collimation, inadequate gonad shielding, and improper exposure switch placement. Post-training noncompliances showed the highest rate in improper exposure switch placement and inadequate film processing, with other noncompliances at a much lower rate.

CONCLUSIONS

Definite improvements in State inspection compliance rates, patient and occupational exposures, and actions taken to improve radiographic quality can be achieved from even a single-day training activity. The improvements noted appear to be greater than those accomplished with programs exclusively emphasizing inspection and enforcement.

ENVIRONMENTAL RADIOACTIVITY AND MAN
- THE 1988 SIEVERT LECTURE -

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INTRODUCTION

Rolf M.Sievert
1896-1966

Rolf Sievert was born in 1896, the year in which Henri Becquerel discovered the phenomenon of radioactivity. This can be regarded as one motive to devote this Sievert Lecture some new and actual aspects of environmental radioactivity and its impact on man. Rolf Sievert himself was strongly interested in this field. Under his leadership important studies on the sources and levels of natural radiation and radioactivity in our indoor and outdoor environment have been carried out in the Department of Radiophysics of the famous Karolinska Institute in Stockholm. As an example I want to mention the doctor thesis of his co-worker Bengt Hultqvist (1956). This study revealed for the first time the wide variation range of radon (^{222}Rn) levels in the indoor air of Swedish houses. At this time Sievert and Hultqvist believed that the high radon levels observed in some houses would be a specific Swedish problem, restricted to houses built of concrete containing alum-shale with a high radium content. These pioneer studies have been successfully continued under the leadership of Bo Lindell and now of Gunnar Bengtsson at the National Institute of Radiation Protection in Stockholm which had its origin in Rolf Sievert's department.

Today we know that radon in houses is a world-wide problem. Taking into account the improved models on lung dosimetry, it has become evident that inhalation of the short-lived decay products of radon (^{222}Rn) in indoor air yields by far the highest contribution to the total population exposure from natural and man-made radiation sources (UNSCEAR 1982; NCRP 1987). The control and mitigation of radon in existing houses and the implications on building codes for new houses represent one of the most important challenges to the radiation protection community and the responsible authorities. The solution of this radon problem in houses is strongly linked with that on the risk of lung cancer which might be associated with this indoor exposure to radon and its decay products. The discussion of this question will be the subject of one part of this Sievert Lecture.

The other part of this lecture is concerned with a problem in the outdoor environment: the transfer of radionuclides through food chains to man. The actuality of this field is well known; it is related with the reactor accident at Chernobyl that happened two years ago. In this part of my lecture I want to describe some results of our work at the GSF-Institute for Radiation Protection on the development of dynamic radioecological models and their validation by measurements after the Chernobyl accident.

ACTIVITY TRANSFER THROUGH FOOD CHAINS

Radioecological Modelling

The transfer of radionuclides through food chains to man is, besides the dose from external irradiation, the most relevant exposure pathway in the case of a contamination of the outdoor environment. For chronic operational releases of activity from nuclear facilities, rather simple transfer models are used which consider only the reached steady-state equilibrium conditions. Many of these simplifying models are designed to lead to an overestimation of the actual exposure.

In the case of high accidental releases more realistic radioecological models are required to enable appropriate decisions on eventually necessary countermeasures. In particular, such models should provide with sufficient accuracy a prediction of the expected activity concentration in relevant foodstuffs as function of time and of the cumulative potential radiation exposure via these pathways. Annual mean values of the overall-transfer coefficients of strontium-90, iodine-131 and cesium-137 in major groups of foodstuffs have been estimated by a regression analysis of data from nuclear weapon tests. These results are summarized in the reports of UNSCEAR /1977, 1982). These values refer, however, to a long-term and a rather uniform spatial deposition of these radionuclides. They cannot be applied to the predictive evaluation of the dietary activity intake and the corresponding ingestion dose to the local and regional population in the case of short-term, accidental releases. This has been confirmed by the observations after the Chernobyl accident.

Already in 1978 we started in our laboratory the development of realistic dynamic models for the predictive evaluation of the spatial and temporal distribution of the external and internal radiation exposure which might occur as a consequence of possible accidents at nuclear facilities. Main emphasis was laid to the simulation of the transfer kinetics through food chains, taking into account typical agricultural conditions and their seasonal variation in the F.R. of Germany. First results of this so-called ECOSYS model have been published in a comprehensive report in 1982 (Matthies et al. 1982). An improved version of this model has been presented three years later (Müller et al. 1985).

Results of the ECOSYS Model

In the following some selected results of this dynamic model are outlined and discussed. For some important radionuclides figure 1 shows the effective dose equivalent commitment from the intake with the total diet, integrated over an intake period of 50 years, as function of the calendar month in which the deposition occurs. The data in this figure refer to the first, simplified version of the ECOSYS model and take into account the average food consumption rates of adults in the F.R. of Germany (Matthies et al. 1982). They are normalized to a deposition density of 1 kBq m^{-2} and an interception factor of 1 on plant surfaces.

The strong seasonal variation of the dose commitment per unit deposition density is due to the direct deposition on plant surfaces and the subsequent uptake through the cuticula of plants during the vegetation period. This variation is, therefore, more pronounced for those radionuclides, like cesium-137, for which the root uptake from normal soils is rather low. The dose contribution by uptake from soil is, under the vegetational conditions in Germany, the dominant source term for depositions during the winter months (see figure 1).

Another sensitive parameter is the interception factor of plants which depends on the type and growth status of plants, and on the type of deposition (wet or dry deposition). Figure 2 shows the cumulative ingestion dose as function of time after deposition for different values of the interception factor, as it results from an improved version of the ECOSYS model for a deposition of 1 kBq m^{-2} of cesium-137 in the first days of May (Müller et al. 1985, 1987). For wet deposition by rainfall the model proceeds from an interception factor of about 0.1-0.2. Higher values in the range of 0.5-1 may be appropriate in the case of dry deposition.

The increase of the ingestion dose with time after deposition, shown in figure 2, reflects again the strong influence of direct foliar uptake. For depositions during the vegetation period the contribution from this source determines the ingestion dose received during the first two years. The contamination of foodstuffs due to the uptake from soil is a long-term process, the kinetics of which are mainly determined by the migration and sorption processes of the deposited radionuclides in the soil. The dose contribution from this pathway is indicated by the dotted line.

Due to the direct deposition and foliar uptake by plants, the specific activity of milk, meat, leafy vegetables and cereal products varies strongly with time after deposition. For example, figure 3 shows the variation of the mean specific activity in major groups of foodstuffs, as it follows from the improved version of the ECOSYS model for the deposition of cesium-137 in early May, normalized to a deposition density of 1 kBq m^{-2} and a mean interception factor of 0.2 in the area where the foodstuffs are produced.

Under these conditions the model yields a peak value of the specific activity in milk and beef at about 5 or 30 days after deposition, respectively. For the rapid decrease afterwards, the growth of biomass is an important factor. At the beginning of the winter period, about 6 months after deposition, the specific activity in beef and milk increases again due to the feeding of contaminated hay and silage which were harvested during the summer period. After this period of dry feeding the contamination decreases and approaches after about two years the contamination level, which is caused by the cesium uptake from the soil.

Comparison with Measurements after Chernobyl

An improved version of the ECOSYS model which involved also a probabilistic dose assessment, was published a few months before the Chernobyl accident (Müller et al., 1985). This accident caused a rather high activity deposition in southern Bavaria in the days between April 30 and May 2, 1986; the main deposition occurred during a heavy rainfall with thunderstorm in a period of one hour in the afternoon of April 30. At our laboratory in the northern area of Munich we measured a total deposition density of about 400 kBq m^{-2} from which about 100 and 20 kBq m^{-2} were allotted to iodine-131 and cesium-137, respectively (ISS 1986, Hötzl et al. 1987). It is interesting to note that the latter value was comparable with the cesium deposition reported for the region of Kiew. This experience shows that, under adverse weather conditions, the activity release from major reactor accidents may lead to rather high depositions far away from the emergency site.

After the Chernobyl accident we have made more than ten thousand measurements in different environmental media and foodstuffs, as well as whole-body counting measurements. These data offered the possibility to validate the predictions of the ECOSYS model and to improve the model. In addition, in cooperation with other institutes, we have carried out specific feeding experiments with cows and pigs to study the transfer into milk and meat, and have started laboratory experiments to investigate the kinetics and mechanisms of the foliar uptake of radiocesium into grass and leaves.

In general, with a few exceptions, the observed mean values of the specific activity in relevant foodstuffs as function of time after deposition agreed rather well with the predictions of the ECOSYS model (Müller et al. 1987). This was particularly valid for milk. As an example, figure 4 shows a comparison between the measured daily mean values of cesium-137 activity in milk from a dairy-farm near Munich during the period from May 1986 to August 1987. The closed line in this figure indicates the variation with time which was calculated with the ECOSYS model, inserting a deposition density of 20 kBq m^{-2} and an interception factor of 0.2 for grass. As mentioned before, the second broad peak during the winter period (200th - 400th day) is due to the feeding of contaminated hay, which was cut mainly in June 1986. Also for other major foodstuffs the measured and calculated mean values of the time integral over the specific activity, integrated over the first year after deposition, agree in most cases within a factor of 2.

The final objective of such models should be the assessment of the cumulative activity intake with the total diet and the corresponding dose. Since the Chernobyl accident many whole-body-counter measurements of persons living in the region of Munich have been carried out. The cumulative frequency distribution of the total intake of radiocesium (cesium-137 + cesium-134) during the first year, derived from these measurements, is shown in figure 5. The most reliable distribution function is represented by the curve for 21 adults who were measured at least once per month. In the same figure the distribution functions are plotted resulting from the probabilistic version of the ECOSYS model for an interception factor of 0.1 and 0.2, respectively (Müller et al. 1987).

This comparison indicates that in the given situation, the ECOSYS model leads for radiocesium to an overestimation of the cumulative intake or dose by a factor of about 3 or 5, respectively, depending on the inserted interception factor. This deviation can be explained by several reasons:

(1) The model assumes that all consumed foodstuffs are also produced in the same region of high contamination;
(2) Normal food consumption rates have been assumed, whereas the real consumption rates of milk, fresh vegetables and partly also of meat were somewhat lower, at least until early summer 1986;
(3) Normal feeding practices have been assumed.

Probably, the main part of the observed deviation may be attributed to the first cause, because a considerable fraction of the foodstuffs consumed in the rather highly contaminated area of southern Bavaria will not be produced in this region. This influence of the spatial distribution of major foodstuffs is confirmed by the whole-body measurements of persons in other parts of Germany, where the deposition density after the Chernobyl accident was considerably lower, in average by a factor of about 5.

In conclusion it can be stated that dynamic radioecological models can be a very important and useful tool for emergency management at early and later times after nuclear accidents, in particular for an appropriate setting of intervention levels for major groups of foodstuffs (Jacobi et al. 1987). Our future work on this field is directed to the improvement and extension of the ECOSYS model, taking into account all experience obtained after the Chernobyl accident. This involves also the dose assessment from external radiation in urban and rural environments. Based on the ECOSYS model, we intend to develop in cooperation with the Federal Health Office, a real time expert system. It will be part of the central monitoring and warning system in the Federal Republic of Germany which is provided by the German government for the future management of emergency situations.

LUNG CANCER RISK FROM INDOOR RADON

The second part of this Sievert Lecture is devoted to an entirely different field of environmental radioactivity. It deals with the population exposure to radon (^{222}Rn) and its short-lived decay products, giving main emphasis to the assessment of the attributable risk of lung cancer.

General Aspects and Problems

Since the first studies of Rolf Sievert and his co-workers, comprehensive surveys on radon in houses have been carried out or started in several countries. One of the most striking results is the observed large variation range of the radon level from house to house. It covers a range from a few up to several thousand Bq m^{-3}, and in a few extreme cases values up to about 100 000 Bq m^{-3} have been measured. In most houses with strongly enhanced levels the main source term is the radon supply from the underlying soil. Especially this radon influx from the soil is a very complex process. Under these circumstances a realistic quantitative prediction of the radon level in existing houses seems rather impossible. Consequently, measurements are finally the only reliable method for the detection of houses with high radon levels. A flood of measured values have been reported in the last years. But these data alone do not solve the radon problem. Important is the fact that during the last five years efficient and low-cost techniques for the reduction of the radon influx from soil, like sub-pressurization, have been developed and tested.

Under these aspects the question arises: What is a reasonable lower boundary below which no actions should be taken? Or, with other words: Where is the borderline between normal natural radioactivity and man-made modifications of this natural source of exposure? One should also have in mind that this is not a new source. Our ancestors were exposed to radon levels which were comparable with the values measured now. But they did not worry about it, because in Germany the saying goes: "Was ich nicht weiß, macht mich nicht heiß", which corresponds to the English saying "Ignorance is bliss". But today we are aware of the problem, and the guiding principle of precaution to which we are obliged in radiation protection makes it necessary to follow the ALARA-principle, or the principle of optimization, respectively.

In publication 39, the ICRP has made a first attempt to outline the basic principles for the limitation of exposures to natural sources, but there is also stated that for several reasons "it would not be helpful to suggest a generally applicable value of an action level" for radon in houses (ICRP 1984). It is obvious that a simple cost-benefit analysis alone cannot give the final answer to the setting of such action levels. Guidelines of national authorities must be in accordance with the demands of public health, having in mind the collective detriment of health from the controllable fraction of this exposure. On the other hand, they should warrant a margin for the individual freedom and voluntariness of decision-making. This underlines the necessity to inform and advise the general public in a clear and objective way.

It seems reasonable to apply a step-wise system of action levels for the concentration of radon or its decay products in indoor air. The higher the observed concentration, the shorter should be the time period for remedial actions, if they are possible. The deciding quantity is the cumulative exposure. Thus action levels c_{AL} of the concentration should be derived from a primary action level E_{AL} of the cumulative exposure, applying the relationship

$$c_{AL} \times T_A = E_{AL}$$

where T_A is the time period in which the action should be taken. The possible lung cancer risk associated with this indoor exposure is an important criterion for the final judgement of the radon problem in houses and the appropriateness of mitigation measures. The epidemiological studies of various groups of Rn-exposed miners, as well as the findings from the externally irradiated atomic bomb survivors indicate that the human lung, and particularly the bronchial epithelium, is a rather sensitive tissue for carcinogenic effects of ionizing radiation.

On the other hand, up to now no direct epidemiological studies on lung cancer from indoor exposure to radon and its decay products are available which enable a quantitative risk assessment. Some smaller pilot studies in Sweden and Canada indicate a positive correlation. But their statistical and systematic error ranges are quite large, taking into account the overwhelming influence of smoking and the uncertainties of long-term exposure estimates. Case-control studies among non-smoking females seem to be more favourable but one should have in mind that lung cancer in real non-smokers is a rather rare event.

Comparison of Risk Estimates

Under these circumstances, risk assessments based on data from Rn-exposed miners seem to be more reasonable, taking into account appropriate correction factors for the different conditions in mines and houses. During the last years reports of three expert groups on this subject have been published in which different approaches have been considered (NCRP, 1984; ICRP, 1987; NRC-BEIR IV, 1988). The central estimates of the age-averaged mortality risk of lung cancer per unit of indoor exposure to ^{222}Rn-decay products, resulting from these studies, are summarized in table 1. The exposure is expressed in terms of the equilibrium-equivalent exposure to ^{222}Rn (unit: 1 Bq a m^{-3} = 8760 Bq h m^{-3}); risk values per unit of the potential alpha energy exposure to ^{222}Rn-decay products are given in brackets (unit: 1 WLM = 0.0035 J h m^{-3} $\hat{=}$ 72 Bq a m^{-3}).

Table 1: Age-average risk of lung cancer per unit of indoor exposure to ^{222}Rn decay products; comparison of central estimates from different studies and types of approaches

Study, type of approach		Attributable cases/10^6 persons per Bq a m^{-3} (per WLM)			
RELATIVE RISK PROJECTION					
ICRP (1987) Refer. Population	males	5.3	(380)	aver. 3.2	(230)
	females	1.1	(80)		
NRC-BEIR IV (1988) US-Population	average, both sexes			4.9	(350)
ABSOLUTE RISK PROJECTION					
NCRP (1984)	average, both sexes			1.8	(130)
ICRP (1987)				2.1	(150)
"Dosimetric approach" (ICRP 1987)				1.4	(100)

In all these studies a proportional exposure-risk relationship has been assumed which yields the best fit to the miner's data in the relevant exposure range. In a subgroup of uranium miners in the CSSR a statistically significant excess frequency of lung cancer has been already observed at an exposure level of about 0.14 J h m^{-3} = 40 WLM. This "statistical threshold" lies only a factor 2-5 higher than the mean lifetime exposure of the populations in most countries to be be expected from the presently measured mean concentrations in indoor air.

More controversial are the conceptions for the risk projection as function of time after exposure. In the NCRP study an absolute risk projection model has been assumed. The data on lung cancer in Rn exposed miners as well as in the A-bomb survivors indicate, however, that the appearance rate of radiogenic lung cancer is similar to the age-dependent distribution of the normal cancer rate in a comparable non-exposed population. On the basis of this finding, in the two more recent studies (ICRP 1987, NRC 1988) preference is given to a relative risk projection model. In the ICRP-report a constant relative excess risk, whereas in the NRC (BEIR IV)-report a slight decrease with time after exposure was inserted. The ICRP Task Group made also a comparison with the results of absolute risk projection models (see table 1). This includes as well the so-called "dosimetric approach" which proceeds from the age-averaged reference risk coefficients of 1 x 10^{-3} Sv^{-1}, recommended by the ICRP (1981) for each of the two target tissues in the lung: the tracheobronchial epithelium and the pulmonary tissue.

It is important to note that the estimates from different types of approaches differ only by a factor of 3 to 4. This range is probably smaller than the total uncertainty of the input data involved in these models.

The relative risk projection models yield primarily values for the relative excess lifetime risk. Due to this proportional hazard model, at equal exposure conditions the relative risk is nearly equal for males and females. Consequently, the absolute number of excess cases increases proportionally with the normal lung cancer frequency in populations, and will be higher for males than for females, as shown in table 1.

This is one of the main reasons for the different absolute numbers averaged over both sexes, which result from the ICRP and NRC studies (see table 1). The ICRP estimate refers to a reference population with a life expectancy at birth of 70 a (males) and 75 a (females), and a normal lung cancer frequency of 600 and 120 cases (average both sexes: 360 cases) per 10^6 persons per year for the male and female population, respectively. These values correspond roughly with the worldwide average frequency, resulting from the reported values in populations with high life expectancy. The NRC estimate refers to the US-population only, for which in the last years an annual lung cancer frequency of about 450 cases/10^6 persons averaged over both sexes, has been reported.

The important question arises: Which fraction of the observed lung cancer frequency might be associated with the indoor exposure to radon decay products? In the UNSCEAR report (1982) a country-averaged mean value of about 15 Bq m^{-3} for the equilibrium-equivalent ^{222}Rn-concentration in indoor air and an occupancy factor of 0.8 is assumed. This

would yield a mean indoor exposure of 12 Bq a m^{-3} per year. Applying the risk coefficients for the reference populations resulting from the ICRP relative risk approach (see table 1), a chronic exposure at this level would yield an attributable annual frequency of about 64 and 13 cases (average 38 cases) per 10^6 males and females, respectively. This means that nearly 10 % of the totally observed lung cancer frequency among males and females might be associated with this indoor exposure. The attributed relative loss of life expectancy would be about 0.06 %, averaged over both sexes, corresponding to an absolute value of about 20 days (ICRP 1987). On the basis of absolute risk projection models, these values are only about a factor of 2 lower.

This risk assessment indicates that if smoking is excluded, the indoor exposure to radon decay products is probably the most important exogenic cause of lung cancer in most countries. It underlines also the necessity of mitigation measures in those houses where strongly higher concentrations due to the radon supply from soil have been measured, taking into account that this source term is to a large extent controllable and can be efficiently reduced by rather low-cost techniques.

Risk Models: Problems and Perspectives

The previous considerations may lead to the impression that a relative risk projection model enables a realistic assessment of the lifetime risk of lung cancer by ionizing radiation. This conclusion is premature. Some epidemiological data on lung cancer in American miners suggest a decrease of the relative risk with time after exposure (NRC 1988). A similar tendency can be deduced from the incidence of lung cancer in the A-bomb survivors (Yamamoto et al. 1987, Preston et al. 1987) if the data for all age groups at time of exposure are combined. However, in both cases due to the large statistical error range, this decrease cannot be quantified. Recently the results of the extended follow-up of uranium miners in the CSSR have been published (Sevc et al. 1988). Figure 6 shows the time dependency of the relative lung cancer risk which can be evaluated from these data. A maximum value in the range of 5 to 10 is reached 10-20 years after start of mining. Afterwards the relative risk decreases strongly, and at the end of the follow-up period, 32 years after start of exposure, a value of only about 1.2 has been found. The decrease is highly significant as can be seen from the confidence limits in this figure.

These epidemiological findings strengthen the doubts in the appropriateness of relative risk models which assume a constant relative risk with time after exposure. Relative risk models also do not lead to a conclusive description of the interaction between inhaled radon daughters and cigarette smoke for the induction of lung cancer (ICRP 1987, NRC 1988). Thus, neither absolute nor relative risk projection models may be appropriate for radiation risk assessments in every instance. This conclusion is also valid for other types of radiation-induced cancers (NRC 1980, Upton 1984).

The future improvement of risk models should take into account that carcinogenesis is a multifactorial multistage process. There is strong evidence that ionizing radiation, particularly at low doses, acts primarily on earlier stages of this process. On the other hand, the effects

of aging and the influence of other promoting factors affect the later stages. Thus the latency distributionof radiogenic lung cancer is mainly determined by aging and other factors, like smoking, which stimulate the tumor growth. Consequently, a risk model considering the multiplicative interaction of initiating and promoting effects seems to be more reasonable.

In a simplified way, a multiplicative two-stage model leads to the following general relationship for the age-specific incidence rate $\lambda(t)$ of lung cancer as function of age t: $\lambda(t) = P(t) \times \mu(t)$. In this equation P(t) characterizes the latent hazard potential of potentially malignant cells which is built up by all initiating agents acting in the early phases of the carcinogenic process. The function $\mu(t)$ defines the probability for the growth and manifestation of tumors per unit time. This function increases strongly with age and will be amplified or shifted to lower ages by smoking.

If ionizing radiation is mainly regarded as an initiating agent, it leads to an increase of the latent hazard potential P(t), but causes no significant change of the promotion probability $\mu(t)$. This is shown schematically in figure 7 for the example of a radiation exposure or dose D received at an age t_e = 30 a; for simplicity no decrease of the radiation-induced increment $P_r(t,D)$ due to the removal or inactivation of radiation-induced potentially malignant cells is assumed. The total hazard function follows to

$$\lambda(t,D) = [P_0(t) + P_r(t,D)]\ \mu(t)$$

for $t > t_e + \tau$ (τ = time lag between irradiation and manifestation) which leads to a relative risk function:

$$\frac{\lambda(t,D)}{\lambda_0(t)} = \frac{P_0(t) + P_r(t,D)}{P_0(t)} = 1 + \frac{P_r(t,D)}{P_0(t)}$$

As the baseline potential P_0 caused by initiating agents other than radiation increases with age, the radiation-induced relative risk decreases with time after irradiation, as shown in the lower graph of figure 7. Furthermore, in accordance with epidemiological findings, the initial relative risk decreases with increasing age at exposure.

Summarizing, a multiplicative, two-stage model suggests a decrease of the relative lung cancer risk with time after irradiation. Consequently the assumption of a constant relative risk might lead to an overestimation of the real lifetime risk.

These considerations shall indicate some perspectives for the future improvement of risk models. Similar views might be valid also for other types of radiation-induced cancer. It is necessary to build a bridge between epidemiology and molecular tumor genetics. In my opinion, only the combination of the findings and experience from both these fields can lead to more realistic estimates of the cancer risk at low doses required for a well-founded system of dose limitation in radiation protection.

FINAL REMARKS

Such thoughts are in accordance with Rolf Sievert's ideas. His pioneer work has demonstrated the success of interdisciplinary cooperation and the strong linkage between research and practical radiation protection. The problems of environmental radioactivity outlined in this lecture confirm the necessity of this view. Like in the past, the concepts and experience of radiation protection can be and should be a guiding model for the protection against toxic non-radioactive chemicals in our environment.

Acknowledgements. I'm greatly indebted to all scientific and technical co-workers in my institute who have contributed in a very efficient way to the work outlined in this lecture.

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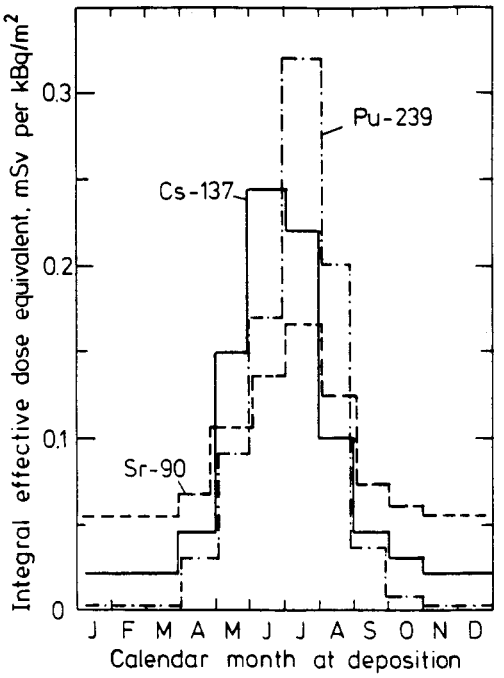

Fig. 1. Calculated ingestion dose commitment as function of the calendar month at deposition, normalized to a deposition density of 1 kBq m^{-2} and an interception factor of one (ECOSYS-model, 1982).

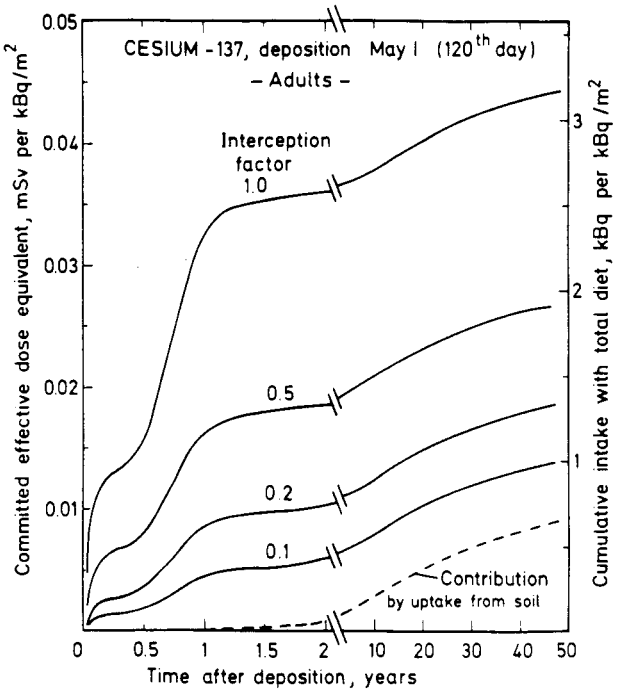

Fig. 2. Cumulative dietary intake of cesium-137 and dose to adults as function of time, predicted with the ECOSYS-model for an uniform deposition of 1 kBq m^{-2} on May 1.

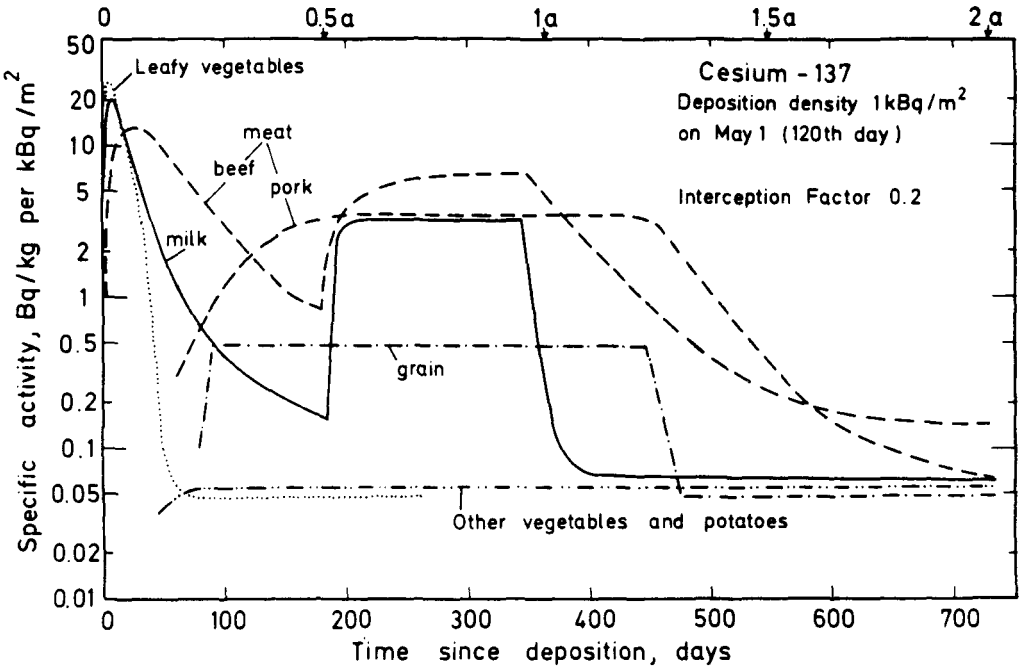

Fig. 3. Mean specific cesium-137 activity in food stuffs as function of time after deposition, expected from the ECOSYS-model for a uniform deposition of 1 kBq m^{-2} on May 1 (interception factor = 0.2).

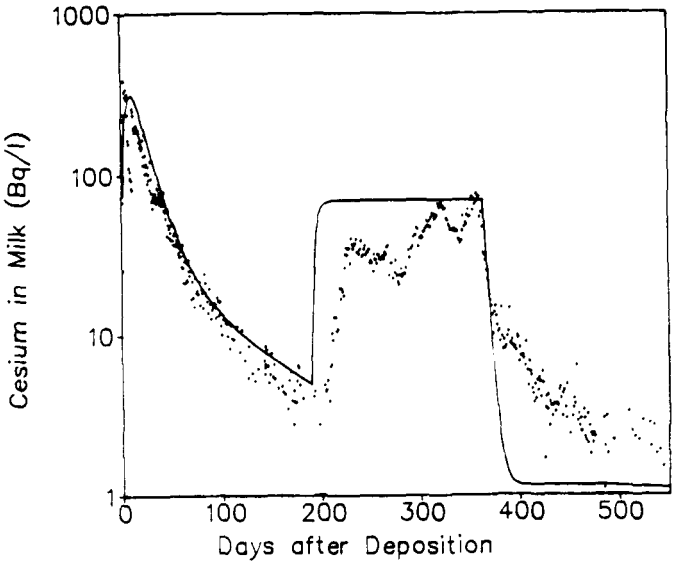

Fig. 4. Measured daily mean values of the cesium-137 in milk from a dairy farm near Munich after the Chernobyl accident (points); comparison with the time dependency predicted with the revised ECOSYS-model (closed line).

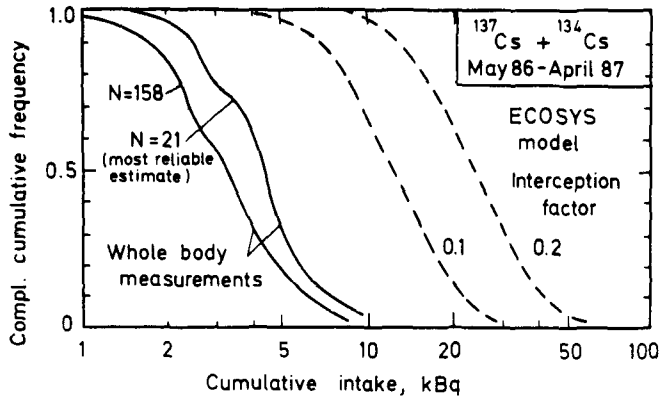

Fig. 5. Complementary cumulative frequency distribution of the radiocesium intake by adults during the first year after the Chernobyl accident; comparison of results from whole body measurements of persons in the region of Munich (closed lines) with the predictions of the probabilistic ECOSYS-model (plotted lines).

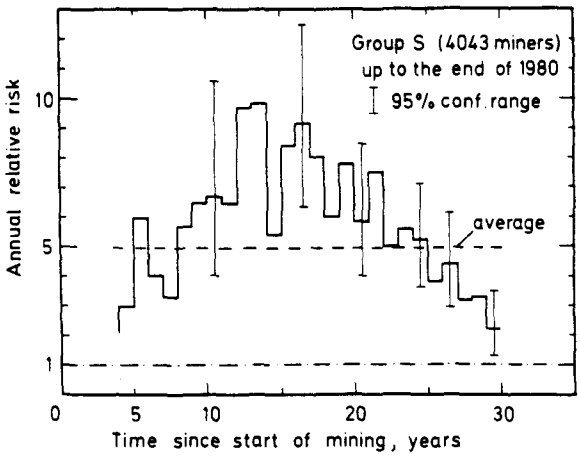

Fig. 6. Relative risk of lung cancer among Rn-exposed uranium miners (group S) in the CSSR as function of time after start of mining (from data of Sevc et al. 1988).

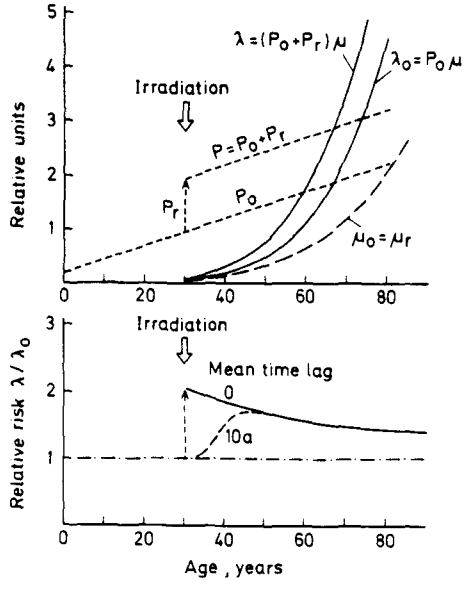

Fig. 7. Two-stage model for the age-specific lung cancer rate (schematically).

EQUIPMENT REQUIREMENTS FOR HIGH ENERGY ELECTRON ACCELERATORS USED IN RADIOTHERAPY FOR EFFECTIVE RADIATION PROTECTION.

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SYNOPSIS:

Use of high energy electron accelerators, for photon and electron therapy, is on the increase in radiotherapy departments (1). Many radiotherapists prefer to use not more than 15-20 MV Photons; but electrons of energy upto 45MeV are used (1,2). The dose rate commonly used is not less than 3Gy min-1 at the normal treatment distance or isocentre (3). Some dual mode accelerators have the capability to produce electron beam with doserate of even 1000 Gy min-1 (4). The presence of high energy and high doserate presents special problems for protection of patients, radiation workers and general public (5). Possible presence of neutrons, inevitable production of high energy photons when electron beam is used, induced and residual activity, noxious gases,capture gammas and annihilation radiation,possible radiation damage to equipment components/other materials and presence of microwave radiation are additional problems (4).

Safety of general public can be easily achieved by proper design and adequate shielding of irradiation areas, provision of door interlocks, radiation warning, and other technical and administrative measures (5). However, protection of patients and operating personnel puts stringent requirements for the accelerator itself. In radiotherapy, large quantity of radiation at high dose rate is deleberately given to man. Both the probablity of cure and chances of hazard to normal tissues is highly dependent on the accuracy of the radiation quantity, quality and treatment planning. Even slight variation in treatment parameters including under exposure is highly undesirable. Failsafe measures should therefore be provided in the equipment to obtain high accuracy and precision. Any chance for accidental exposure should be made impossible. Strict evaluation of equipment design and performance is therefore essential right from preparation of tender documents and should be ensured through acceptance testing and routine quality control procedures and radiation protection surveys (3,10).

This paper presents important equipment parameters relevant for protection in high energy medical electron accelerators. The requirements available in piecemeal in various national regulations, codes of practice and international recommendations or standards relating to radiation protection are compiled and grouped into five major categories as follows:

1. Provisions for accurate selection and delivery of intended dose as per treatment plan to the correct treatment volume.

2. Provisions for reduction of integral dose to patients from other than useful beam.

3. Provisions to avoid accidents and to deal with emergencies including power failure.

4. Radiation beam quality and accessories.

5. Other general requirements.

The current requirements (1,2,3,4,5,6,7,8,9,10,11) for each item under the above five categories including need for display are mentioned.

REMARKS/DISCUSSION:

1. The classification of items is given only for convenience but the requirements in various groups are interconnected.

2. The paper deals with only those parameters involving radiation hazards. But it should be remembered that stable and accurate performance of all other components is also essential for the proper functioning of the equipment and effective treatment to the patient.

3. Though the requirements can be found included elsewhere, the authors have segregated parameters relating to equipment and tried to introduce new recommendations in the light of the reported accident (9) in accelerator rooms.

4. It should be remembered that effectiveness of therapy is directly related to the accuracy of estimating the treatment volume and setting up of patient. The accuracy in these by means of simulators, computer tomography or other means is equally important as the correct functioning of the accelerator unit and delivery of dose.

5. The paper is intended as guidance for potential and current users not having fully developed radiation protection units.

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RADIATION ENVIRONMENT IN THE TUNNEL OF A HIGH-ENERGY PROTON ACCELERATOR AT ENERGIES NEAR 1 TeV

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tively. We do not fully understand the differences between peaks but point out that the primary interactions that do not occur on N_2 most likely occur either on H_2 or He in cryogenic sections or with machine materials.

Monte-Carlo simulations performed by Gabriel et al. (not shown here) are in good agreement with the spectrum of Fig. 5 derived from slopes [3]. They furthermore indicate that about 80% of the neutrons at 200 cm from the beam line are albedo neutrons, *i.e.*, scattered from tunnel walls. This was tested experimentally by repeating certain runs at 39 cm from the beam line (Position 2, Fig. 2). The result was consistent with the expected radial distribution for the direct fluence, assuming that the albedo fluence is uniform across the tunnel section.

Integration over the "slope" z-distribution of Fig. 4a and correcting for the albedo fluence gives a value 10 neutrons produced per 900-GeV proton passing A-17 and per g cm^{-2} of N_2 target. A similar calculation based on an integration over the intercept z-distribution (Fig. 4b) gives 5×10^{-9} neutrons produced per passing 900-GeV proton. We caution the reader that this latter value is not well understood, is subject to the vagueries of machine operation and will likely vary widely from place to place around a given accelerator ring.

The above observations suggest a common "filter" for the neutrons, regardless of the nature of the original interactions which produce the parent cascade. Prominent parts of the filter must be the iron magnet yokes as well as the concrete tunnel lining. The neutron field dominates the tunnel radiation field in terms of absorbed dose to tissue. Because of their capability of producing lattice defects and transmutations, neutrons of energy $E_n \geq 150$ keV are the most important potential cause of radiation damage to solid state electronic devices in accelerator tunnel environments similar to those studied [1].

This work was partially supported by the U. S. Department of Energy under Contract No. DE-AC03-76SF00098 with the University of California.

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RECONSTRUCTING FALLOUT EXPOSURES TO THE U.S. POPULATION FROM WEAPONS TESTING IN NEVADA DURING THE 1950's

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ABSTRACT

Techniques used to reconstruct fallout depositions which included re-analyses of gummed-film and other monitoring data, retrospective soil and sediment sampling, and analyses of recent airborne gamma spectrometric survey data to locate fallout "hotspots" are discussed. Based on the resulting estimates of cumulative fallout deposition, external radiation exposure probably resulted in whole-body doses on the order of only 1 mSv to most of the U.S. population.

INTRODUCTION

During the years 1951 to 1958, over 100 above-ground weapons tests were carried out at the U.S. government's Nevada Test Site (NTS), located ~150 km NW of Las Vegas, Nevada. In recent years, considerable public concern has surfaced regarding possible radiation exposures to the population of states downwind from the NTS. This concern has manifested itself in a number of lawsuits filed against the U.S government. In response, a number of major reassessment efforts were begun, starting in 1979. These include: the U.S. Department of Energy-sponsored Offsite Radiation Exposure Review Project (ORERP), whose goal is to reconstruct both external and internal radiation exposures to the populations of states downwind from the NTS (Church et al. 1987); a Congressionally-mandated U.S. National Cancer Institute (NCI) study to estimate the thyroid exposure of the entire U.S. population to ^{131}I from NTS tests (Wachholz 1987; Bouville et al. 1987); and two NCI sponsored epidemiology studies being carried out by the University of Utah attempting to correlate the incidence of leukemia and thyroid cancer in certain downwind populations with exposure to NTS fallout (Rallison and Lotz 1987; Lloyd et al. 1987).

All of these studies have one common requirement, namely, estimates of fallout deposition from each weapons test. These deposition estimates are then incorporated into various food-chain models, structure-shielding models, etc., to estimate either external population doses or doses to particular organs. Most of the deposition estimates have been inferred from one of three major data sources: a) extensive survey meter monitoring in the region within a few hundred km of the NTS after each shot, b) estimates of daily fallout deposition at about 100 sites monitored throughout the U.S. using gummed-film, and c) retrospective estimates of cumulative fallout deposition from all NTS detonations inferred from residual amounts of ^{137}Cs and plutonium present in soil samples from undisturbed sites. Since the latter two resulted from the efforts of the Environmental Measurements Laboratory (EML) and represent the only comprehensive sources of actual data on deposition beyond the

immediate downwind vicinity of the NTS, we have been closely involved, either directly or in an advisory capacity, in all the reconstruction efforts mentioned above. In addition, we have also developed and utilized several other innovative methods to supplement and corroborate the soil sampling, survey meter, and gummed-film data.

METHODS

In order to corroborate the deposition estimates, and corresponding radiation exposures inferred from survey meter monitoring, as well as to extend the estimates further downwind to areas where no monitoring was done, we developed a method of estimating cumulative fallout deposition from measurements of ^{137}Cs and plutonium isotopes in undisturbed soils (Beck and Krey 1983). The NTS ^{137}Cs contribution is distinguished from the generally greater "global" fallout contribution from the large thermonuclear explosions, conducted external to the continental U.S. during the early 1960's, by the greatly different ratio of $^{240}Pu/^{239}Pu$ for the two sources. Using this technique, we corroborated earlier fallout exposure estimates for close-in sites, and also showed that significant deposition occurred in areas as far away as the Salt Lake valley in northern Utah (Beck and Krey 1983). This finding led the ORERP to extend the scope of their study from the area immediately downwind from the NTS to as far east as western Colorado and northern New Mexico, utilizing the EML soil analysis technique.

The retrospective soil analysis technique, however, provides information on the cumulative NTS fallout deposition only. Its sensitivity is also limited due to the large "global" fallout "background" signal. Since most of the reconstruction efforts required more detailed information on exactly when fallout occurred at a given site as input to their dose pathway models, it was necessary to utilize the only other source of extensive data on fallout during this period, namely, the results of the EML gummed-film fallout monitoring network. Originally, these measurements of total beta activity deposited on gummed-film were intended only to give a gross indication of where significant fallout occurred after each shot. Quantitative deposition estimates based on these data were highly suspect due to uncertainties in collection efficiency, calibration, and other factors. An extensive re-analysis of the original data, however, has resulted in much more accurate estimates of individual radionuclide deposits from individual shots on a daily basis (Beck 1984; Beck et al. 1987). As a result, these gummed-film data are now being utilized as a prime source of deposition estimates for regions outside the area of survey meter monitoring in all the ongoing reconstruction studies.

Both the gummed-film deposition estimates and retrospective soil estimates have been shown to provide comparable results where data are available from both methods. Additional corroboration of both the temporal variation and the cumulative deposition has also been obtained for a few areas by examining the activity of ^{137}Cs and Pu as a function of depth in sediment cores taken from reservoirs and/or lakes in areas downwind from the NTS. The activity and isotopic composition in the various sediment segments can be related

to deposits on the watershed occurring during particular time intervals, utilizing certain fiducial time markers. Comparison of the total inventory in the core with that on the shoreline then allows an independent estimate to be made of the average NTS and global fallout depositions over the watershed. The results obtained for two such watersheds in the state of Utah tended to corroborate the NTS fallout depositions inferred by other methods, providing an additional source of confidence in our total deposition estimates (Krey et al. 1987).

The retrospective soil sampling technique described earlier, while powerful in its ability to distinguish the source of the soil activity, is not an efficient method for surveying large areas. The total number of sites sampled by either EML or ORERP was only about 200, mostly in the more populated towns and cities. The survey meter monitoring which was carried out during the testing period was also limited in geographical extent. Thus concern continued to be expressed regarding the possibility of isolated undetected "hotspots" having occurred, perhaps as a result of washout by thundershowers or other localized meteorological phenomena. An additional retrospective survey technique was therefore utilized to search for these "potential hotspots". Fortunately, during the 1970's most of the U.S. had been systematically surveyed for possible sources of uranium ore by the National Uranium Resource Exploration Project (NURE), utilizing airborne gamma spectrometry. By carefully reexamining the gamma spectra so obtained over regions downwind from the NTS for indications of extraordinary high fluxes of 662 kev ^{137}Cs gamma rays, i.e., well above that which would be expected in that area from global fallout ^{137}Cs soil inventories, we were able to conclude that no NTS fallout "hotspots" of any significant extent or magnitude were likely to have occurred (Beck 1983).

In addition to the major sources of information on deposition discussed above, other less extensive sources also were utilized to supplement and confirm the data input into the various reconstruction programs. These sources included results of reanalyses of soil samples collected in various areas during the 1950's, meteorological modeling of cloud tracks for particular shots (Cederwall and Peterson 1987; Hoecker and Machta 1987), and data on fallout monitored in air and rain by the U.S. Public Health Service beginning in 1956.

DISCUSSION

Except for areas just outside the NTS, the cumulative depositions of fallout from Nevada weapons tests, and the resultant radiation exposures to populations, were quite small. Although detailed results for particular populations and pathways must await the publication of the final reports for the various reconstruction programs, the rough geographic variation in cumulative fallout deposition and resultant doses from external radiation exposure can be estimated from the gummed-film network data (Beck et al. 1987). For persons living in the eastern and midwestern U.S., cumulative whole body and bone marrow doses were on the order of 1 mSv, or of the same order as the external radiation dose ascribed to "global" fallout in these areas (UNSCEAR 1982), and also about the same

order as that received by this population in a single year from natural background exposure. Corresponding cumulative ^{131}I depositions ranged from about 50,000 to 150,000 Bq m^{-2}. Even in states just downwind from the NTS, fallout depositions, and resulting external exposures, were only about 3 to 4 times higher, although some small towns just downwind from the NTS received fallout and hence doses about an order of magnitude higher (Beck and Krey 1983).

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DOSE TO MAN FROM THE CONSUMPTION OF MARINE SEAFOODS: A COMPARISON OF THE NATURALLY-OCCURRING ^{210}Po WITH ARTIFICIALLY-PRODUCED RADIONUCLIDES

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INTRODUCTION

The consumption of above-average quantities of marine seafoods has for a long time been a characteristic of critical groups associated with the authorized discharge of low-level liquid radioactive wastes into UK coastal waters. The concentrations of artificially-derived radionuclides in these seafoods are routinely monitored and the committed effective dose equivalent calculated for the purposes of radiological protection.

There is an increasing interest in the epidemiological aspects of risk and dose associated with certain populations in the vicinity of nuclear establishments, but little attempt is made to consider the total dose which might be received by such groups. A contribution to such a dose would clearly be that of the naturally-occurring radionuclides in the food consumed. In this respect it is of interest to consider the doses which could arise from naturally-occurring radionuclides present in seafoods: such nuclides include those of uranium, thorium and radium, but of particular interest is ^{210}Po. The average dose from dietary intake of ^{210}Po is considered by UNSCEAR (1982) to be of the order of 0.13 mSv a^{-1}. This paper briefly examines the concentrations of ^{210}Po in some UK marine seafoods, and estimates the dose to high seafood consumers compared with the the doses which such consumers in the Cumbrian coastal area are estimated to receive from radionuclides discharged under authorization by the British Nuclear Fuels (BNFL) reprocessing plant at Sellafield.

^{210}Po CONCENTRATIONS IN SEAFOODS

It has long been known that the tissues of certain marine organisms contain high concentrations of ^{210}Po, in particular the hepatopancreas of some shrimps and other small crustaceans (Heyraud and Cherry, 1979; Cherry and Heyraud, 1981) and the pyloric caeca of tuna (Hoffman et al., 1974). Of greater value in estimating dose to man are some recent data on the concentrations of ^{210}Po in the soft tissues of mussels collected around the UK coast; these indicated a range of 111 to 279 Bq kg^{-1} dry weight (McDonald et al., 1986).

In order to broaden the range of data available, particularly with regard to species eaten by man, a few samples of shellfish were obtained from the south-west coast of the UK and analysed for ^{210}Po. The duplicate samples consisted of bulked 'white' meat (flesh) and 'brown' meat (hepatopancreas and gonad) of the crab,

Cancer pagurus, and the flesh of the mussel, Mytilus edulis, and of the winkle, Littorina littorea. The results are given in Table 1. Previous analyses of the flesh of individual fish had

Table 1 Concentrations of ^{210}Po (Bq kg^{-1} wet)

Crab (White meat)	Crab (Brown meat)	Mussels (Flesh)	Winkles (Flesh)
1.53	50	19	7.7
1.33	45	18	7.5

resulted in the following mean values: plaice (Pleuronectes platessa) 1.52 Bq kg^{-1} wet; mackerel (Scomber scombrus) 1.70 Bq kg^{-1} wet; and cod (Gadus morhua) 1.27 Bq kg^{-1} wet (Pentreath et al., 1979).

DOSE TO HIGH SEAFOOD CONSUMERS

The estimated dose received from the consumption of seafoods obtained in the vicinity of nuclear establishments in the UK is kept under annual review by MAFF. The critical group approach is used for control purposes, and the techniques used in the collection of data in relation to consumers on the Cumbrian coast have been described by Leonard et al. (1982) and Leonard (1984).

Table 2 Seafood consumption rates and dose received by consumers in the vicinity of Sellafield in 1986 (Hunt, 1987)

Population	Annual consumption rate (kg a^{-1})		Committed effective dose equivalent on the basis of different gut transfer factor values (f_1) for Pu and Am (mSv a^{-1})	
			f_1 0.0001	f_1 0.0005
Consumers in local fishing community	Fish	36.5	0.12	0.34
	Crustaceans	6.6		
	Molluscs	6.6		
Consumers associated with commercial fisheries	Fish	82	0.10	0.18
	Crustaceans	18		
	Molluscs	15		
Typical member of the fish-eating public	Fish	15	0.009	0.009

For the 1986 assessment (Hunt, 1987), as in previous years, two high-consumer groups were considered: a local fishing community eating seafood taken fairly close to Sellafield, and a larger population associated with commercial fisheries at nearby fishing ports. The former tend to eat plaice and cod, crab and lobster, plus mussels, cockles, limpets and, particularly, winkles (Leonard and Hunt, 1985). Annual consumption rates by these two groups in 1986 are given in Table 2, together with the fish consumption rate of a typical local member of the public (Hunt, 1987). The committed effective dose equivalents for these groups, resulting from Sellafield-derived nuclides, are also given in Table 2. Two values are given, the difference being the result of using generic (0.0005) and realistic (0.0001) values for the gut transfer factor for Pu and Am nuclides present in molluscan shellfish and ingested by man.

Although the seafoods analysed for ^{210}Po were collected at a different site, it is of interest to estimate the dose which could be received at similar rates of consumption to those given in Table 2. In fact the ^{210}Po values for mussels are, on a dry weight basis (121 and 126 Bq kg^{-1}), lower than those for mussels (279 Bq kg^{-1} dry) collected at Ravenglass near Sellafield as cited by McDonald et al. (1986).

Table 3 Estimated dose to man from ^{210}Po in certain seafoods

Assumed consumption rate (kg a^{-1})		Assumed diet	Estimated committed effective dose equivalent (mSv a^{-1})
Fish	36.5	Cod, plaice)	
Crustaceans	6.6	Crab white + brown)	0.13
Molluscs	6.6	Mussels + winkles)	
		Cod, plaice)	
		Crab, brown only)	0.18
		Winkles)	
		Cod, plaice)	
		Crab, white only)	0.02
		Winkles)	
Fish	82	Cod, plaice)	
Crustaceans	18	Crab white + brown)	0.33
Molluscs	15	Mussels + winkles)	
		Cod, plaice)	
		Crab, brown only)	0.50
		Mussels + winkles)	
		Cod, plaice)	
		Crab, white only)	0.15
		Mussels + winkles)	
Fish	15	Cod, plaice, mackerel	0.01

Because of the large differences in ^{210}Po concentration between brown and white crab meat, a number of different dietary combinations were considered. A value of 4.3 10^{-7} Sv Bq^{-1} was used to calculate dose per unit intake (NRPB, 1987). The results are given in Table 3.

DISCUSSION

It would appear from a comparison of the data in Tables 2 and 3 that the dose which could be received from ^{210}Po by high seafood consumers is of the same order as that received by the critical groups from the Sellafield discharges. Indeed the range is considerably greater, and for the highest consumers ^{210}Po is likely to be the principal contributor regardless of the source. Further studies are being made to examine the concentrations of ^{210}Po in other seafoods, and to assess their variation in relation to season and location around the UK coast.

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CONCENTRATION FACTORS OF STABLE ELEMENTS AND RADIONUCLIDES IN PO RIVER FISH

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INTRODUCTION

The concentration factors (CF) of stable Co, Cs, Mn, Fe, Zn and Sr in different fish from six stretches in the middle course of the Po river (N. Italy) have been investigated. The space-time variation in water has been followed for 14 months. The investigation has been undertaken to study CF variations in the same fish species as a function of the physico-chemical form of the different elements in water (dissolved, dissolved and exchangeable fraction of the particulate, total). CF values of ^{103}Ru, ^{131}I and $^{134-137}Cs$ were also investigated for **Cyprinus carpio** reared, with artificial food, in two semi-natural environments.

EXPERIMENTAL

Trace elements concentration in water was determined by ICP-AES or AAS either directly (Sr) or after enrichment with NH_4 form chelex-100[1] (Mn,Fe,Zn,Co) or copper ferrocyanide[2](Cs). Radionuclides were determined by gamma-ray spectrometry (Ge-Li detector) after preconcentration on NCFC columns ^{137}Cs or on ion exchange resins. The exchangeable fraction and the total content of elements in the particulate (>0.4 μm) was determined after leaching with 0.3 M HCl, or total acid digestion[3]. Trace elements in fish were determined by ICP-AES (Fe, Mn, Zn, Sr) or AAS (Co, Cs) following wet digestion with HNO_3-HF-$HClO_4$ mixture. Cs was separated before analysis, with copper ferrocyanide. Radionuclides were determined by direct gamma-ray spectrometry.

CF FROM FIELD OBSERVATIONS

Po river water (<0.4 μm) quality and trace elements content are reported in tab. 1. The analyzed species and their subdivision in different diets[4] are : **Planctivorus** = Alburnus alburnus alborella (sample size=7): **Carnivorus** = Barbus barbus plebejus(8), Cyprinus carpio(1), Perca fluviatilis(4), Lepomis gibbosus(5), Ictalurus melas(3), Tinca tinca(1), Rutilus rubilio(4); **Omnivorus** = Carassius carassius(1), Leuciscus cephalus cabeda(11), Chondrostoma sp.(1), Chondrostoma toxostoma(9), Chondrostoma soetta(9), Gobio gobio(3), Scardinius erythrophthalmus (1), Padogobius martensi(1); **Predator** = Esox lucius (1), Micropterus salmoides(3).

Notwithstanding a space-time variability of some stable element concentrations in water (Co, Zn, Sr) was observed (analysis of variance; $P<0.05$), the geometric mean on the all samples and

for each element was used to compute CF.

CF values (wet weight) were computed for each species and trace element both for edible and not edible part. The Duncan's test showed differences among species (for the same element) of little significance; for this reason CF values were analysed only considering the feeding habits (tab. 2). Duncan's test on these data showed difference only for Zn, ^{137}Cs and Fe (edible part); higher values for planctivorus (Zn and Fe) and both planctivorus and predator (^{137}Cs) were obtained. For planctivorus fish (Alburnus a.a.) the data refers to the total sample. The Duncan's test was also applied to search differences between edible and not edible parts for each element. In this case differences were observed only for Sr (omnivorus and carnivorus) and Mn (omnivorus). CF values for ^{137}Cs are generally lower than those for stable Cs, thus suggesting the occurrence of a non equilibrium condition. CF values (edible part) were also evaluated (tab.3) according to the different physico-chemical forms of each element in water (dissolved: <0.4 μm; dissolved plus exchangeable fraction of the particulate; total). Only for Sr quite similar values were obtained; for the other elements higher CF values are obtained when the dissolved elements concentrations are considered. The latter confirms the difficulty to compare experimental and literature data, also considering that, for some elements (e.g. Cs and Sr), CF are greatly depending on the water and sediment quality (e.g. trophic state, K and Ca concentration etc.)[5].

CF FOR RADIONUCLIDES IN **CYPRINUS CARPIO** AFTER CHERNOBYL ACCIDENT

Cyprinus carpio were periodically collected after Chernobyl accident. Space-time variability of ^{103}Ru, ^{131}I and ^{137}Cs concentrations in edible and not edible parts as well as those in water (dissolved: <0.4 μm, solid phase: >0.4 μm) are shown in fig. 1 for ^{137}Cs. Only fish and water samples collected after May 12th. 1986 were considered in computing CF (quasi-steady state condition of radionuclides concentration in water). Geometric mean of CF data was computed,as suggested by De Bortoli et al.[6], for two different period of time. In the first period (30 days) equilibrium was not reached yet and ^{137}Cs CF is only due to the direct contamination from water; the obtained value (CF=29) is in good agreement with homogeneous literature data[7]. Also in the second period (90 days), equilibrium was not yet completely reached, but the value for ^{137}Cs (CF=74) include contamination both from water and food[7]. CF values for ^{103}Ru and ^{131}I, computed for the same period of time, agree fairly well with literature data [8] and, as the case of ^{137}Cs, point out the occurrence of a non equilibrium condition.

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Tab. 1 : Po river water (<0.4µm) quality and trace elements content

quality parameters	min	max	trace elements	sample size	geometric mean	σ	min	max
O_2 (ppm)	7.5	10.5	Co (ppb)	28	0.099	3.03	0.022	0.95
T(°C)	7.7	20.9	Cs (")	20	0.010	1.69	0.004	0.027
cond.(µScm^{-1})	74	437	Mn (")	26	15.2	2.11	5.46	54.0
pH	7.02	8.25	Fe (")	28	4.15	2.48	1.17	4.01
T.O.C. (ppm)	2.2	3.7	Zn (")	28	4.36	1.94	1.67	22.5
Cl^- (")	3.0	17.6	Sr (")	28	339	1.32	238	591
SO_4 (")	23.0	71.6						
PO_4 (")	0.011	0.228	^{137}Cs (Bq/m^3)	22	14.5	1.77	7.5	43.7
NO_3 (")	4.4	18.4						
Na (")	3.7	20.5						
K (")	1.6	4.7						
Ca (")	32.5	67.8						
Mg (")	6.0	16.7						
NH_4 (")	0.02	0.69						

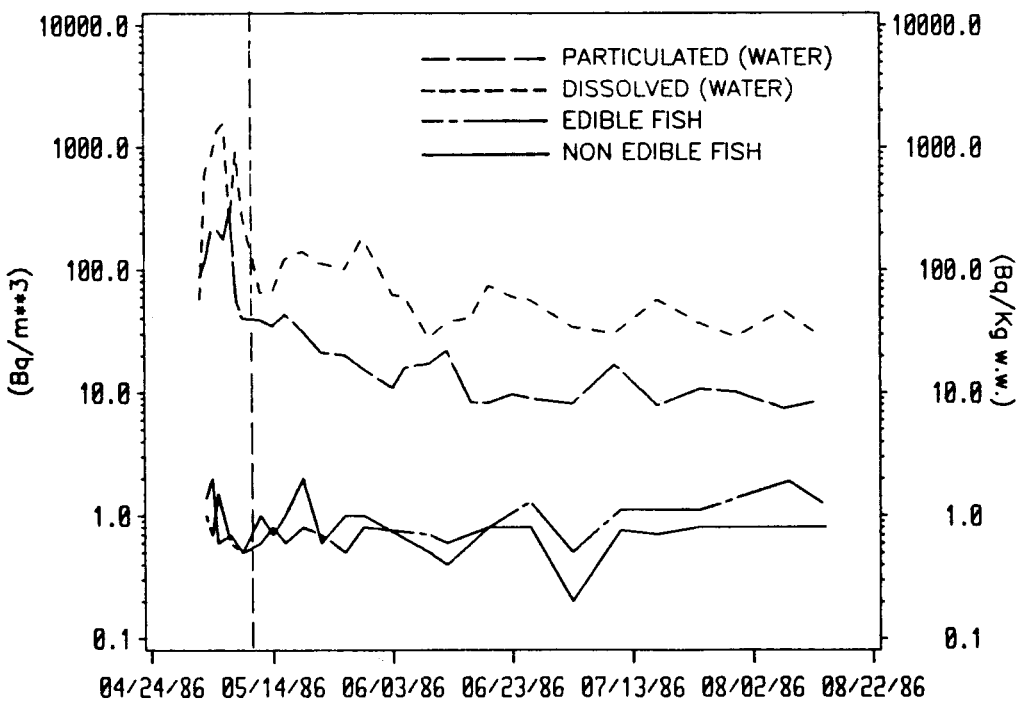

Fig. 1: Temporal variations of the ^{137}Cs concentration in Cyprinus carpio and in the Po river water

Tab. 2 - CF values (wet weight) for different feeding habits and elements

Element	PLANCTIVORUS		CARNIVORUS				OMNIVORUS				PREDATOR			
	TOTAL		EDIBLE		NOT EDIBLE		EDIBLE		NOT EDIBLE		EDIBLE		NOT EDIBLE	
	n.	mean	n.	mean	n.	mean	n.	mean	n.	mean	n.	mean	n.	mean
Co	7	2420	23	3840	23	4140	33	3840	30	3840	4	4240	2	2320
Cs	7	1000	24	680	19	970	32	900	30	760	4	330	2	980
Mn	7	370	26	185	23	360	32	250	30	570	4	60	2	170
Fe	7	10390	27	3950	24	6070	36	3930	33	6190	4	2320	2	3570
Sr	7	69	27	47	24	121	36	57	33	160	4	24	2	110
Zn	7	14700	27	3810	24	5410	36	4771	33	8170	4	2150	2	6580
^{137}Cs	3	2050	10	360	9	310	12	510	9	521	1	(4870)	1	3060

Tab. 3 - CF values (wet weight) for feeding habits with respect to different physico-chemical form of each element in water. 1 (dissolved), 2 (dissolved plus exchangeable fraction of the particulate matter), 3 (total).

Element	PLANCTIVORUS			CARNIVORUS			OMNIVORUS			PREDATOR		
	1	2	3	1	2	3	1	2	3	1	2	3
Co	2420	510	330	3840	810	520	3840	810	520	4240	890	380
Cs	1000	530	38	680	360	26	900	470	35	330	170	13
Mn	370	88	71	185	44	35	250	61	40	60	14	11
Fe	10390	140	33	3950	53	13	3930	52	13	2320	31	7
Sr	69	62	62	47	42	42	57	51	51	24	22	22
Zn	14700	2850	2110	3810	1020	760	4771	1280	950	2150	580	430
^{137}Cs	2050	410	-	360	73	-	510	100	-	4870	970	-

EVOLUTION AND TRENDS OF THE CONCEPT OF DOSE LIMITS

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INTRODUCTION

The purpose of this presentation is to review the evolution of an important concept of radiation protection, discussing the change of emphasis or even meaning involved in its use. The concept discussed is that of the limits and the extension of their use to situations not covered in the past. While the author is involved in the work of the ICRP, it should be noted that, except when they are straight quotations, the materials presented in this paper do not represent necessarily the formal position of the ICRP.

DOSE LIMITS

The dose limits are at present only one component of the radiation protection recommendations. However, they are usually judged to have great importance, and the fact that the ICRP recommended values have been widely accepted has resulted in a global uniformity that is very positive for radiation protection.

Limits in the past

The real meaning of the limits has changed basically with the improving knowledge of radiobiology. Radiation protection has evolved enormously since the early concepts of "tolerable" levels of radiation that were analogous to toxicological ideas of thresholds and safety factors. In the old days, protection was considered to be absolute, provided that the technical means used to control and restrict the radiation levels did not fail.

This approach was very reasonable when only what is now called "non-stochastic effects" of radiation were known. However, in spite of advances in radiobiological knowledge, this approach persisted, even if qualified by some statements. A far-reaching consequence of the threshold and safety factor concept was the tendency to interpret radiation protection solely as an "individual-related" activity, a tendency which persisted long after non-threshold effects were beginning to be taken into account. With this perspective, the relevant protection parameter was deemed to be the total dose to individuals, irrespective of the sources of origin.

Conceptual basis of present dose limitation

The identification and mainly the quantification of what is called now the "stochastic effects" of radiation caused a continuing change of attitude in radiation protection, which crystallized in the recommendation contained in ICRP Publication 26. It is not the purpose of this paper to describe in detail what is prescribed in the recommendation, but to stress the conceptual changes.

The role of the limit is to provide a minimum level of protection. The dose limits recommended by the ICRP are not intended, as they were in the past, to be design or planning values, but the lower boundary of a non-acceptable region of values. Values above the limits are specifically not permitted, but values below the limits are not automatically permitted. In this sense, the limits are basically constraints for the optimization procedures.

However, the use of limits as constraints for optimization presents some conceptual difficulties, especially in the case of exposures of members of the public. The limit is an individual-related requirement, while optimization is a source-related requirement. Because the limits apply to the combined exposure from many sources, they cannot be used to restrict a given single source even when optimization would allow it. In fact, exposures at the limit from one single source would leave no margin for other practices exposing the same critical group. The problem of overlap of exposures from different practices is not restricted to any given instant of time. Each year of operation of a continuing practice can cause exposures which would be delivered in the future and which would add to the contribution of other years of operation in the practice.

It is possible to control the combined future average effective dose equivalent from all practices by applying practical limits to the dose commitment per unit practice for each of the practices. This concept has been used in some countries to set constraints to the collective effective dose equivalente commitment per MW year of electrical energy produced by nuclear means.

Upper bounds

For the purpose of constraining optimization of the protection of specific practices or sources, it seems reasonable that national authorities select "dose upper bounds" which are only fractions of the dose limit, allowing for the exposure overlap from all practices, and even reserving some margin for unforseen but justified sources in the future, which otherwise could be precluded.

Upper bounds are the real basic constraints for the optimization of radiation protection of specific practices or sources. An upper bound of dose corresponds to an upper bound of the radiation risk due to a specific practice. It is therefore possible to select upper bounds for given practices by comparison with what is considered acceptable by society for other types of risks, either of similar practices or of practices giving similarly beneficial products.

The dose limits recommended for the case of occupational exposures are typical group upper bounds for a situation where many different exposing radiation sources can exist. In spite of their historical development, they really are at present source-related group upper bounds. This flavour is strongly reflected in the ICRP policy underlying the use of committed dose equivalent.

In a statement the Commission confirms that its policy is to limit the risk committed by each year of operation, no credit being taken for earlier years, if they have committed lower risks or for future years in the expectation of improved conditions of exposure.

It is also clear that the purpose is to limit the risk committed in a given period, and not the time distribution of the expression of the risk. The same is true when considering continuing practices, where the limitation applies to the dose commitment from a year of practice (and therefore to the maximum per caput annual dose in the future), thus limiting the risk committed by one year of practice.

Limits in the near future

It is likely that no revolutionary changes will occur with the limit concept. Limits will be presented as combined upper bounds, having the aim of restricting the risk committed during one year of work or practice. The ensuing risk expression will be distributed in time even in the case of pure external exposures, existing a double time distribution for the case of intakes of radionucleides and a triple time distribution in the case of releases into the environment.

It is also likely that there will be more emphasis on upper bounds and group upper bounds for specified limited groups of sources. The value of the limits will probably depend on radiological knowledge accumulated during the development of new recommendations, but the criterion of comparison of risks with safe industry and with hazards of everyday's life are likely to be maintained.

LIMITATION OF RISK IN THE CASE OF PROBABILISTIC EXPOSURES

The system of dose limitation recommended by the ICRP gives no guidance on design criteria for features reducing the probability of disruptive events leading to radiation exposures. However, the underlying philosophy of the system could be used for such purpose.

This possibility stems from the basic concept that the limits really restrict the risk due to a year of work or practice. With a random accidental event, the risk depends on the probability of the event and on the resulting dose in the individuals receiving the highest dose. 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 events 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 all P_2 are proportional to dose at the lower doses (where only stochastic effects are possible), but increase steeply with whole-body dose at values above 2 Gy due to acute non-stochastic effects. Probability values must be assigned to the disruptive events that are expected to cause significant exposures if they occur. Event-tree assessments will yield probability values P_{1i} due to the various potential events.

There is no inherent reason to limit the risk due to probabilistic events to the same level as that implied by the dose upper bound set for a particular source, specially if such an upper bound was established specifically for normal operations. Nevertheless, if it is assumed that the purpose is to restrict the risk to that order of magnitude, for members of the public it should correspond to annual values of 10^{-6} to 10^{-5}. For example, selecting an annual value of 10^{-6}, the design of safety features should be such that

$$\sum_i P_{1i} P_{2i} \leq 10^{-6}$$

Some countries use this type of probabilistic criterion for other applications, such as in nuclear safety assessments. The main use, however, is expected to be in the design of safety features involved in the use of large radiation sources.

CONCEPT AND VALIDATION STUDIES OF THE REAL-TIME REACTOR-ACCIDENT CONSEQUENCES ASSESSMENT MODEL "ECOSYS"

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ABSTRACT

The Chernobyl accident has demonstrated the urgent need for computer programs for real-time assessment of potential radiological consequences of major reactor accidents and for timely recommendations of useful and cost-efficient countermeasures. During the past decade the dynamic radioecological program "ECOSYS" has been developed by us for nuclear accident consequence assessment with high resolution in space, time and exposure pathways. The Chernobyl reactor accident leading to relatively high contamination of Southern Germany provided excellent conditions for realistic validation studies of concept, sub-models and parameters of ECOSYS. More than 10 000 low level and in-situ gamma spectroscopy measurements were performed to study the behaviour of radionuclides in food chains and in the urban environment and to compare the results to theoretical predictions of ECOSYS. The results show good agreement in the contamination levels of important food stuffs and in external exposure dose rates from a given surface contamination. Improvements were necessary in the assumptions regarding the food consumption habits - which changed considerably - and in the functions describing the weathering off from urban and plant surfaces. Some results of this validation study and the concept of the improved computerized model, which is being converted into a real-time code, are discussed in this paper.

INTRODUCTION

For the last ten years we have been developing a detailed time-dependent radioecological model (ECOSYS) for the realistic assessment of present and the prediction of future exposures of members of the public after major accidental releases of radionuclides to the atmosphere from nuclear facilities. This model has been and is being used in generic probabilistic risk assessment studies for reprocessing plants, deep underground waste storage sites, and for nuclear power plants. The reactor accident of Chernobyl forced us to use this model for the first time in a given actual situation of large scale environmental contamination (initially around 0.5 MBq/m² in Bavaria (1)) for dose assessment and analysis of the efficiency of mitigating countermeasures. On the other hand this situation gave us also the unique possibility to test the validity of the predictions of this model. More than 10 000 laboratory and in situ gamma spectroscopy measurements have been performed since May 86 for this purpose. This paper reports some of our experiences made in this validation study and draws some conclusions as to the design of adequate real time reactor accident consequences assessment programs; further results can be found in ref. (2-8).

THE ECOSYS MODEL

ECOSYS is a time-dependent radioecological model for the prognostic assessment of radiation exposures of members of the public after accidental releases of radionuclides into the atmosphere. It takes into account external radiation from a passing cloud and from deposited radionuclides (including shielding by houses) and internal exposure from inhalation and ingestion. The food chain is modelled in great detail in temporal aspects (considering time in the year as well as time after the accident) and taking into consideration normal agricultural and food processing practices. The program does not account for atmospheric dispersion (i.e. input from atmospheric transport codes is needed) and for aquatic exposure pathways.

EXTERNAL EXPOSURE

From the specific activities of all deposited radionuclides a prognosis of the external gamma exposure of members of the public was made. To check its validity about 250 TLD-dosemeters were issued to members of the institute and their families. These detectors were either used as personal dosemeter or deposited at home at and below the window (area dosemeters; to account for shielding). Although the results (9) have to be interpreted on an individual base, the mean values agree remarkably well with the calculation assuming a 80 percent stay in houses and a mean shielding factor of 10.

It could be shown that at early times after aerosol deposition by heavy rainfalls initial run-off on paved areas leads to a gamma exposure reduction factor of about 0.7 as compared to grass land which in turn showed a shielding by a faktor of 0.7 due to surface roughness and initial migration of the radionuclides in soil. Later occuring weathering off effects on these paved places, streets, etc. are efficient in removing e.g. Cesium much faster than the slow migration into deeper soil layers in grass land. Thus we found dose rates above grass land to be by a factor of 4-5 higher than over paved areas one year after the same initial deposition. Such differences must be accounted for in assessments of external exposures.

INTERNAL EXPOSURE

The prognostic assessments made by ECOSYS for the nuclide specific concentrations as a function of time in various food products agreed very well with the measurements if the local deposition values were known (2). However, whole body counter measurements (4) indicated internal burdens of Cesium which were only one third of those predicted under the assumption of unchanged food consumption habits and of no influx of food from areas with less initial contamination. The actual extent of this factor and of changes in consumption habits are difficult to predict in a generic way.

CONCLUSIONS

In general the comparison of ECOSYS with experimental data showed very satisfying agreement. Therefore it can now be used as the basis of an early warning system in cases of elevated environmental radioactivity to optimize

countermeasures and to assess potential radiation exposures with high resolution in space and time. It needs nuclide specific contamination data for air and rain as input data. Complicated and very time consuming surveillance measuremets in agricultural products can then be restricted to the most relevant foodstuffs and locations. If relative radionuclide contaminations are known for some stations, simple gamma dose rate measurements and information on local precipitation at different stations can be used by ECOSYS to make reasonable estimates of local future radiation exposures on all relevant exposure pathways.

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ADDCOR: An Atmospheric Dispersion and Dosimetry Code for Operators and Regulators of Nuclear Facilities

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Introduction

A general purpose atmospheric transport, dispersion and dosimetry code is described. The code is intended for use by operators or regulators charged with the evaluation of the impact on individuals of releases of radioactivity to the atmosphere. The code is especially applicable to the normal atmospheric releases from a nuclear establishment with a diverse and varying range of activities. The code provides a ready means for estimating the degree of compliance of radiation doses received by exposed workers and members of the public with the recommended dose limits of the International Commission on Radiological Protection (ICRP). Such applications of this and similar codes are essential where effective dose rates are low and where committed effective doses must be evaluated. The code can also be applied to certain accidental releases. The code has its origins in the desire by the Australian Nuclear Science and Technology Organisation (ANSTO), formerly the Australian Atomic Energy Commission, to have an up-dated means of monitoring compliance of airborne releases of radioactivity from the Lucas Heights Research Laboratories (LHRL) with Regulations(1959) made under the NSW Radioactive Substances Act(1957) and the current recommendations of the ICRP(1977).

Sources, Pathways and Individual Exposures

There are many potential pathways by which activity released into the atmosphere may give rise to doses to individuals. The importance of each pathway depends in part upon the nuclide involved. The Figure shows the major pathways included in the code by which releases of airborne radioactivity can lead to the exposure of individuals. Major emphasis in this code is given to the inhalation and external pathways. For normal releases of airborne activity from nuclear establishments the radioactive rare gases ^{41}A, ^{85}Kr and ^{133}Xe for example, are most important. Other less volatile airborne radioactivity usually has various forms and levels of filtration applied to reduce the releases to minimal levels.

Features of the Code

The properties of individual sources of airborne activity including location, the nuclides released, the release times with respect to both time of day, weekday or weekend and quarter (Jan-Mar etc), and the duration of each release are taken into account. These data may be either source monitoring data or other estimates. Present applications involve some 10 sources and 30 nuclides. Both can be varied to suit a particular application.

Meteorological data used in the code is expressed as an array giving the frequency of occurrence of winds in each of sixteen sectors centred on N, NNE, NE etc, five wind speed groups, 0-1, 1-2, 2-4, 4-8 and > 8 m s^{-1} and seven Pasquill-Gifford stability classes, A through G, for each of three source heights to fifty metres, for each hour of the day and for each quarter.

The Gaussian atmospheric transport and dispersion model in sector averaged form is used to evaluate the airborne concentration of activity downwind from a source (Hanna et al ,1981).

The code presently uses the Briggs rural dispersion parameters (see Gifford, 1975). These data can be readily altered to suit other circumstances. The airborne concentration for all source-receptor pairs for each hour of the day and for each quarter is obtained by averaging over the appropriate components of the frequency distribution of the meteorological data.

As shown schematically in the Figure, four contaminated media are considerd in the code which lead to five possible exposure modes. Data required for the different dose conversion factors are discussed below. Data for biophysical transport of activity from the activity deposited on the ground through the ingestion pathways uses dynamic rather than steady state data (ICRP 1979, IAEA 1982).

The behaviour or habitat data for individuals at various receptor locations (ie locations where the dose to an individual is evaluated) is taken into account in two respects. First, the likely occupancy by the critical group at each location time is flagged to count or neglect a potential dose contribution. Occupancy at a given time has two dimensions, time of day exposure and weekday or weekend exposure. Second, the likely exposure pathways are also flagged for the critical group at each location. In both cases no contributions to individual doses are accumulated if the flags are zero. Examples are individuals exposed only during weekend recreational activities near a source; individuals exposed occupationally during the week; finally, members of the public at some distance from a source who may be chronically exposed with all pathways potentially operative. Some 40 locations where doses are evaluated are considered in present applications. These are at 1.6 and 4.8 km from the origin of coordinates centred on HIFAR, the 10 MW DIDO type research reactor at the LHRL. There is no great difficulty in varying this number.

Three sources of dosimetry data have been incorporated into the present data base in the code. The data provide effective dose and committed effective dose conversion factors. The code output gives the sum of the effective dose and committed effective dose at each designated location. External doses arising from contaminated air (semi-infinite cloud approximation) use the Kocher(1983) data and subsequent refinements. External doses arising from contaminated ground due to deposition processes use the Jacob et al(1986) data. Internal doses due to inhalation and ingestion of radioactivity use ICRP(1982) data and the more detailed data of Killough and Eckerman(1983). Physiological data such as breathing rates have been obtained from the ICRP(1975) publication on Reference Man. Data are being assembled to modify the external airborne dose module in the code to account for finite cloud effects.

The code has been written in double precision FORTRAN 77 and conforms as closely as possible to the ANSI standard so that it can be run on a variety of mainframe and mini-computers. The code has been structured to aid internal and external reviewing and includes extensive comments. The modular structure allows the alteration of various parts of the code without difficulty. A number of checks have been included to flag errors in input data when the code is being routinely used by operational staff. Typically with 10 sources, 30 nuclides and some 40 receptor locations where doses are to be evaluated the code takes about 20 minutes in batch mode on an IBM-4381 machine and uses some 6 megabytes of memory.

Other features of the code worth noting are that

- it provides operators and regulators with a tool with which to assess the impact of any new or altered operations;
- it provides information on the various contributions to the effective dose to an individual such as the proportion from a given source or particular nuclide or both; and
- although it does not at present possess a module for the evaluation of collective doses, the structure of the code readily allows the inclusion of such a module. Such an addition is planned.

Detailed reports on both the code and its applications in the context of authorised releases of

airborne radioactivity from the LHRL are in preparation.

Summary and Conclusion

A powerful general purpose multi-source, multi-nuclide and multi-receptor atmospheric transport, dispersion and dosimetry code with considerable time resolution has been developed for routine use by operators and regulators. The code has been structured to make adaptation to other situations relatively straightforward and to enable relatively simple checking of each module.

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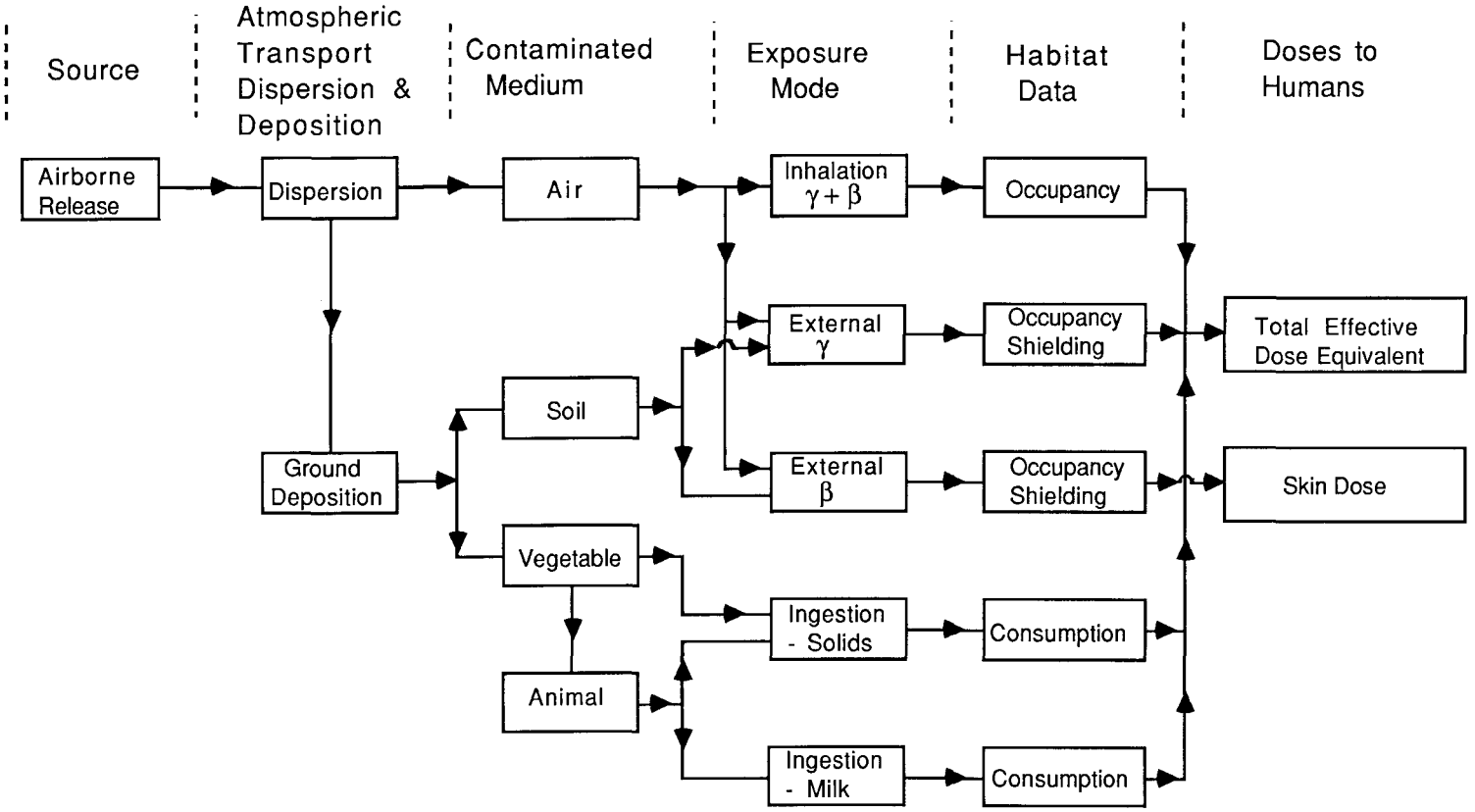

FIGURE: Major Atmospheric Pathways

EXPERIENCE WITH A FIVE ELEMENT NEUTRON DOSIMETER

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1. INTRODUCTION

The Health Physics Department of the Netherlands Energy Research Foundation (ECN) operates a personnel dosimetry service, specialized in the measurement of beta and neutron radiation. The Department serves a wide variety of institutes and companies throughout the country, ranging from research establishments to construction and drilling companies.

The number of neutron dosimeters issued each month is at present 750. An additional two hundred dosimeters are issued on a quarterly basis, mainly as accident dosimeters for power stations and fire brigades.

2. CONSTRUCTION AND OPERATION

The dosimeter is sensitive to beta, gamma and neutron radiation. The construction of the dosimeter is shown in figure 1. It contains four TLD-elements in a cassette (type Panasonic UD813). The cassette is placed in a borium-carbide holder, with appropriate windows for the use as an albedo dosimeter (manufactured by Wellhöfer Kernphysik, W. Germany). In addition to the TLD-cassette a polycarbonate nuclear track etch detector is inserted as a fifth measuring device. This foil is read out only in case of a suspected significant dose equivalent due to fast neutrons, as indicated by the albedo detection system.

All four TLD-elements are made of lithium-borate ($Li_2B_4O_7$), two enriched in ^6Li and ^{10}B, and two enriched in ^7Li and ^{11}B. The $^6Li_2{}^{10}B_4O_7$ elements are extremely sensitive to thermal neutrons, due to the large (n,α) cross sections.

The TLD-system measures the individual dose equivalents due to fotons (element 3) and beta rays (element 2 - element 3). Neutrons are detected by elements 1 and 4. This last element gives the TL-signal caused by neutrons thermalized in the body of the person wearing the badge. The B_4C-holder effectively shields the element from the thermal neutrons in the incident beam. This albedo signal is a measure of the individual neutron dose equivalent, provided additional information is available about the incident neutron spectrum. Part of this information is provided by the reading of element 1, which is proportional to the incident thermal neutron fluence. However, the relation between the albedo signal and the dose equivalent depends not only on the thermal component, but also on the type of neutron source. These relations are presented in figure 2. It is clear that the operation of this dosimetry system depends on the knowledge of the type of neutron source used by each of the monitored individuals. This type of information is therefore available in our fully automated dose calculation and registration system.

3. EXPERIENCE

As may be readily seen in figure 2 the dosimetry around reactors is expected to proceed without problems. The sensitivity of the neutron signals is even better than for fotons. In practice, a detection limit of 20 μSv is easily achievable, even in the presence of a tenfold more intense gamma-ray field. Experiments with dosimeters mounted on fantoms around the reactors have shown an excellent correspondence between remcounter readings and dosimetric data. The fraction of non-zero neutron doses varies between 5 and 25% of the total number of dosimeters used around reactors. The values measured remain below 0.1 mSv/month, a small fraction of the total dose equivalent.

Dosimetry around accelerators is less sensitive. As may be observed from figure 2 the response of the dosimeter to fast neutrons is often a factor of ten less than for fotons. However, the monitored individuals usually remain outside of shielding structures. The shielding significantly increases the relative contribution of thermal neutrons in the spectrum. This improves the sensitivity of the albedo dosimeter considerably with respect to the unshielded situation. In practice the dosimetry around fixed fast neutron sources hardly poses any problems. Measurable neutron dose equivalents are rarely encountered and center around the practical detection limit of 0.4 mSv/month.

A different situation is encountered with portable (α,n) and fission neutron sources. Here the shielding is often minimal. The corresponding dose equivalent rates are also limited in value. This means that, in practice, significant dose equivalents are seldom encountered during normal operation. The dosimeter is therefore mainly worn for the registration of doses in unforeseen events. The vast majority of the results lies below the detection limit and is reported as a zero dose. This applies to both gamma-ray and neutron irradiation.

In general, the albedo dosimeter serves its purpose very well. The registration of individual dose equivalents is as good as may be expected for a neutron dosimetry system, but certainly better than the film dosimeter. The new system has proved to be a major improvement, both from a dosimetric and from a administrative point of view.

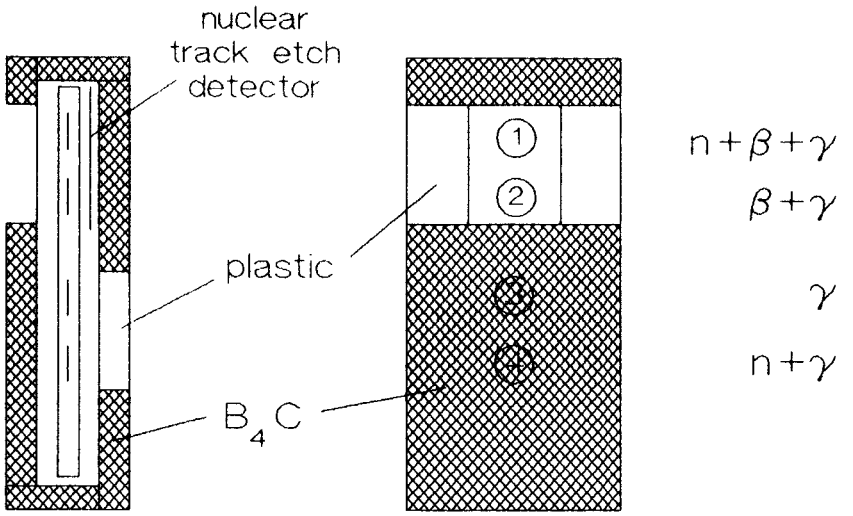

Figure 1. Construction of the dosimeter

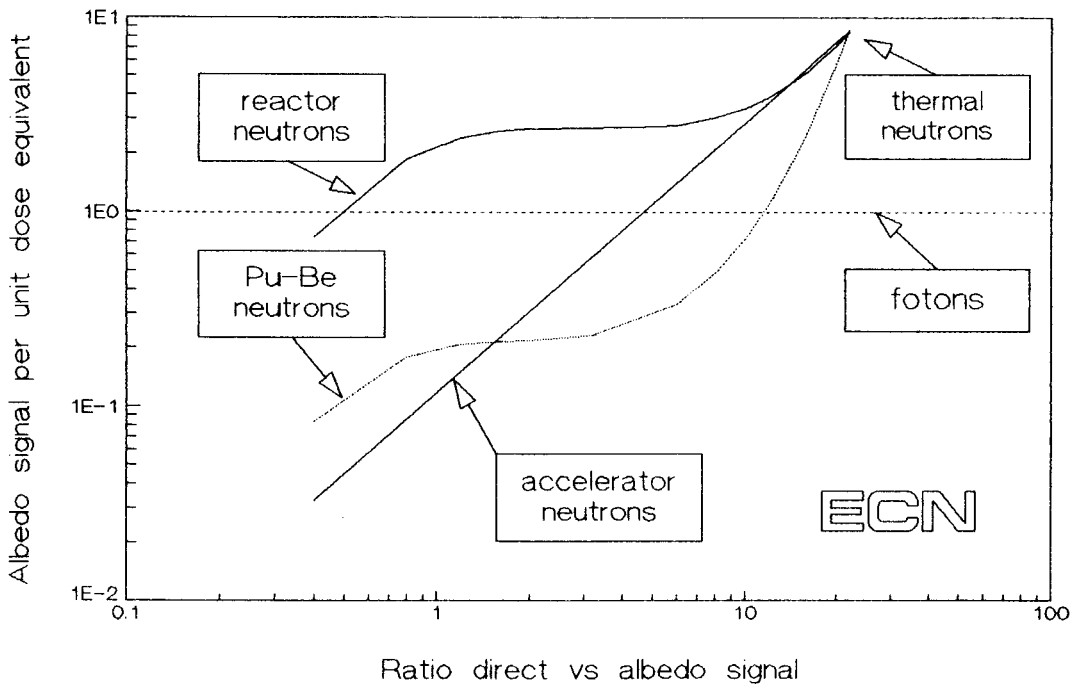

Figure 2. Neutron response

DOSIMETRIC QUANTITIES FOR 300 keV NEUTRONS

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ABSTRACT

Dosimetric quantities for 300 keV neutrons in the ICRU standard tissue sphere were evaluated. The Monte Carlo code NEDEP which performs neutron-photon-charged particles coupled transport was used in the direct estimation of absorbed dose and dose equivalent. Some important quantities calculated are as follows :

Deep dose equivalent index $H_{I,d}$: 1.78×10^{-11} S_v-cm^2

Shallow dose equivalent index $H_{I,s}$: 2.08×10^{-11} S_v-cm^2

Ambient dose equivalent $H^*(0.07)$: 1.7×10^{-11} S_v-cm^2

Ambient dose equivalent $H^*(10)$: 1.78×10^{-11} S_v-cm^2

Effective quality factor $\bar{Q}^*(10)$: 12.4

SAFETY ANALYSIS AND RISK ASSESSMENT FOR THE VITRIFICATION OF HIGH-LEVEL RADIOACTIVE WASTES

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ABSTRACT

In 1980, the United States Congress authorized the U.S. Department of Energy (West Valley Demonstration Project Act; PL-96-368) to solidify 2.1 million litres of liquid high-level radioactive waste (HLW) currently stored at the Western New York Nuclear Service Center (approximately 50 miles south of Buffalo, New York) into a form suitable for transport to and disposal in a federal geologic repository. The method chosen to accomplish this task is vitrification in borosilicate glass.

These wastes were produced during the period 1966-1972 when facilities at the center were used for the only commercial nuclear fuel reprocessing plant to have operated in the U.S. These wastes contain approximately 30 million curies of mixed fission products and transuranic radionuclides and are presently being stored in two underground tanks at the West Valley site.

This paper presents a brief overview of waste treatment and vitrification processes under development and construction at West Valley and the results of safety analysis/risk assessments that have been performed to estimate the radiological impacts to on-site and off-site individuals as a result of the waste treatment and vitrification programs. Projections have been made regarding the radiological doses expected from normal opertions as well as a result of postulated, theoretical abnormal conditions and major accidents. The West Valley Safety Analysis and Environmental Impact Assessment program is performed in strict accordance with the requirements of the U.S. Department of Energy and other federal agencies.

TWO IMPORTANT FACTORS INFLUENCING THYROID INTERNAL DOSE FOR RADIOIODINE

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INTRODUCTION

It is well known that physiological changes can cause gross variation in metabolism and considerable error can result if data for the standard man is used in dose calculation [1]. Although Reference Man and other similar model data are still used for dose calculation, it is of more interest to pay attention to non-standard physiological situations. One of the most representative cases involves ^{131}I. Previous workers [2-8] have indicated that thyroidal function is one of the most important factors that determines the thyroid dose due to the intake of radioiodine. Further observations are needed to determine its significance. The reported data has considered only thyroid uptakes of 5%, 15%, 25%. Additionally, since different physiological conditions are a factor evident in influencing the organ doses, it is of interest to enquire whether certain medicines which can affect the general metabolism of the thyroid gland could be used to reduce the internal dose received by the thyroid following the intake of radioiodine. In this paper, we consider two aspects.

METHODOLOGY

Women were selected from individuals who had received a thyroid iodine uptake examination. Their ages were limited to the range 25 to 35 years. Excluded from the observations were individuals who were identified as having a rich iodine diet, disease of internal organ of the body etc.

Each subject drank a glass of water containing between 3.2 and 6.4 μCi of ^{131}I(NaI). The total urine output following administration was collected daily throughout a 15-day period. Urine sample was analysed immediately each day. First, stable iodine carrier was added and radioiodine concentrated by AgI-^{131}I co-precipitation. The ^{131}I gamma-ray counts were measured using a sodium iodide well counter connecting to a multichannel analyzer. The system calibration factor was determined using a standard of known activity.

Radioactivity was monitored by orientated scintillation detectors, each with a crystal-to-skin distance of 24 cm and a collimator-to-skin distance of 8 cm. One detector (3 in. diam x 1.5 in. deep crystal) was placed over the thyroid area (neck detector), another identical detector was centered over the left thigh at a point 10 cm proximal to the patella (thigh detector). Measurements by the two detectors were made at 4 and 24 hours following administration, and repeated at 24-hour intervals throughout the 15-day period. The calibration factor was determined with a standard of known activity contained in a plexiglass neck phantom. Counts in the thigh detector were used to correct for neck background in thyroid counting.

RESULTS AND DISCUSSION

BIOLOGICAL MODEL

According to the urinalysis, the total amount of radioiodine retained at the end of each day following administration of ^{131}I can be calculated if it is assumed that the removal of radioiodine occurred only by physical decay and excretion in urine. Application of linear regression and curve-stripping techniques showed that the retention of radioiodine can be described by a two-compartment model, given in equation (1) below, and in which parameter values for fast and slow components in the kinetic model can

be determined for each subject respectively

$$R(T) = k_1 e^{-b_1 t} + k_2 e^{-b_2 t} \quad (1)$$

In this equation k is the fraction of ^{131}I distributed in each compartment and b is the corresponding biological clearance rate for each compartment.

INTERNAL RADIATION DOSE

The internal radiation doses were estimated by the method presented by the MIRD Committee of Society of Nuclear Medicine [9]

$$\bar{D}(T) = \sum_h \tilde{A}_h S(T \leftarrow h) \quad (2)$$

where $\bar{D}(T)$ is the mean absorbed dose in T, $\tilde{A}_h$ is the cumulated activity in source region h, $S(T \leftarrow h)$ is the mean absorbed dose in target region T per unit cumulated activity in h.

S(T←RB) cannot be obtained from MIRD Pamphlet No. 11 [10], S(T←RB) is calculated using the method presented by Coffey et al. [11]. A_h is estimated by

$$\tilde{A}_h(0,\infty) = \frac{Aok}{b + \lambda r} \quad (3)$$

where Ao is the administered activity of ^{131}I, is the physical decay constant of ^{131}I, b is the biological clearance rate.

The thyroid internal dose factors for each subject were calculated. In most case, the thyroid internal dose factors were intimately related with thyroidal functions. However, the physical decay constant is much greater than the biological clearance rate of the slow component for ^{131}I. Consequently fluctuation in the effective clearance rate is insignificant even if the biological clearance rate varies. Hence the effect of the biological clearance rate on the thyroid radiation dose is far less than that due to the thyroid iodine uptake rate. For example, an increase in the biological half-life from 100 to 250 days [12], results in only a 5% rise in the thyroid dose; however, if the thyroid uptake rate increases by the same magnitude, then this causes about 250% rise in thyroid dose. From our experimental results, a linear correlation analysis was made relating the thyroidal functions and the thyroid internal dose factors for all subjects. The following regression equation has been obtained

$$\bar{D}_{Thy} = 2.75 + 15.2K_{Thy} \quad (4)$$

where the $\bar{D}$ is in mGy/MBq (3.70 mrad/μCi), K is the thyroid iodine uptake rate, the maximum thyroid iodine uptake rate is assumed to be 100%. Experimental results are shown in Fig. 1.

EXAMINATION FREQUENCY, EFFECTIVE DOSE EQUIVALENT, COLLECTIVE EFFECTIVE DOSE EQUIVALENT

Based on the maximum thyroid iodine uptake rates of those individuals who underwent a thyroid uptake examination, ten groups were distinguished. The variation among groups is 10%. Using formula (4), the mean absorbed dose per unit of administered activity D_{Thy}, the mean effective dose equivalent per examination H_E (the administered ^{131}I activity per procedure is assumed 148 KBq) and corresponding collective effective dose equivalent S_E for each group were calculated. They are all shown in Table 1. Table 1 also gives the distribution of examined individuals by the thyroid uptake rate values.

From the 1986 census data of Suzhow City and its near suburb, the following results were derived: (a) the examination frequency of thyroid iodine uptake is 0.88 man per thousand population; (b) the annual per caput effective dose equivalent is 3.95 Sv.

ACCELERATING EFFECT OF MEDICINE

It is well known that stable iodine is effective in blocking the uptake of radioiodine by the thyroid. However, stable iodine is of no use for the radioiodine already deposited in the thyroid. Therefore, we questioned whether certain medicines that can affect the biological metabolism of thyroid might accelerate the release of radioiodine from the thyroid. For this reason, four subjects took Methimazolum together with Thyroideum after 24-hours of administration. The Methimazolum dose was 10 mg orally, three times daily and Thyroideum 40 mg orally. Both were given for 10 days. Monitoring methods were similar to those for non-accelerating subjects. Table 2 presents the mean biological half-lives and thyroid doses of the accelerating group and control group from two measurement techniques.

From Table 2, we can see that taking Methimazolum together with Thyroideum can accelerate radioiodine release from the thyroid, increase the urine radioiodine output, and decrease the thyroid internal dose.

It is of significance that the results calculated from urine and from the thyroid extracount show a marked difference for the accelerating group. The biological half-lives from urine output is about four times greater than that from the thyroid extracount. We are presently unable to explain this result. We suggest that the form of accelerated iodine from the thyroid may be organic, and its main removal path is not via the kidney. This needs to be investigated further.

SUMMARY

In summary, we have observed that: (1) the thyroid function should be taken into account in dose calculation for the intake of radioiodine; (2) the release rate for radioiodine deposited in thyroid can be accelerated by taking certain medicines, such as Methimazolum together with Thyroideum.

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TABLE 1
THE DISTRIBUTIONS OF EXAMINED INDIVIDUALS, D_{Thy}, H_E, S_E BY THE THYROID UPTAKE RATE VALUES

MTUR, %	0-	10-	20-	30-	40-	50-	60-	70-	80-	90-	Total
Males	12	20	50	53	38	28	24	23	18	12	278
Females	55	68	144	241	209	197	156	108	108	83	1370
Both	67	88	194	294	247	225	180	132	126	95	1648
D(Thy.) Gy/MBq	0.078	0.23	0.38	0.53	0.69	0.84	0.99	1.15	1.30	1.45	
H_E, mSv	0.47	1.37	2.27	3.17	4.08	4.98	5.88	6.78	7.69	8.59	
S_E, man Sv	0.031	0.12	0.44	0.93	1.01	1.12	1.06	0.90	0.97	0.82	7.39

MTUR: maximum thyroid uptake rate

TABLE 2
THE RESULTS OF ACCELERATING EFFECT CALCULATED FROM URINE OUTPUT AND THYROID EXTRACOUNT (MEAN + DEVIATION)

			From urine output		From extracount	
Groups	No.	MTUR %	T_b days	TIDF Gy/MBq	T_b days	TIDF Gy/MBq
Accelerated	4	0.91 ± 0.01	45.0 ± 10.4	1.15 ± 0.043	11.3 ± 5.5	0.83 ± 0.23
Control	4	0.89 ± 0.01	89.7 ± 4.9	1.36 ± 0.035	73.5 ± 16.0	1.32 ± 0.030

MTUR: maximum thyroid uptake rate

T_b: the biological half-life of slow component

TIDF: thyroid internal dose factors

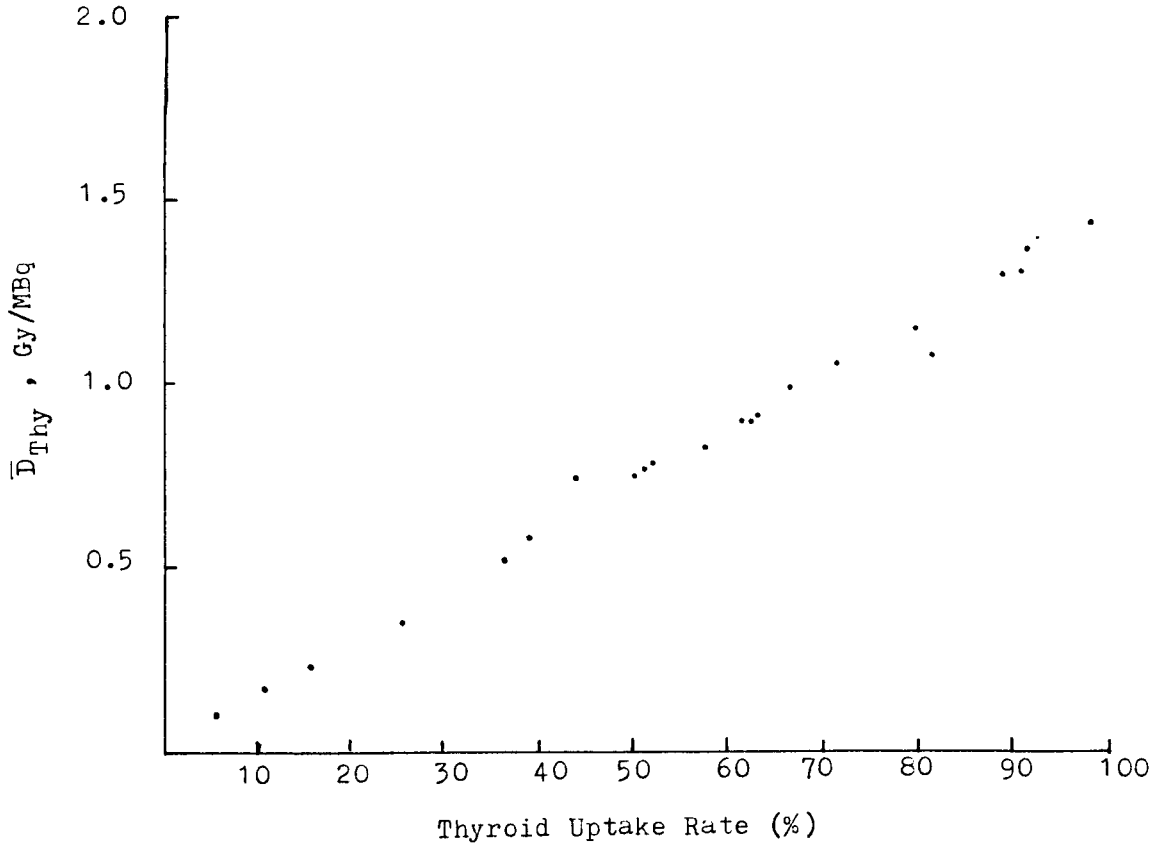

Fig.1. Thyroid internal dose factors as a linear function of thyroid iodine uptake rates

UK STANDARDS FOR EXPOSURE TO RADON DAUGHTERS IN DWELLINGS

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INTRODUCTION

The NRPB has a formal Direction from Health Ministers, given in 1977, which requires it to advise the appropriate Government departments and statutory bodies on the acceptability of the recommendations or proposals to, and on their application, in the United Kingdom, whenever ICRP and various other bodies publish those recommendations or proposals. The NRPB studied the matter of exposure to radon exposure in dwellings and issued formal advice on standards in January 1987 [1] together with a supporting document [2]. This paper describes the background to the advice and its implementation.

LIMITATION SCHEMES

In 1984, the International Commission on Radiological Protection (ICRP) published principles for limiting exposure of the public to natural sources of radiation which dealt, in the main, with indoor exposure to radon daughters [3]. ICRP drew a distinction between the control philosophies to be applied to existing and future dwellings and developed limitation criteria for radon daughters in both circumstances. For existing dwellings, where any action would have to be remedial, it suggested an Action Level which corresponds to an annual effective dose equivalent of 20 mSv (200 Bq m^{-3} equilibrium equivalent radon concentration, EER). At this Action Level, simple action to reduce exposure might be considered: at a level several times higher, severe action might be appropriate. For future dwellings, which could be subject to control at the stages of decision and planning, it suggested an Upper Bound corresponding to an annual effective dose equivalent of 10 mSv. This value was determined mainly from a comparison of the related risk of lung cancer with other risks in the home.

Earlier in 1984, the Royal Commission on Environmental Pollution (RCEP), considered the quality of indoor air [4]. It recommended action in existing dwellings where the annual effective dose equivalent exceeded 25 mSv and, although it discussed a lower value of 5 mSv for future dwellings, was content to await an assessment of the forthcoming ICRP recommendations. The views of both Commissions were therefore similar in concept although they differed slightly with regard to numerical values. In its response to the RCEP report, the Government indicated its intention to await the advice of NRPB before considering what measures might be taken.

RADON RISKS

From the risk factor of 1.65 10^{-2} Sv^{-1} for serious health effects underlying the present basic recommendations of ICRP, which derives from an absolute risk model, the lifetime risk of lung cancer from chronic exposure to radon daughters may be calculated: it is about 1% for 10 mSv per annum. At its meeting in Como in September 1987, however, ICRP addressed itself to the question of risk estimates [5] and concluded that the results of a definitive study of the new calculations of doses to the atom bomb survivors, coupled with the longer follow-up time, would raise the risk estimate approximately twofold. An ICRP report on lung cancer risk from radon daughters was also published in August 1987, in which a relative risk

model was adopted to extrapolate the epidemiological data for miners to the general population [6]. For a reference population defined by life expectancy and underlying frequency of lung cancer, a lifetime risk around 1.3% was estimated for continuous exposure to 100 Bq m^{-3} EER or at 10 mSv per annum. For the population of the UK, the lifetime risk would be approximately doubled. These recent developments suggest therefore that the risk associated with indoor exposure to radon daughters may at least be twice that inferrable from the basic ICRP position a year or so previously.

The potential significance for human health in the UK of various levels of continuous exposure to radon daughters can be assessed by comparisons with other causes of premature death [2]. As with ICRP, such a comparison of risks weighed heavily in the NRPB consideration of this matter.

NRPB RECOMMENDATIONS

The essentials of the NRPB recommendations [1] are:

"Action should be taken in some existing dwellings to reduce exposure to radon and measures introduced to limit exposure in future dwellings. The scheme suggested by ICRP is an appropriate basis on which to proceed. Taking into account factors such as the relative level of risk, the feasibility and costs of implementation, and the numbers of dwellings involved, the Board makes the following recommendations for the UK".
"For existing dwellings, graduated measures related to effective dose equivalent are recommended above an Action Level of 20 mSv per annum. At this Action Level, simple action to reduce exposure to radon should be taken; at a level several times higher, severe action is likely to be necessary. Cognisance should also be taken of the magnitude of the annual dose, in particular cases, when deciding how soon action should be taken: the dwellings with the highest levels should be given priority."
"For future dwellings, an Upper Bound of 5 mSv per annum is recommended as the basis for implementing changes to building procedures in areas of the country where high levels of radon are likely".

The NRPB recommendations introduced a temporal condition not included in the ICRP recommendations [3], namely, that the magnitude of the annual dose should determine the urgency of the action to reduce it. This principle was recognised by the RCEP when recommending that priority be given to existing dwellings where levels were such as to cause anxiety [4]. It was also recognised, however, that the matter of primary concern is prolonged exposure over many years at high levels. Given the variety of domestic circumstances that were likely to obtain, it seemed unreasonable to put forward a firm formula, but guidance along the following lines was offered. For all doses above the Action Level, action should follow as soon as reasonably practicable, and for annual doses above about 50 mSv, this should be within a year or so. For annual doses from 20 mSv to 50 mSv, it would be desirable for action to have been completed within a few years. In each case, the interval would apply from the date of confirmation of high levels by appropriate measurements.

NUMBERS OF DWELLINGS

Members of NRPB staff had measured radon in 2300 dwellings chosen at random throughout the UK and supplemented these results with measurements of 700 dwellings in areas with geological characteristics that were expected to cause high levels of radon indoors, mainly in southwest England [7]. The geometric mean value of the annual effective dose equivalent from radon

daughters in dwellings is about 0.7 mSv, but there is a considerable spread of results. Analysis of the results indicated that the approximate numbers of dwellings, out of a total stock of 21 million, in which the annual dose exceeds nominal values, increases very rapidly as the nominal value decreases. The areas of the country in which an appreciable number of dwellings occurs also increases rapidly with decreasing nominal value. There are also practical problems in identifying dwellings over nominal values: these increase steeply as the nominal value is decreased, mainly because the geographical areas cannot be clearly defined and the total number of dwellings that need to be surveyed to find dwellings over the nominal value becomes much greater. With the Action Level adopted, however, most of the 20 000 dwellings estimated to have higher levels were likely to be found in southwest England, generally in and around granite areas, and it was judged that these areas should be given priority.

The numbers of new dwellings built each year that might ordinarily exceed the Upper Bound of 5 mSv y^{-1} were likely to be a few thousand. Most of these would again be found in the same areas of southwest England. The long-term rate of replacement of dwellings in the UK is about 200 000 per annum although this fluctuates markedly with economic conditions.

In round terms, therefore, about 0.1% of existing dwellings were likely to be affected by the Action Level, whereas 1% of new construction would be affected by the Upper Bound. The geographical impact is fairly well circumscribed.

REMEDIAL MEASURES

The cause of high doses from radon daughters in dwellings is the high rate of entry of radon gas from the ground. Although increased ventilation can reduce radon concentrations, the most appropriate action is to decrease the transfer of radon to the living space either by making the floor more effective as a barrier or by diverting the gas from the underfloor space. Rough estimates indicated that costs in existing dwellings might range from £100 for a simple action such as sealing floor cracks, through £1000 for an intermediate action such as underfloor extraction, to £10 000 for a severe action such as floor replacement. It would therefore be possible to choose actions commensurate with doses. Appropriate techniques of construction could be adopted for future dwellings, without radical departure from present practices, so as to prevent high exposures of future occupants. Such changes might entail some, but not substantial, additional costs.

GOVERNMENT ACTION

The Government accepted the advice, and the Minister of State for the Environment made a statement on radon in the House of Commons and announced a programme of work to tackle the problem. There are two parts to the programme, identification of houses with radon concentrations above the Action Level and the development of remedial and preventive measures.

The identification of houses above the Action Level is being tackled in three separate but interconnected ways: systematic surveys, for a full year, of all dwellings in the immediate neighbourhood of houses with the higher known radon concentrations; on-demand screening measurements, at government expense, in areas with the potential for high indoor radon levels; on-demand screening measurements, at householders' expense, in other areas. Surveys commenced in the summer of 1987, in southwest England, and early results support the estimates of the numbers of dwellings affected by the Action Level.

The Government is also funding a research programme to develop remedial and preventive measures. This will result in the provision of guidance notes for householders and builders on practical measures against radon. On the important question of costs, the Minister made the following comment "The responsibility for remedial measures in houses must rest with the house owner or the landlord in the case of both public and private rented accommodation. The Government are prepared to consider offering financial assistance towards the costs of remedial work to the most needy owner-occupiers".

CONCLUSION

It is now generally agreed that radon daughters are, on the average, the dominant cause of human exposure to ionising radiation: with thoron daughters, they contribute about 50% of the total. This circumstance is a fact of life, but the problem of excessive exposure to radon daughters in dwellings throughout the UK is being tackled in a sensible and measured manner. The extent of the problem has been defined, action levels have been set, surveys are in progress to identify particular houses at risk, and remedial and preventive measures are being developed.

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INDOOR EXPOSURE TO NATURAL RADIATION IN DENMARK

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INTRODUCTION

A survey on natural radiation in a representative sample of Danish dwellings (1) was carried out by measuring the radon concentration and the absorbed dose rate in air from external radiation in about 500 dwellings. The householders of a total of 926 dwellings were invited to participate in the survey by postal contact. The dwellings were selected randomly from a central register of all Danish buildings. The measurements were carried out with one half in the summer period from April to September 1985 and the other half in the winter period from October 1985 to March 1986. The radon concentration and the external radiation were measured by passive closed dosemeters (2). The sensitive elements were nuclear track detectors (Cr-39) for the measurement of radon and thermoluminescence detectors (LiF TLD-700) for the measurement of external radiation. In each dwelling the radon concentration was measured in the living-room and in a bedroom. The external radiation was measured only in the living-room. When returning the dosemeters the occupants of the dwellings were asked to fill in a questionnaire giving information pertinent to the conditions that may influence the radon concentration in a dwelling: type and age of the dwelling, building materials, ventilation conditions etc.

RESULTS AND DISCUSSION

Measurements of the radon concentration were obtained for 496 dwellings. The arithmetic mean values for different subgroups of the measured dwelling are shown in table 1.

Table 1. Arithmetic mean values of the radon concentration for different subgroups of dwellings.

	Single-family houses		Multi-family houses	
	Arithmetic mean Bq/m^3	Number of dwellings	Arithmetic mean Bq/m^3	Number of dwellings
Living-room, summer	58	168	17	78
Living-room, winter	93	180	24	70
Bedroom, summer	49	168	17	78
Bedroom, winter	73	180	21	70

The arithmetic mean values of the radon concentration for the single-family houses are seen to be higher than the corresponding mean values for multi-family houses. This indicates that the subsoil is the main radon source for single-family houses. For the single-family houses the mean radon concentration is higher for the living-room than for the bedroom and higher during the winter than during the summer. In multi-family houses all the mean values are close to 20 Bq/m^3. For each dwelling an annual average radon concentration has been calculated by multiplying the mean value of the measured semi-annual radon concentration in the living-room and in the bedroom by a correction factor. The correction factor was estimated for different subgroups from the arithmetic mean values for the summer and winter period. The distributions of these annual mean radon concentrations in single-family houses and in multi-family houses have been combined in figure 1. This distribution is expected to be representative of the annual mean radon concentration in Danish dwellings. The corresponding cumulative distribution is shown in figure 2. The mean value of the annual radon concentration was found to be 47 Bq/m^3. The representative cumulative distribution shows that the annual average radon concentration exceeds 200 Bq/m^3 in less than 2% of the Danish dwellings, and that the average radon concentration exceeds 800 Bq/m^3 in only a very limited number of Danish dwellings. By weighting with the distribution of the Danish population (69% of the population are living in single-family houses and 31% in multi-family houses) the population-averaged radon concentration was found to be 53 Bq/m^3.

The most important parameter for the single-family houses was found to be the character of the subsoil under the house. Single-family houses placed on primarily moraine clay was found to have a significantly ($P > 99\%$) higher mean radon concentration (geometric mean 68 Bq/m^3) than houses placed on the other types of subsoil (geometric mean 39 Bq/m^3). This is in good agreement with measurements of the radon emanation from samples of Danish soils (3). No significant differences were found between other types of subsoil, which generally may be characterized as "sand and gravel". The analysis further shows a significant difference within the moraine clay group between the results from the summer period (geometric mean 54 Bq/m^3) and the winter period (geometric mean 85 Bq/m^3). No similar significant difference was found within the sand and gravel group. After the character of the subsoil, the analysis has shown that the type of foundation of the single-family houses is dominant. The highest mean radon concentration was found for houses without basement or with a basement under only a part of the house (geometric mean 65 Bq/m^3). A lower mean radon concentration was found in houses with a crawl space (geometric mean 46 Bq/m^3) and the lowest values were found in houses with a full basement (geometric mean 34 Bq/m^3). These findings reflect different possibilities of entry of radon from the underlying soil through the different types of foundation into the living-rooms and bedrooms.

Measurements of the external radiation were obtained for 489 dwellings. The results are normally distributed. The arithmetic mean values of the absorbed dose rates in air were found to be 83 nGy/h and 90 nGy/h for single-family houses and multi-family houses, respectively. The mean value was found to be higher in buil-

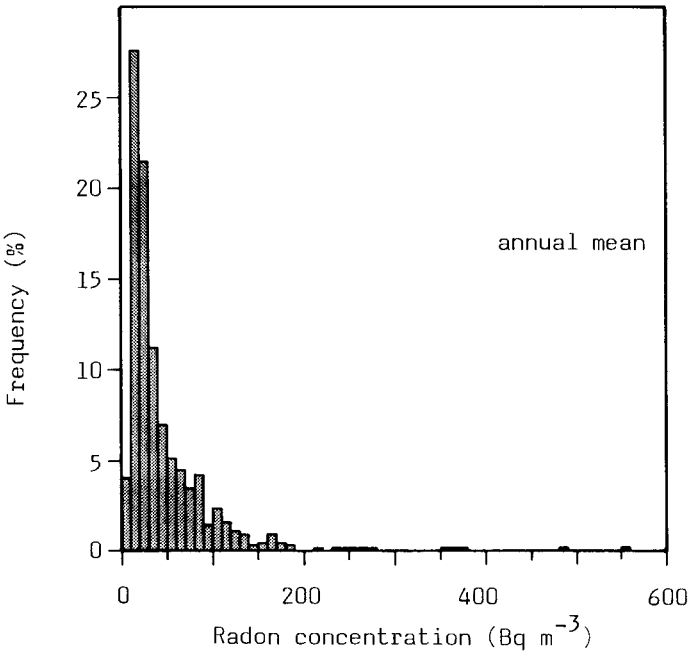

Fig. 1 Frequency distribution of the annual average radon concentration in Danish dwellings based on the 496 investigated dwellings.

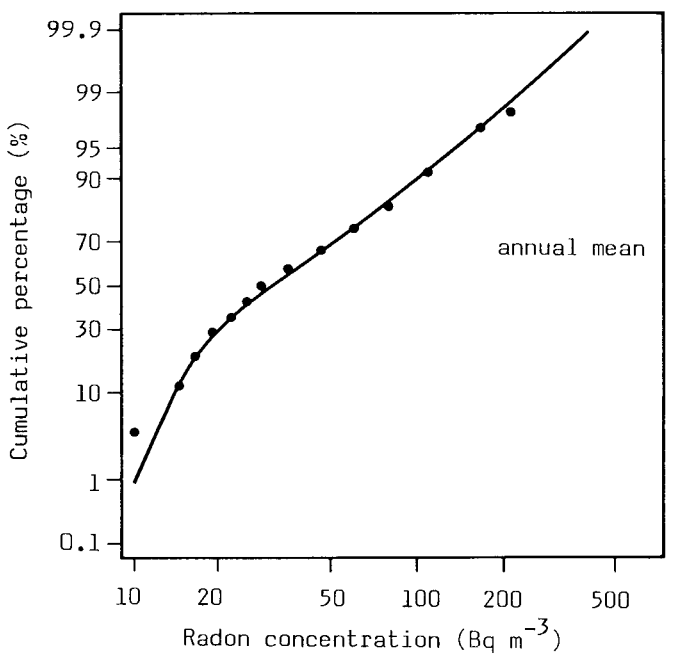

Fig. 2 Cumulative distribution of the annual average radon concentration based on the 496 investigated dwellings. The solid line represents the weighted sum of the two log-normal distributions for the single-family houses and multi-family houses.

dings constructed of clay bricks (single-family houses: 84 nGy/h, multi-family houses: 95 nGy/h) than in buildings constructed of concrete (single-family houses: 74 nGy/h, multi-family houses: 76 nGy/h). This is in good agreement with earlier measurements of natural radioactivity in Danish building materials (4). Furthermore the external radiation was found to be higher in buildings constructed before 1960 (single-family houses: 87 nGy/h, multi-family houses: 98 nGy/h) than in buildings constructed after 1960 (single-family houses: 77 nGy/h, multi-family houses: 77 nGy/h). This is due to a more frequent and more substantial use of clay bricks before 1960.

DOSE TO THE DANISH POPULATION

Assuming residence probabilities of 0.7, 0.2 and 0.1 for staying indoors at home, indoors elsewhere and outdoors, respectively, and assuming the mean value of the annual radon concentration to be 47 Bq/m^3 indoors and 8 Bq/m^3 outdoors, the total annual radon exposure in Bq · y/m^3, E, may be expressed as (5): $E = 0.7 \cdot C_{Rn} + 10$, where C_{Rn} is the annual average radon concentration in the home. With an equilirium factor of 0.5 and a conversion factor to effective dose equivalent of 0.076 mSv per Bq · y/m^3 EER (6) the population averaged radon concentration of 53 Bq/m^3 in Danish dwellings corresponds to a mean effective dose equivalent to the Danish population of 1.8 mSv in a year from radon daughters. The mean effective dose equivalent to the Danish population from terrestrial gamma rays has been calculated to be 0.32 mSv in a year. A contribution of 31 nGy/h from cosmic radiation has been subtracted from the mean absorbed dose rate in air indoors in Denmark of 85 nGy/h to give a mean indoor gamma ray dose rate of 54 nGy/h. The mean outdoor gamma ray dose rate in Denmark is 35 nGy/h (7).

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ACTIVITIES OF THE INTERNATIONAL NON-IONIZING RADIATION COMMITTEE OF IRPA

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It is now about 10 years ago that, during the 4th International Congress of IRPA in Paris, in 1977, the IRPA Executive Council and General Assembly amended the Constitution of IRPA to include non-ionizing radiation into its scope of activities and set up the International Non-Ionizing Radiation Committee (INIRC) to deal specifically with the different aspects of protection against non-ionizing electromagnetic radiation and ultrasonic waves.

To achieve in a rational way the tasks that the IRPA had imparted to it, the Committee agreed that the different types of NIR would be dealt with successively, each according to the following sheme :

1) compile all available background information and determine the basic health criteria for the relevant NIR.
2) develop guidelines on appropriate limits of exposure for workers and the general public.
3) provide guidance on the practical measures to be taken for the safe use of the various NIR.

During the ten years which have elapsed since INIRC was created, its members devoted their efforts to carrying out this programme and at the present time about 15 reports have been produced by IRPA/INIRC either alone or in cooperation with other international organizations.

BASIC HEALTH CRITERIA

To assess the health risk associated with any noxious agent, it is first necessary to analyze all available data relating to its biological effects, the 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. Although on an incomparably smaller scale, this task is in some way similar to that achieved by UNSCEAR in the field of ionizing radiation.

Therefore, the IRPA, the World Health Organization (WHO) and the United Nations Environment Programme (UNEP) agreed that the IRPA/INIRC and the WHO/Environmental Health Division would cooperate in the development of Environmental Health Criteria documents relating to the different NIR within the frame of the Environmental Health Criteria programme for chemical and physical noxious agents which was carried out by the WHO with the

financial support of the UNEP. The purpose of these documents is to provide information for health authorities and regulatory agencies on the possible effects of exposure to the relevant NIR on human health and to give guidance on the assessment of risks from occupational and general population exposure.

The first four UNEP/WHO/IRPA documents - E.H.C. 14 for Ultraviolet Radiation (1979), E.H.C. 16 for Radiofrequency and Microwaves (1981), E.H.C. 22 for Ultrasound (1982) and E.H.C. 23 for Lasers and Optical Radiation (1982) - are already well known, and I shall therefore only present briefly the two last ones which were published respectively in 1984 and 1987.

E.H.C. 35 for Extremely Low Frequency Fields (1984) is mainly concerned with the effects of ELF electric fields at the power frequencies of 50 and 60 hertz. Some of the main conclusions drawn by the Task Group after a careful review of laboratory studies and observations made on human subjects are that :

- adverse human health effects from exposure to ELF electric fields, normally encountered in the environment or the workplace, have not been established ;
- whilst it would be prudent in the present state of scientific knowledge not to make unqualified statements about the safety of intermittent exposure to electric fields, there is no need to limit access to regions where the field strength is below about 10 kV/m. Even at this field strength, some individuals may experience uncomfortable secondary physical phenomena such as spark discharge, shocks, or stimulation of the tactile sense;
- it is not possible from present knowledge to make a definitive statement about the safety or hazard associated with long-term exposure to sinusoidal electric fields, in the range of 1-10 kV/m. In the absence of specific evidence of particular risk or disease syndromes associated with such exposures, and in view of experimental findings on the biological effects of exposure, it would be prudent to limit exposure, particularly for members of the general populations.

The last document, E.H.C. 69 for Magnetic Fields (1987) includes a detailed review of the data on biological effects of static and time varying magnetic fields at extremely low frequencies.

For static fields, available knowledge indicates the absence of any adverse effects on human health due to exposure to fields up to 2 teslas.Time-varying magnetic fields generate internal electric currents which may perturb biological functions. Levels of induced current densities (and correlated magnetic flux densities) are given for which biological effects ranging from minor to hazardous have been reported.

Several recent epidemiological reports present preliminary data suggesting an association of an increase in cancer incidence, with exposure to very weak 50 or 60 Hz magnetic fields. The preliminary nature of the epidemiological evidence, and the relatively small increment in reported incidence, suggest that, although these epidemiological data cannot be dismissed, there must be considerable further study before they can be accepted.

Besides the INIRC cosponsored several International Symposia on the biological effects of electromagnetic waves with the Union Radioscientifique Internationale and the Bioelectromagnetics Society or of ultraviolet radiation with the Commission of the European Communities and the Radiation Protection Society of the Netherlands.

GUIDELINES ON EXPOSURE LIMITS

The recommendation of appropriate exposure limits for workers and for the general public is the chief objective that has been assigned to the Committee by the Executive Council and the General Assembly of IRPA. The IRPA/INIRC guidelines on limits of exposure to the different NIR are established on the basis of the scientific data collected for the relevant environmental health criteria and on any later published research data. The purpose of the guidelines is to deal with the basic principles of protection against the relevant NIR, so that they may serve as guidance to the various international and national bodies or individual experts who are responsible for the development of regulations, recommendations or codes of practice to protect the workers and the general public. All guidelines are published in Health Physics.

The first IRPA/INIRC Interim guidelines on limits of exposure to airborne ultrasound and to radiofrequency electromagnetic fields were published in 1984 (Health Physics, 46, 969-974 and 975-984). These represent the first attempt made at an international level to achieve harmonization of the exposure standards used for NIR in the different countries.

This topic was especially controversial in the case of radiofrequency radiation where the Committee had to reconcile very different scientific opinions and to reach consensus on exposure limits at a time when the values recommended in various countries varied over a range from 10 to 10,000 microwatts per square centimetre. Today, I think we can say that this objective has been achieved and the IRPA guidelines are recognized in many countries as a highly valuable reference in radiofrequency radiation protection. However, since 1984, si gnificant new data were obtained concerning the interaction of radiofrequency fields and living systems. Therefore, some amendments were needed and a revised version has been published in the January 1988 issue of Health Physics . In particular, the occupational basic limits for partial-body energy absorption were amended on account of new dosimetric data showing

that the specific absorption rate in various body organs is much more heterogeneous than previously assumed. Except for pulsed fields, derived limits were only slightly modified, and some practical rules intended to facilitate the application of the guidelines were added.

Guidelines on limits of exposure to ultraviolet radiation and to lasers were published in 1985 (Health Physics, 49, 2, 331-340 and 341-359). In the field of optical radiation, because of better agreement on the basic health criteria, there were only minor differences in the few national standards already in use. The laser exposure limits are complex functions of wavelength, exposure duration and viewing conditions. Recently, the INIRC drafted some amendments to the laser guidelines, which simplify and clarify the rules for determining the applicable exposure duration and the additivity of repetitive pulses (Health Physics, 1988, in press). Several areas of concern still exist regarding exposure limits for ultrashort pulses and for long-term chronic or repeated exposures. It is also intended to update the guidelines for ultraviolet radiation taking into account new data on the relative spectral effectiveness in the UV-A region (315 to 400 mm).

These different guidelines together with a Review of concepts, quantities, units and terminology for NIR protection (Health Physics, 1985, 49, 1329-1362) will be available soon in the form of a single book.

Furthermore the Committee has made a Statement on the alleged radiation risks from visual display units (Health Physics, in press) and is completing Guidelines on limits of exposure to 50/60 Hz electric and magnetic fields as well as for the assessment of safety of patients during magnetic resonance diagnostic examinations.

GUIDANCE FOR SAFE OCCUPATIONAL PRACTICE

Concerning operational protection and the application of the guidelines in occupational practice the IRPA/INIRC collaborates with the International Labour Organization. Two reports on NIR have been issued up to now as part of the International Programme for the Improvement of Working Conditions and Environment (PIACT). Occupational hazards from non-ionising electromagnetic radiation, Occupational Safety and Health Series n°53, ILO 1985, and Protection of workers against radio-frequency and microwave radiation : a technical review, O.S.H.S. n°57, ILO 1986.

Finally, in the name of the INIRC, I want to express my most sincere thanks to all Committee members for the work they achieved during these 10 years, to the IRPA Executive Council for its continuous support, to the World Health Organization, the International Labour Office and the Commission of the European Communities for their cooperation and financial support.

EFFECTS OF 2450 MHZ MICROWAVE RADIATION ON MEIOSIS AND REPRODUCTION IN MALE MICE.

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We reported previously that repetitive exposures of male mice over 12 days, i.e. the duration of the first meiotic prophase (P I) of the spermatocyte I stage of spermatogenesis, to pulsed 9.4 GHz (Manikowska et al.,1979) or continous 2450 MHz (Manikowska-Czerska et al., 1985) microwaves induces chromosomal aberrations, primarily multivalent configurations, in germ cells. The experiments described below were designed to examine effects of exposure over consecutive stages of P I, and to ascertain whether the induced aberrations may be transmitted to the offspring, and affect their development. The results indicate that aberrations may be induced by exposures over the duration of initial prophase and early pachytene, and only to a limited extent after exposure over later stages of P I. An increase in fetal postimplantation loss and live balanced translocation carriers were observed among the litters sired by exposed males.

MATERIAL AND METHODS.

Adult male ICR mice were exposed 30 min. daily to continous 2450 MHz microwaves in an environmentally controlled waveguide system (Ho et al.,1973). Whole body average specific absorption rate (WBA-SAR) was determined from measurements of forward, reflected, and transmitted power. The animals were immobilized in polyethylene containers, and exposed tail to head to 0 (sham), 1 and 10 W/kg WBA-SAR, resulting in a local SAR to the testes about 2 - 2.5 times higher than the WBA. Increases in colonic temperature after exposure did not exceed 0.5^{o}C. In the 1st series of experiments the spermatogenic cell population was synchronized by prior administration of hydroxyurea and Trenimon (Oud et al., 1979). The animals were exposed for 3 consecutive days timed so, as to expose spermatocytes I over initial prophase, early pachytene, mid- and late pachytene, or diplotene of P I. The animals were sacrificed when the cells reached the metaphase of the first meiotic division (M I), i.e. 10, 7, 4 or 1 day after termination of exposures. Chromosome preparations from testes were made as described previously (Manikowska-Czerska et al., 1985), and M I plates were examined for aberrations.

In the 2nd and 3 rd series mice were exposed for 2 weeks, six days a week, 10 animals per exposure level, a total of 60. A heritable translocations test (HTT) was performed on animals of the 2nd series (Manikowska-Czerska et al., 1984). The males were mated with unexposed females during the 4th and 5th weeks after termination of exposure, i.e. at the time cells exposed during PI reached the ejaculate (Oakberg, 1956). Their male offspring was kept until sexual maturation and caged for a week with 2 unexposed females. Males which produced normal size litters were

rejected, the remaining ones were again caged with another pair of unexposed females, the selection procedure was repeated, and the males were mated once more. Males selected after the second mating, were mated again. Males which proved sterile or produced litters of reduced size over 3 consecutive matings were sacrificed, and their testes were examined for chromosomal aberrations. A dominant lethal test (DLT) was performed on animals of the 3rd series (Czerski et al., 1985). After termination of exposure the males were caged with successive pairs of unexposed females for consecutive 3-day periods over 9 weeks. Pregnant females were dissected the 14±1 day of gestation and the numbers of total, live, and dead implants, and of resorptions were determined. The significance of data was examined using multivariate analysis of variance techniques and Duncan's multiple range test by Dr P.M.Silverman of CDRH/FDA.

RESULTS AND DISCUSSION.

In the first series of experiments at least 20 M I plates were analysed per animal. No chromosomal aberrations were seen in 16 sham exposed males,i.e. in a total of 328 M I plates. After exposure at 1 W/kg aberrations (multivalent configurations) were seen in 7 out of 8 males, 4 exposed during initial prophase (leptotene/zygotene) and 4 exposed during early pachytene. 8 M I plates out of 168 were abnormal. In 8 animals exposed during later stages (4 at mid- and late pachytene, and 4 at diplotene) no aberrations were seen in a total of 185 M I plates. Following exposures at 10 W/kg during initial prophase and early pachytene aberrations were seen in 7 out of 8 males, i.e. in 11 out of 178 M I plates. After exposure during mid- and late pachytene multivalent configurations were seen in 1 male out of 4 (1 M I plate out of 94). A similar result was obtained following exposure during diplotene: multivalent configurations were seen in 1 male out of 4, i.e. in 1 M I plate out of 96. At this level of exposure aberrations were seen in 9 out of 16 animals, i.e. in 13 plates out of 368. The small number of animals precludes a meaningful statistical analysis, and the test should be evaluated as a qualitative one. In these terms, it seems that early prophase, and mid- and late pachytene are more susceptible to induction of chromosome abnormalities by microwave exposure, than are later P I stages.

In the second series (HTT) 10 sham exposed males sired 174 F_1 males, 4 of these were selected for cytological examination of testes based on the outcome of 3 matings. No chromosomal abnormalities were found. 10 males exposed at 1 W/kg sired 148 F_1 males, 7 of these were selected for cytological examination, and no aberrations were detected. 10 animals exposed at 10 W/kg sired 119 males, 5 translocation carriers were observed among 11 F_1 males selected because of sterility or semisterility. This result indicates that translocations induced by exposure can be recovered from offspring of irradiated males. Translocations carriers occurr among laboratory mice naturally, Leonard (1972) reports their incidence as 2.3 to 9.1 per 10 000 animals.

In the third series the overall pregnancy rate was about 60%. No significant differences between the sham exposed and microwave exposed groups were seen in DLT results following matings during the 1st, 3rd, 6th, and 8th weeks after termination of exposures. The test was not carried out during the 7th week. The results obtained following matings during the remaining weeks are shown in the table below:

Males exposed at (W/kg)	Week	N° litters	Total implants*	Fractional postimplantation loss*
0	2	29	14.8±0.40	0.09±0.017
1	2	30	13.0±0.67^{x}	0.14±0.038
10	2	41	12.5±0.57^{x}	0.12±0.024
0	4	38	13.7±0.52	0.06±0.013
1	4	41	13.3±0.48	0.17±0.022^{x}
10	4	39	12.2±0.83	0.19±0.032^{x}
0	5	37	15.1±0.51	0.05±0.009
1	5	34	13.4±0.42^{x}	0.15±0.026^{x}
10	5	39	14.9±0.40	0.11±0.018^{x}
0	9	36	15.1±0.47	0.07±0.012
1	9	38	15.1±0.43	0.11±0.015
10	9	34	14.1±0.52	0.14±0.037^{x}

* mean ± standard error, x difference significant at 0.05 level when compared with the sham (0 W/kg).

The preimplantation loss (decrease in total implants) noted the 2nd week indicates that late spermatid stages were affected. The increase in postimplantation loss observed the 4th and 5th week indicates effects on P I, and thus is in agreement with the results obtained in series 1 and 2, and in previous experiments (Manikowska et al., 1979, Manikowska-Czerska et al., 1985). There is no obvious dose-dependence of the effects. The 9th week, however, an increase in postimplantation loss was seen only after exposure to the higher SAR.

Only a few reports on effects of microwaves on male meiosis are available. Goud et al. (1982) exposed Swiss albino to 170 mW/cm^2 for 70 seconds and found effects on sperm morphology, and an increased postimplantation loss during 3 weeks after exposure. These results are in agreement with our observations. However, Beechey et al.(1986) attempted to verify the results reported by us earlier (Manikowska-Czerska et al., 1985) exposing male hybrid C3H/HeH mice to 2450 MHz radiation amplitude modulated at 100 Hz in an anechoic chamber using the same exposure regime WBA-SARs as in our experiments. The mice were restrained, and could not change their position. The authors did not indicate, however, whether the animals were exposed head to tail or tail to head which significantly affects the dose to the testes, the difference between those two positions being about a factor of 4

to 5 in local SAR. Chromosomal aberrations consisting in exchanges and presence of fragments were more frequent in the group exposed at the highest level (20 W/kg) when in other groups and shams combined, but did not exceed the incidence in shams alone. Thus Beechey et al.(1986) did not confirm our findings. Apart from a difference in exposure conditions (100 Hz amplitude modulation), a different strain of mice was used. It is well known (Lovell et al., 1987) that various mouse strains respond differently to chemical germ cells mutagens, and this may also be the case with effects of microwaves. Kowalczuk et al. (1983) and Saunders et al.(1983) exposed rear halves of male C3H mice for 30 minutes to 42-44 W/kg 2450 MHz and found later sperm morphology alterations. The DLT revealed a significant increase in preimplantation loss during weeks 2, 3, 5, 6, and 8 post exposure, the pregnancy rate was reduced to 10-13% of control values during weeks 4 and 5. Postimplantation loss did not increase. However it was reduced during weeks 4 and 5, possibly indicating that aberrations induced were so severe as to lead to early embryonal death.

In conclusion, it seems that single high level or repeated 2450 MHz exposure may induce chromosomal aberrations in male germ cells, spermatocyte I (early prophase I) and late postmeiotic cells being the most sensitive stages. More dosimetric studies and exploration strain and species specific differences in reponses are needed, before any extrapolations are made, and conclusions about the relevance of these findings to the human situatiuon are drawn (see Czerski and Davis, 1987).

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COLLECTIVE DOSES IN SWEDEN AFTER THE CHERNOBYL ACCIDENT CALCULATION FOR INHALATION AND EXTERNAL IRRADIATION

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Introduction

The Chernobyl fallout over Sweden was essentially deposited between the 28 and 30 of April 1986. The radioactive material was dryly deposited with a rather even spatial distribution in most parts of the contry, but in the eastern and middle-northern part of Sweden the plume was washed out by rain during the 29 of April. This caused a quite high and uneven distribution. A second rainfall caused a lower increase along the southwest coast of Sweden on the 8 and 9 of May. A large number of cloud and ground radiation measurements were performed during the first two months after the accident. These measurements have been used to calculate the collective dose from inhalation and external radiation to the Swedish population. Here a short review of the measurements and calculational methods is given as well as an estimated sum of the collective effective dose equivalent.

Measurements of ground deposition

The cesium deposition has been mapped by aerial sodium-iodide spectrometric measurements by the Swedish Geological Co (Lindén and Mellander, 1986). Measurement heights were 60 or 150 m and the distances between flight lines were 2, 4, 10 and 50 km, depending on deposition density and variation. A deposition map for Cs-137 constructed from measurement data is given in fig 2. For the dose calculations, Cs-134 rather than Cs-137 is chosen as a reference nuclide in the aerial measurements, because it has a gamma energy which does not interfere too much with the gamma lines from natural nuclides or with the activity of old Cs-137 from nuclear fallout. The activity ratio of Cs-134 to Cs-137 was initially 0.6 : 1, so the Cs-137 deposition could be calculated from the measured Cs-134 values.

In situ high resolution gamma spectrometric measuremetswere performed 1 m above ground at about 50 locations by the National Defence Research Institute. These measurement were used to gain information on the nuclide composition in different regions and as a base for calibration of the aerial measurements. The individual activities and dose rates for the deposited nuclides Zr-95, Nb-95, Mo-99, Ru-103, Ru-106, Ag-110m, Sb-125, Te-129m, I-131, Te-132, I-133, Cs-134, Cs-136, Cs-137, Ba-140, Ce-141, Ce-144, Nd-147 and Np-239 were calculated. From these individual dose rate values, the ratio of total dose from all nuclides to the activity of Cs-134 was calculated for all measurement points. The ratio varies between 0.095 - 0.3 mSv per kBq/m^2 of Cs-134 for the first year, depending on geographical location and wet or dry deposition.

Measurements of air concentrations

The concentration of radionuclides in ground level air was measured at seven air filter stations by the National Defence Research Institute. These stations, Kiruna, Umeå, Östersund, Stockholm, Grindsjön (near Stockholm), Göteborg, and Ljungbyhed (near Malmö) are in continuous use for monitoring particulate air radionuclide concentration. Inhalation and external cloud doses for different regions were calculated from measured values. Gaseous components, such as gaseous iodine and noble gases could not be measured this way, since they penatrate the

air filter. For gaseous iodine, a correction factor of 5 has been applied in the dose calculation in order to include this component. This factor is based on measurements performed at one of the Swedish nuclear power stations. (Ingemansson, 1986). Noble gases and pure alfa- or beta emitters, such as Xe-133, Pu-239, and Sr-90 are not included.

Calculational metods

The effective dose equivalent from inhalation was calculated from the time integrated air concentrations using dose conversion factors from ICRP 30. These are valid for inhalation of 1 μm amad aerosol by adults. The collective dose for each of the 24 different counties of Sweden was calculated by selecting the dose equivalent for the nearest air filter station and multiplying by the population number.

Dose conversion factors for calculation of external cloud and ground effective dose were taken from Kocher, 1983. Kocher gives values which do not include daughters, but these have been added separately when the dose contribution from the daughter was larger than 1 %.

For calculation of collective dose from ground, the different deposition values from the aerial measurements of Cs-134 were used as main reference. For all counties, the number of people living within different deposition intervals were counted. The intervals used were 0-0.5, 0.5-1, 1-2, 2-5, 5-10, 10-15, 15-20, 20-25, 25-30, 30-35, 35-40, 40-45, 45-50, 50-55, and 55-60 kBq/m^2. Since the depth distribution in the soil is unknown, the deposition values are given as equivalent surface deposition density, which means that the numerical values correspond to the values which are obtained by using a calibration factor for an infinite plane surface area without any depth penetration. Since the cesium actually has penetrated into the ground, the real deposition values are higher. How much higher the values actually are, depend on soil type and the extent of agriculture cultivation. Analysis of depth distributions from a few samples taken in the wet deposition area indicate that the relaxation length for undisturbed soils is somewhat less than 1 cm. A value of 1 cm is used for all counties in the present work.

Other nuclides than cesium, have been taken into account, by using the in situ measurements of total effective dose to cesium deposition ratio for representative parts of all counties. The collective dose from ground within a county can then be calculated by the expression:

$$C = \sum_i f_{shield} \cdot f_{snow} \cdot g_{depth} \, N_i \cdot (D_{total}/A_{Cs\text{-}134}) \cdot (A_{i+1,Cs\text{-}134} + A_{i,Cs\text{-}134})/2$$

where N_i is the number of people living within a Cs-134 deposition interval,

$A_{i,Cs\text{-}134}$ is the lower limit of the interval,

D_{total}/A_{134} is the ratio of total dose to deposition of Cs-134,

f_{shield} is the time and population averaged shielding factor for dwellings

f_{snow} is the time averaged shielding factor for snow cover, and

g_{depth} is a correction factor for pentration into the ground.

The values for individual counties are added to obtain the collective dose for total Sweden. Separate calculations are made for the first year, where the depth penetration is assumed to be exponentially distributed whith a mean relaxation length of 1 cm and for the following years where the relaxation length is assumed to be 3 cm according to the proposal of UNSCEAR, 1982.

Table 1
Collective effective dose equivalent in Sweden
from inhalation, cloud and external ground irradiation

Calculated collective dose values are based on measurements of gamma emitting nuclides at six air filter stations, in situ high resolution gamma-spectrometric measurements and aerial sodium iodide spectrometry. The ground activity is assumed to follow an exponential depth distribution with a relaxation length of 1 cm during the first year (0-1 a) and 3 cm during the following 49 years (1-50 a). Time and population averaged shielding factors for buildings and snow cover are applied.

County	Population	Collective dose equivalent			
		Inhalation and cloud manSv	Ground 0-1 a manSv	Ground 1-50 a manSv	Total 0-50 a manSv
AB Stockholm	1562490.	25.	83.3	156.	264.
C Uppsala	250762.	4.01	86.8	520.	610.
D Södermanland	250515.	4.01	25.6	153.	183.
E Östergötland	392887.	6.29	16.8	95.1	118.
F Jönköping	300924.	4.81	8.59	33.6	47.
G Kronoberg	174265.	2.79	2.49	9.74	15.
H Kalmar	239380.	3.82	11.	43.4	58.2
I Gotland	56203.	0.899	7.51	20.	28.4
K Blekinge	151562.	2.42	1.96	8.96	13.3
L Kristianstad	280330.	2.02	4.78	16.6	23.4
M Malmöhus	747140.	5.39	10.8	47.5	63.7
N Halland	238347.	0.81	2.94	14.6	18.4
O Göteborg-Bohus	712078.	2.42	26.4	131.	160.
P Älvsborg	426325.	1.45	15.9	71.	88.4
R Skaraborg	270382.	0.919	3.72	16.6	21.2
S Värmland	280499.	0.954	17.	43.4	61.3
T Örebro	270961.	4.34	21.8	85.2	111.
U Västmanland	254691.	4.07	54.2	363.	422.
W Kopparberg	285113.	4.56	18.1	74.4	97.1
X Gävleborg	289217.	4.63	126.	747.	877.
Y Västernorrland	262072.	1.08	144.	958.	1100.
Z Jämtland	134731.	0.556	30.3	123.	154.
AC Västerbotten	245181.	0.463	51.3	337.	389.
BD Norrbotten	263684.	0.311	6.4	16.5	23.2
Total Sweden	8339739.	88.	777.7	4085.	4951.

Table 2

Collective effective dose equivalents from inhalation and external cloud and ground irradiation for different assumptions of the relaxation length of the exponential depth distribution during the first year (0-1 a). The relaxation length for the following years (1-50 a) is assumed to be 3 in all cases.

Relaxation length first year cm	0 - 1 a manSv	1 - 50 a manSv	0 - 50 a manSv
0	729.3	1967.	2696.
0.1	763.1	2371.	3134.
1	865.8	4085.	4951.
2	895.4	5231.	6126
3	965.7	6681.	7646.

Results

The collective effective dose equivalents in different counties and total Sweden for the first year and for the first fifty years after Chernobyl are given in table 1. For inhalation and cloud, the collective dose is around 100 manSv. For external radiation from ground the collective dose is estimated to be around 800 manSv for the first year, and 5000 manSv for the first fifty years. From foodstuffs, the collective dose is reported to be around 1000 manSv for fifty years (Holmberg et al, 1987). This adds up to a total of approximately 6000 manSv from all radiation sources. Using a risk figure of 0.02 fatal cancers per sievert, the Chernobyl fallout will cause roughly 100 - 200 cases of fatal cancers in Sweden over a fifty year period.

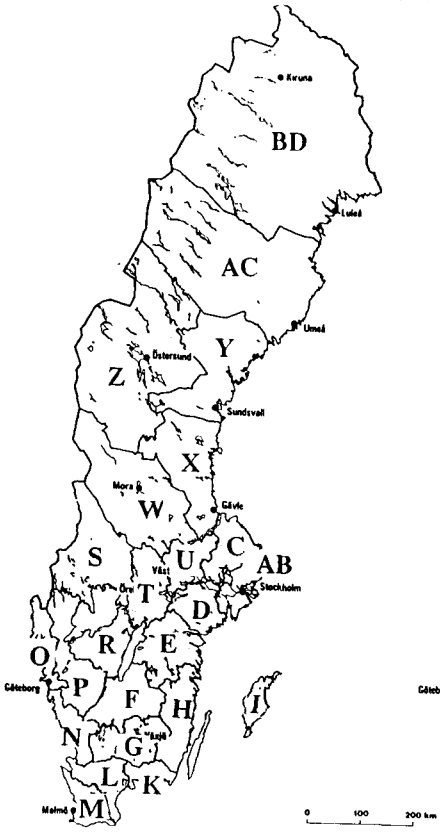

Fig 1. Map showing the different counties of Sweden.

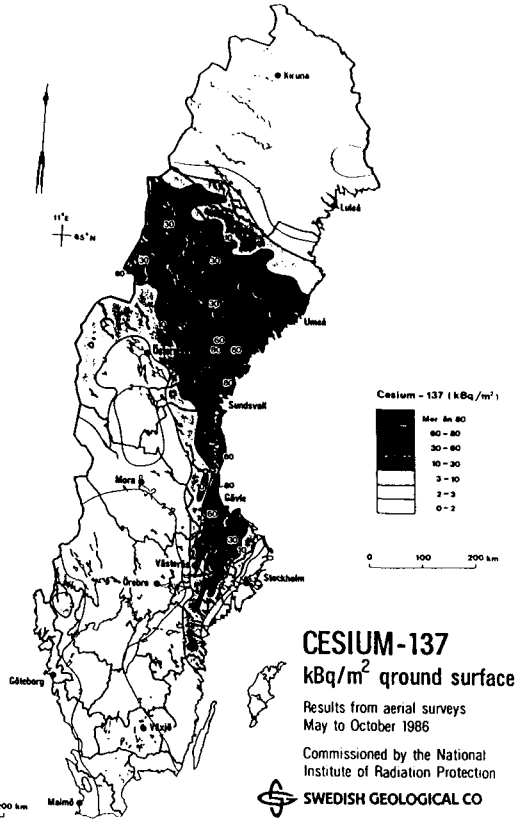

Fig 2. Cs-137 equivalent surface deposition based on aerial measurements.

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RADIATION PROTECTION IN THE MINISTRY OF NUCLEAR INDUSTRY

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I. PREFACE

The complete system of fuel cycle has been established and operated in China. The two nuclear power plants are under construction, and more and more radioisotopes are produced and used year by year. Since founding the Ministry of Nuclear Industry much attention has been paid to radiation protection by our government. The principles "Safety First, Quality First" and "Radiation Protection Leading the Way" have been worked out and implemented strictly, and a series of necessary protection measures taken, so that no acute radiation sickness has occurred for near 30 years. A large amount of radioactive waste has been handled, treated and conditioned, and it has been proven through environmental survey that no serious pollution has taken place. Following the issue of ICRP Publication No. 26 a new radiation dose limitation system has been adopted and occupational and public, individual and collective, radiation doses have been reduced gradually.

II. RADIATION PROTECTION MANAGEMENT

(1) Regulations

The first national radiation protection regulation was issued in 1960 and a revised edition was published in 1974. Since then, the Ministry of Nuclear Industry has issued a series of safety regulations; Table 1 shows the title of these regulations. We began revision of the national radiation protection regulation in 1981; the final edition was completed in 1986 and submitted to the government for approval. The basic contents of this regulation is the same as IAEA safety series No. 9 "The Basic Safety Standards for Radiation Protection", but contains more detail relatting to optimization and assessment of radiation protection. Table 2 shows the title of regulations to be issued.

TABLE 1
RADIATION PROTECTION REGULATION (ISSUED)

1. Regulations for Radiological Protection of Nuclear Fuel Element
2. Regulations for Radiological Protection of Reactors
3. Regulations for Criticality Safety of Fissile Material Processing
4. Regulations for Criticality Safety of Fissile Material Storage and Transport
5. Regulations for Occupational Radiation Exposure Records
6. Regulations for Monitoring and Data Report of Personnel

TABLE 2
RADIATION PROTECTION REGULATION (TO BE ISSUED)

1. General Regulations for Quality Assessment of Nuclear Radiation Environment
2. Regulations for Safe Transportation of Radioactive Materials
3. Safety Standards for Package of Radioactive Solid Waste with Low- and Intermediate-Level
4. Regulations for Radioactive Waste Management
5. Regulations for Radiation Protection in Production of Radioactive Isotopes
6. Regulations for Short-Term Storage of Radioactive Waste
7. General Quality Assurance for Ionization Radiation Monitoring
8. Regulations for Medical Supervision for Workers in Nuclear Power Plant
9. Emergency Planning and Preparedness for the Accident of Nuclear Power Plant

(2) The Organization

The Bureau of Safety, Protection and Health, MNI was set up in 1960 in order to manage research for radiation protection, to protect the environment and to establish and implement radiation protection programmes. The infrastructure of the safety and radiation protection authority, MNI is shown in Fig. 1. The Bureau is divided into five divisions:

- Administrative Office,
- Nuclear Safety and Protection,
- Technological Safety,
- Environment Protection,
- Radiation Hygiene and Health.

There are several institutes engaging in research relating to radiation protection.

1) The Institute of Radiation Protection was founded in 1962. This is a multidisciplinary institute which covers radiation protection measurement, radiation dosimetry, environment protection, radioactive waste management, standardization of radiation protection, radiomedicine, radiobiology, radioecology etc.
2) The Institute of Atomic Energy situated in Beijing is another research unit in the field of radiation protection and nuclear safety, which covers radiation protection, environmental protection, radioactive waste management, nuclear safety, criticality safety etc.

In addition to these, each nuclear factory, research and design unit has its own department of radiation protection.

In order to strengthen the administration of radiation protection, the Ministry of Nuclear Industry decided to set up four centres in the Institute of Radiation Protection in 1986:

- Personnel dose monitoring service and assessment,
- Environmental monitoring and assessment,
- Calibration of radiation protection instruments,
- Occupational disease registration.

The tasks of these centres are as follows:

- Collect and analyse the records of occupational individual and public exposure, arising from the release of liquid and airborne effluents, solid waste and occupational disease cases,
- Write an annual report on radiation safety and submit it to the Ministry of Nuclear Industry,
- Ensure quality assurance of radiation protection measurements,
- Conduct environmental surveillance and individual radiation exposure monitoring.

III. RADIATION PROTECTION AND ENVIRONMENT ASSESSMENT

For all nuclear facilities, radiation monitoring should be performed in accordance with an appropriate radiation monitoring programme worked out on the principle of radiation protection optimization.

(1) Personnel Monitoring [1]

The personnel monitoring devices used to measure external radiation exposure are X-ray film badges in the 1960s, both film badges and fluorescent glass in the 1970s and TLD for about one third of monitored workers. Internal dose monitoring commenced in 1963. The major nuclides measured were uranium and plutonium, iodine and cesium in some cases, as well as radon and its daughters in uranium mining and milling. The internal monitoring techniques are not yet complete; major methods are bioassay and whole body counting. Routine monitoring measures uranium or tritium in the excreta of the workers; the methods recommended by ICRU publication 30 are used to estimate the internal dose. The average intake may be calculated by using data relating to the concentration of aerosols in the respiratory zone of the working place.

The annual average dose equivalent from external exposure during 1959-1984 is shown in Fig. 2. From 1968-1976 the annual average dose equivalent was much higher than other years because radiation management was not good during that period.

The total number of workers monitored is 60 000 man.year. The collective dose equivalent from external radiation is about 600 man.Sv. The annual average individual dose equivalent is 9.2 mSv.

As personnel monitoring and management, and protective measures are improved, collective and individual doses are reduced. A trend has shown that the doses to occupational workers are gradually decreasing. The annual average equivalent was less than 5 mSv, during 1981-1984 and the annual average committed effective dose equivalent from uranium and tritium was less than 5 mSv. The results are shown in Table 3.

(2) Environment Monitoring and Assessment [2]

The requirements for environment protection in siting nuclear facilities are considered to be an important factor. In order to minimize the population dose, the sites of nuclear facilities are selected as far as possible in areas with small populations. Each nuclear facility and research unit has an environment protection department, responsible for monitoring and assessment. The ranges of monitoring areas are from 10 to 20 kilometres. All nuclear units are required to investigate the background radiation levels before the nuclear facility begins operation.

In order to assess the impact of the nuclear industry completely and systematically we have a programme of environmental assessment; the programme began in 1980. Results show that the annual average dose of critical groups is well below 5 mSv; 77 per cent are below 0.25 mSv. The collective dose within 80 kilometres of sites are about 0.01% of the collective dose due to natural radiation.

Most nuclear facilities are located in the Yangzi river basin and members of the public are concerned about the possibility of impact. Comprehensive investigations were carried out twice in the 1970s and in 1983 [3,4]. The main results are shown in Table 4 and these indicate that the Nuclear Industry has no impact on Yangzi River.

IV. CONCLUSION

China's policies for controlling occupational exposure and release of radioactive materials into the environment have proved effective. From the viewpoint of optimization of radiation protection there are many problems that need to be researched. In order to strengthen radiation protection and nuclear safety, we should improve the scientific management of radiation safety, clarify the goals of radiation safety, establish a data bank of radiation safety information and enhance research related to risk analysis and management, safety system engineering and optimization of radiation protection.

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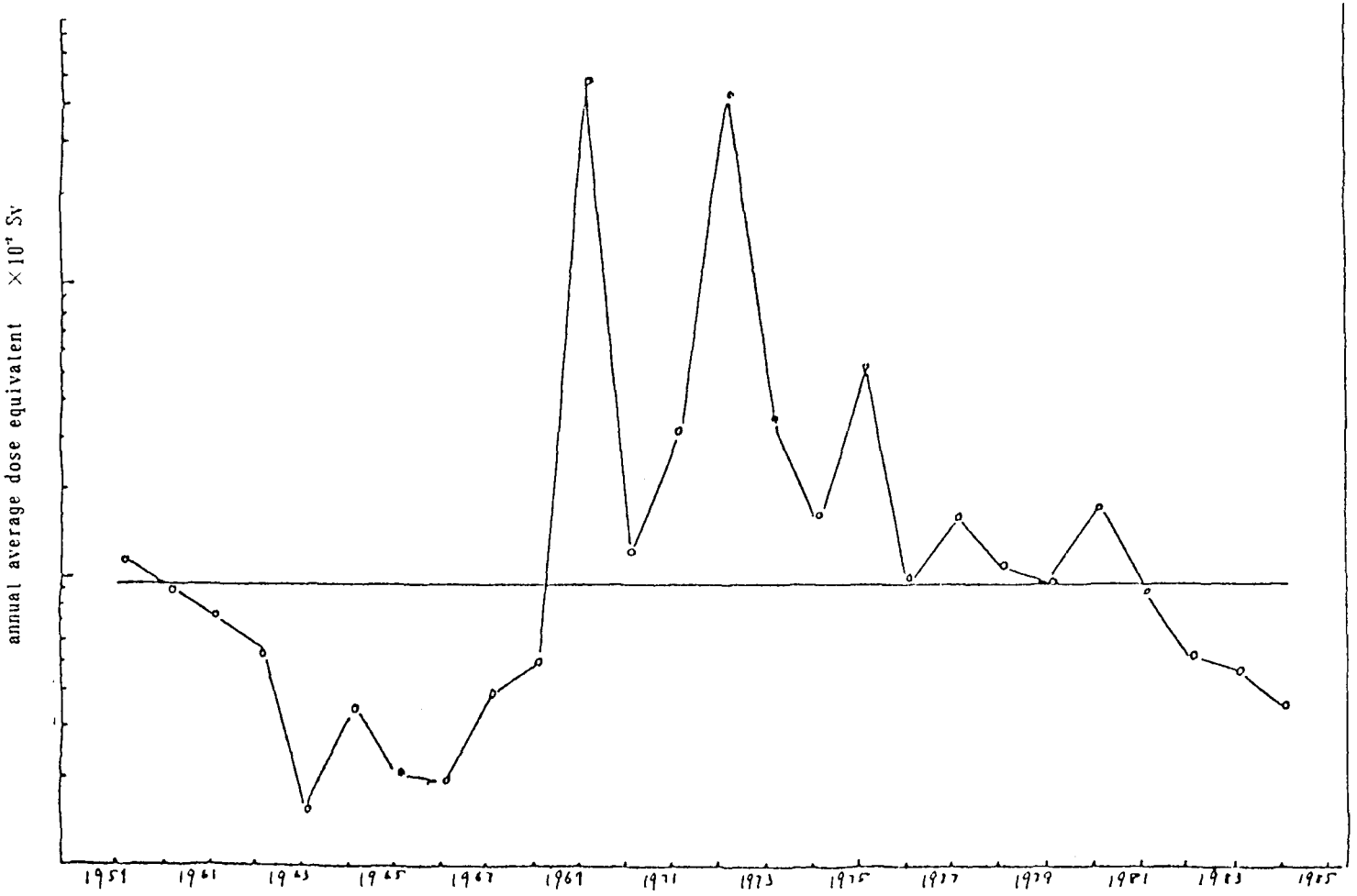

Fig. 2 The annual average dose equivalent from external exposure during 1959-1984

TABLE 3. INDIVIDUAL EXTERNAL DOSE EQUIVALENT
AND DISTRIBUTION FOR OCCUPATIONAL WORKERS IN 1980—1984

year	total number	collective dose equivalent S (man.Sv)	average dose equivalent per a.person H ($\times 10^{-4}$)	⩽ 5 mSv number	%	5-15 mSv number	%	15-25 mSv number	%	25-50 mSv number	%	⟩ 50 mSv number	%
1980	5112	60.49	1.16	2984	57.3	1244	23.9	370	7.10	331	6.35	283	5.43
1981	5356	46.92	0.88	3206	59.9	1357	25.3	361	6.74	302	5.64	130	2.43
1982	5485	29.26	0.53	3780	68.9	1225	22.3	296	5.40	165	3.01	19	0.35
1983	5658	27.83	0.49	3930	69.5	1327	24.7	261	4.61	135	2.39	5	0.09
1984	6322	23.62	0.37	4915	77.7	1112	17.6	207	3.27	74	1.17	14	0.02

TABLE 4 RADIOACTIVE LEVEL OF WATER SAMPLE IN YANDZI RIVER

nuclides		average level 1979-1980	average level 1983-1984
total α	Bq/l	0.059	0.063
total β	Bq/l	0.104	0.104
U	Bq/l	0.86	0.86
Th	Bq/l	0.17	0.15
^{226}Ra	Bq/l	0.001	0.010
^{3}H	Bq/l	10.36	8.039
^{137}Cs	Bq/l	0.0002	0.0006
^{90}Sr	Bq/l	0.0089	0.0118

AN APPROACH TO CONTROLLING RADIATION EXPOSURES OF PROBABILITIES LESS THAN ONE

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1. INTRODUCTION

The radiation protection system recommended by the International Commission on Radiological Protection (ICRP) [1] has been incorporated into many national and international regulations, including the IAEA/ILO/WHO/OECD-NEA[b] "Basic Safety Standards for Radiation Protection" (BSS) [2]. The system, however, is intended only for application to the preplanned control[c] of exposures which are expected to occur with certainty as a result of practices involving radiation sources. Such exposures, resulting from the so-called 'normal' operation of the sources, are therefore assumed to occur with a probability of unity. They are usually called 'normal' exposures, although sometimes they should be more properly referred to as exposures resulting from planned occurrences of events concerning radiation sources.

Rather different are those situations in which there is only a potential for radiation exposure, i.e. the probability that the exposure actually occurs is less than one. These situations are usually called 'accidental', not meaning that they necessarily arise from accidents but that exposures may or may not occur as a result of probabilistic events concerning a radiation source. The exposures that may potentially arise from such situations have frequently been called 'probabilistic' exposures. The ICRP system cannot be applied in its current form to the control of either the magnitude or the probability of occurrence of 'probabilistic' exposures and a much needed international consensus on a coherent and consistent system of control seems to be lacking.

(a) Mr. Ahmed and Mr. Gonzalez were Scientific Secretary and Chairman respectively of the IAEA Advisory Group on the Application of the Principles of Radiation Protection to Sources Potentially Causing Radiation Exposure. This paper, however, expresses the authors' personal opinions and should not be interpreted as indicating the Advisory Group's position on the subject.

(b) International Atomic Energy Agency/International Labour Organization/World Health Organization/Nuclear Energy Agency of the Organization for Economic Co-operation and Development.

(c) The term control is used to mean the act or fact of exercising restraint, rather than checking, testing or verifying.

The IAEA has a project to develop guidelines for a unified approach to the application of radiation protection principles to both radiation exposures occurring with certainty and exposures which are not certain to occur, and a consultative document has already been prepared [3]. The present paper briefly summarizes some aspects of the progress made towards this unified approach to radiation safety.

2. STATUS QUO

2.1 TERMINOLOGY AND SOME CONCEPTUAL PROBLEMS

When dealing with probabilistic situations, some concepts, such as risk, uncertainty and probability, are used with different connotations by different authors. Consideration of radiation exposures cannot escape this problem and the imprecise usage of the term radiation 'risk' is a good example. The IAEA had adopted [2] the ICRP usage of the concept of risk as the probability of harm resulting from a given radiation dose [4] and precisely defined risk in the IAEA Radiation Protection Glossary [5] as "the probability that a given individual will incur any given deleterious stochastic effect as a result of radiation exposure". However, the IAEA's Nuclear Safety Standards (NUSS) [6] subsequently used the term risk "in the general sense of a combination of probability and consequences of an event", i.e. in a sense of a mathematical expectation of harm rather than a probability of harm. However, in the limited area of radioactive waste disposal, both the ICRP and the OECD/NEA have used the term 'risk' to mean the product of the probability of exposure and the probability that the dose received will produce serious health effects [7] [8], i.e. as a probability of harm resulting from convolution of two stochastic phenomena. Risk has also been used in the literature in its imprecise colloquial meaning, and also as the equivalent of consequences.

The concept of probability itself is also being used with different connotations, either based on the classical frequentistic interpretation or in the more modern subjective formulation. This in turn has had an effect on the interpretation of the meaning of uncertainty.

Many of the current conceptual problems which appear when dealing with 'normal' and 'probabilistic' exposures are linked to the imprecise use of terminology.

2.2. SCENARIOS OF EXPOSURE

Effectively all radiation sources may cause 'normal' exposures and have a potential to deliver 'probabilistic' exposures. As a simple example showing the coexistence of both exposure scenarios, consider a radiation generator enclosed in a radiotherapy room fitted with an interlocked entry system. On the one hand, people will be exposed to the radiation that penetrates the shielding during normal operation. On the other hand, if the interlock fails, then someone might enter the radiotherapy room when the generator is operating and thus receive an unplanned exposure.

More sophisticated examples of normal and probabilistic scenarios of exposure can be given for nuclear installations. The relative importance of each scenario may naturally differ enormously for different sources and installations but in principle both modes should always be considered. The coexistence of both types of exposures seems to indicate that a unified approach for the radiation safety of the source is needed in order to ensure coherence and consistency in the safety objectives.

2.3. CURRENT PRINCIPLES OF RADIATION PROTECTION

The current system of radiation protection is intended for planning radiation protection for 'normal' exposures. It includes two types of requirements: (i) the individual related requirement of the limitation of individual doses; and (ii) the source related requirements of justification of practices and optimization of protection.

Individual dose limits have traditionally been recommended by ICRP and were incorporated in the BSSs. They apply to the combined exposure due to all sources, excluding that due to natural background radiation and the medical exposure of patients. For members of the public the ICRP recommends a dose limit of 1 mSv for exposures committed in any single year [9]. Taking into account the currently used hypothesis of a linear non-threshold dose-response relationship with a risk factor of approximately 10^{-2} per sievert, an individual dose limit of 1 mSv per annum implies a constraint of less than about 10^{-5} $year^{-1}$ on the risk incurred by the individual.

Since individuals may be subject to exposures due to several sources, in order to ensure compliance with individual dose limits, both the ICRP [4] and IAEA [10] have suggested the setting of dose upper bounds to be assigned to particular sources of exposures. Although specific dose upper bounds have not yet been recommended, it is assumed that they should be established by national authorities and applied to exposures resulting from the normal operation of a radiation source or installation whenever the exposures are assumed to occur with certainty. De facto dose upper bounds used by national authorities suggest that the constraint imposed in practice on individual risk due to a single radiation source is of the order of 10^{-6} per annum.

Optimization of protection, however, requires that the radiation protection applied to a source of exposure must be optimized in order that all doses be kept as low as reasonably achievable, economic and social factors being taken into account (ALARA). The application of the optimization requirement implies in practice that the actual risk incurred is much lower than the corresponding upper bound. Optimization requires an evaluation of the various possible options for protection and a judgement of their different features against preference criteria. The features to which these criteria apply include the achievements in protection, such as reductions of doses and favourable changes in their distribution in time and level of dose, and the efforts, such as costs and difficulties, required in achieving such protection. Since some of these criteria may be in conflict with others, evaluation for other than the

simplest prob'ems will require some kind of decision aiding technique to differentiate between alternative options from the point of view of radiation protection. One particular technique recommended by the ICRP is cost-benefit analysis [11], but it has been emphasized that this is only one way of quantifying some of the inputs to the optimization decisions. Other techniques, such as multiattribute analysis, are also being investigated by the ICRP.

Finally, the requirement of justification of practices provides that no practice shall be adopted unless its introduction yields a positive net benefit. A practice, such as the generation of electrical energy by nuclear fission or the radiosterilization of medical products, means the sum of all processes, industrial operations and actions associated with that activity which produces the benefit. The net benefit should be determined by assessing both benefits and efforts, including the possible harm due to radiation, resulting from the introduction of the practice.

2.4. CONTRASTING SAFETY PRINCIPLES

In contrast to the common international approach for controlling 'normal' exposures, there seems to be a lack of unified procedures for controlling the likelihood of 'probabilistic' exposures. There are, however, implicit procedures for the assessment and control of 'accidental' situations at some installations, notably nuclear reactors. They have been developed in parallel with, and to some extent separately from, the principles of radiation protection. It is therefore not surprising that in practice different principles are applied to exposures presumed to be certain and to potential exposures.

An example of these contrasting approaches is given by the treatment of the safety of nuclear reactors. The study of nuclear reactor safety developed from that of conventional safety and initially inherited the essentially deterministic concepts of that discipline. Engineering safety standards were set either as a result of experiments and tests or more subjectively using engineering judgement. A concept of 'maximum credible accident'- then called 'design basis accident'- was developed by means of such a deterministic approach and engineering safety features were designed to cope with this accident in an attempt to ensure absolute safety. Time and experience showed, however, that there remained some probability of accidents occurring and not being coped with by the safety related engineering systems. As a result, the nuclear accident scenarios - including beyond design basis accidents - their causes, probabilities and consequences, began to be studied by the more comprehensive technique of probabilistic safety assessment (PSA). In order to evaluate the results of PSA, it was found necessary to establish comparison criteria in the form of probabilistic safety criteria (PSC). One useful form of PSC is that of a limit on individual risk, which can be derived from the radiation protection principles, but this has not been the sole approach used. Rather, limits for 'societal' risk, expressed in terms of relationships between probability and number of people affected, have also been used as PSC. In any case, there has been a marked contrast between these various principles of nuclear safety and the radiation protection principles. These in turn lead to a situation in which the overall safety

objectives cannot be easily formulated and create a real conflict, particularly in situations in which a trade-off is unavoidable. For instance, in some cases an increase in the protection against exposures presumed to be certain may lead to a reduction in the safety measures for potential exposures, and vice versa; a typical example of this is the trade-off between occupational protection requirements and operational safety requirements such as maintenance and inspection.

For other sources the principles have also been in conflict. Procedures for assessment and control relating to waste disposal also started to evolve separately but have now been tackled by extending and developing the ICRP Basic Recommendations to deal with the particular problems of wastes [7][8]. In the uses of radioisotopes and radiation sources there is a contrast between the many standards and criteria used for normal operations and the few, if any, standards for preventing accidental exposures.

3. SUGGESTED APPROACH FOR A UNIFIED POLICY

There follows a brief summary of the suggested approach for a unified safety policy.

3.1. USAGE OF SOME RELEVANT CONCEPTS

It is suggested to use the concept of probability as a number between 0 and 1 assigned to the likelihood of the actual occurrence of an event, a radiation exposure due to the event, or a radiation effect due to the exposure. The number must be interpreted as a measure of the degree of belief that the event, exposure or effect will occur rather than as a measure of the actual frequency of occurrence. It must comply with the rules of coherence as follows: (i) the complement of an event with probability p should be assigned a probability (1-p); (ii) events which may occur at a greater frequency should be assigned a greater probability; and (iii) if event a is more probable than b, and b is more probable than c, then a must be more probable than c. This definition encompasses the usual frequentistic definition of probability but does not require statistical information, and is particularly tailored to be used for controlling 'probabilistic' exposures.

However, it is recommended that risk be defined as the probability that a serious detrimental health effect will occur to an individual exposed to either 'normal' or 'probabilistic' exposures. Risk is, therefore, the product of the probability of an event occurring and the probability of radiation exposure occurring given the event and the probability of harm given the exposure, because each one of these probabilities is assumed not to influence the others. For normal operation the probability of both the event and the 'normal' exposure occurring is assumed to be unity and the definition matches the usual meaning of risk for radiation protection purposes.

3.2. ASSUMING A RISK-DOSE RELATIONSHIP

When considering probabilistic exposures, the possibility of doses exceeding the linear (stochastic) region of the dose-response relationship and entering the 'non-stochastic' region must be taken into account. Figure 1 presents the extremely simplified scheme suggested for the relationship between the probability of suffering a severe harmful effect as a result of a radiation dose versus the incurred dose. At levels up to a fraction of a sievert only stochastic effects are assumed to occur, including fatal cancers in the irradiated individual and severe genetic effects in the succeeding generations of descendants; the risk factor being of the order of one in a hundred per sievert. For doses that exceed 0.1 Sv, say, delivered in a short period of time, non-stochastic effects may occur and at doses higher than about 5-10 Sv also delivered in a short period of time, practically all irradiated individuals will suffer an acute radiation syndrome and eventually die. As a simplified but practical approximation, therefore, the risk-dose relationship is assumed to be linear with a slope of 10^{-2} Sv^{-1} in the low dose region and to approach asymptotically a probability of unity for doses higher than about 5-10 Sv.

3.3. SOURCE CHARACTERIZATION

Radiation sources should be characterized from a safety point of view according to their potential for delivering harm. For normal exposures the relevant quantity characterizing the individual risk is the individual dose assumed to occur with certainty. For the low doses expected in normal operation, the probability of harm for the individual is assumed to be proportional to the incremental dose received and the collective dose is a measure of the total expected harm. For 'normal' exposures, therefore, individual and collective doses are the quantities which characterize the safety of the source.

For 'probabilistic' exposures, however, there can be identified a probability of individual harm, or risk, and a probability distribution of consequences. These quantities characterize the safety of the source in this case. The mathematical expectation of harm is not a good indicator of the safety of the source owing to the large uncertainties associated with low probability-high consequence events.

4. LIMITATION OF INDIVIDUAL RISK AS A BASIS FOR A UNIFIED APPROACH

Probabilistic events could be brought under a unified system of control in a way which is consistent with the current system of dose limitation. A possible method of incorporating accident situations into a risk based system would be to define a total individual risk limit for the combined sum of 'normal' and 'probabilistic' exposures. However, this approach has two disadvantages: firstly, it could be construed to imply an allowable trade-off of risk between normal and accident situations; secondly, such a method would involve changing the current system of dose limitation for 'normal' exposures.

The simplest method would be to define separate risk limits for 'probabilistic' exposures and to retain the current dose limits for 'normal' exposures. In the context of radioactive waste disposal the ICRP recommended for members of the public an annual risk limit of 10^{-5} to deal with probabilistic events [8], which is of the same magnitude as that implied by the dose limit for normal situations. The ICRP proposal aims to achieve compatibility with the current system of dose limitation and thereafter to improve safety by reducing doses below this limit by optimization. Thus, for consistency, it is suggested that the proposal be extended to the control of individual risk from 'probabilistic' exposures in general. This limit would apply to the individual risk in the most highly exposed critical group and would encompass all sources of probabilistic exposure.

In addition to the individual risk limit, there need to be alloted risk upper bounds in order to constrain the individual risk due to a single source. The risk upper bound should be apportioned from the risk limit. The apportionment may be different according to circumstances; thus a smaller fraction may be chosen for waste disposal than for nuclear power plants because of our uncertain knowledge about the variety of sources to which a future individual may be exposed. A risk upper bound is to be used in the design and regulation of a particular facility in the same manner as dose upper bounds are currently used.

Compliance with a risk limit or with a risk upper bound, either for a source or even for a single scenario, can also be shown by means of a criterion curve, which allows the use of dose distributions or doses and probabilities directly without the need to convert them to risks. An example of such a criterion curve is shown in Fig. 2 for an annual risk limit of 10^{-5}. The shape of the curve is derived from the dose-effect relationship of Fig. 1. Every point on the curve represents the same risk, that is 10^{-5} in a year. The curve starts at the point where the probability is one and the dose is about 10^{-3} Sv, where the risk is then 10^{-5}. Doses below 10^{-3} Sv correspond to a risk of less than 10^{-5} and therefore automatically comply with the risk limit. The first part of the curve is a straight line of slope 45°; it represents the region of doses up to about 1 Sv where the risk due to a given dose is assumed to be proportional to that dose. Thus a dose of 10^{-2} Sv occurring with a probability of 10^{-1} confers the same risk as a dose of 1 Sv occurring with a probability of 10^{-3}. From doses of 10 Sv or greater, from which death is certain to occur, the risk no longer depends on the dose but only on the probability. Since the risk limit is assumed to be 10^{-5} in a year, the criterion curve is then horizontal at an annual probability of 10^{-5}. In the short region between 1 Sv and 10 Sv the two straight parts of the curve are joined by some smooth path whose exact shape is irrelevant. If the probability distribution (given that its integral is normalized to one) or – in the simplified case of discrete doses – the number pair of the annual dose and the annual probability of that dose falls everywhere below the criterion curve, then compliance with the risk limit has been shown.

However, assuming arbitrarily simultaneous risks from ten different sources or scenarios, then the risk upper bound per source would be 10^{-6}

and the corresponding criterion curve of the same type of the described earlier will be that shown in Fig. 3. If the risk upper bound refers to the source, then the sum over all scenarios has to be used for the dose distribution; if, however, the risk upper bound has already been apportioned to a single scenario, then the dose distribution of that scenario is to be entered into the diagram with the criterion curve.

A basic conceptual policy question remains, however. This question is whether the certain occurrence of a dose with a given probability of causing death should be considered equivalent to the potential occurrence with this same numerical probability of a fatal dose. In other words, the question is whether the risk due to the certain occurrence of a dose with a given probability of causing death is equal to the risk due to a dose which has this same numerical probability of occurring and which would certainly be fatal. Quantitatively, the risks in the two cases are the same and, therefore, for risk limitation purposes the two situations have been assumed to be identical. This assumption is basic to the unified approach for limiting individual risk.

5. OPTIMIZATION OF RADIATION SAFETY

It seems reasonable to consider on the idea of a limit on individual risk as an essential requirement for a unified approach to radiation safety in general and, particularly, to the control of probabilistic exposures. However, the limit on individual risk should be viewed as a necessary but not sufficient condition for ensuring the appropriateness of the level of safety of a radiation source. The question remains for the responsible authorities whether that level should be improved further by taking into account, for instance, that a high number of individuals incurring an acceptably low probability of harm may still represent an unacceptably high expectation of harm. For exposures resulting from normal operation the basic requirement is that the radiation protection applied to the source must be optimized. Similarly, for accident situations the full assessment of the consequences must take into consideration the number of people affected and the level of harm to them, and the costs of and efforts required for improved safety. This aspect is sufficiently close to the ideas involved in the optimization of protection.

6. LIMITS ON SOCIETAL RISK OR JUSTIFICATION?

As indicated before, societal risk limits have been proposed and utilized for many aspects of radiation safety, for instance in relation to the probability and consequences of reactor accidents. Societal risk limits do not follow directly from the principles of protection developed by the ICRP, whose extension to probabilistic exposures has been suggested before. Conceptually, however, societal risk limits may have a logical connection with the ICRP principle of justification. This logical connection has still to be explored.

7. SUMMARY

The problems in dealing consistently and coherently with normal exposures assumed to occur with certainty and potential exposures of a

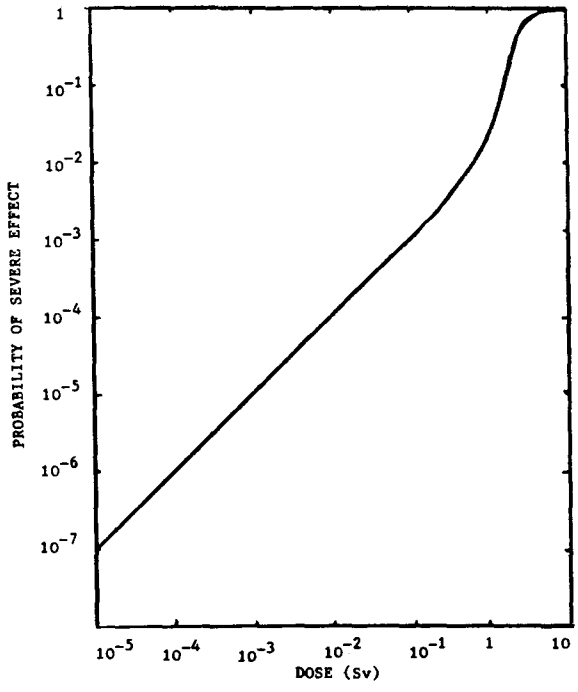

Fig.1 : INDIVIDUAL PROBABILITY OF SEVERE HARMFUL EFFECT VERSUS RADIATION DOSE

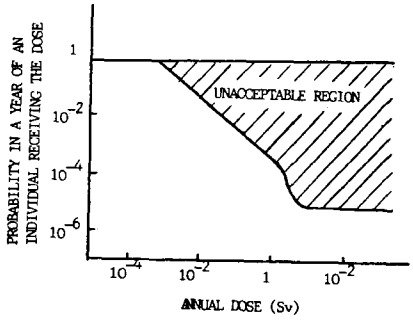

Fig. 3: CRITERION CURVE CORRESPONDING TO AN ANNUAL RISK CONSTRAINT OF 10^{-5} FROM ALL EVENTS

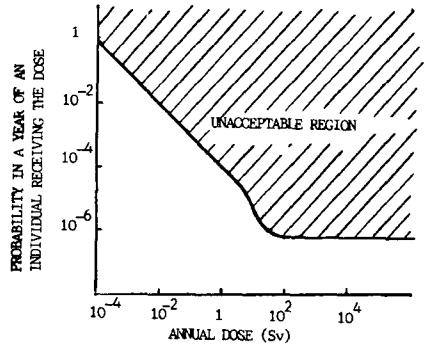

Fig. 3: CRITERION CURVE CORRESPONDING TO AN ANNUAL RISK CONSTRAINT OF 10^{-6} FROM ALL EVENTS

probabilistic nature have been explored. An IAEA project has suggested how a start might be made to work towards such a unified approach to radiation safety and has proposed the use of a risk based system. The paper briefly summarizes some aspects of the proposal.

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THE ROLE OF PROBABILISTIC EVENTS IN THE APPLICATION OF THE JUSTIFICATION CRITERION

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Probabilistic events (potential accidents) at large installations -specifially nuclear power plants- play a major role in the public debate. According to the ICRP criteria an installation is only justified if it results in a net benefit for society. It is logical and in accord with the importance assigned by public opinion that probabilistic events be included in an evaluation of its justification.

This entails an assessment of the probability of occurrence and the consequences of such events. However, even a crude evaluation of such events shows that if only casualties are considered and evaluated using a realistic monetary equivalent (e.g. 2 million DM per fatality), the resulting damage per unit of time is extremely low compared to the benefit of such an installation.

If one does not resort to the simple argument that the public debate is irrational, the only other explanation is that social effects play a larger role in such an evaluation than the effects on individuals.

We have attempted to evaluate these social effects by dividing the environs of a nuclear installation situated in a densely populated area into social units. Within 10 km of the plant, each community was considered to be a unit. Between 10 and 120 km counties were generally used as units. Therefore, the size of one unit varied from 1 000 to 75 000 inhabitants (communities) and from 90 000 to 1 400 000 (counties).

Impact on these social units is investigated on the basis of the following:

a) fraction of early fatalities

b) fraction of late fatalities

c) fraction of land to be evacuated (ground contamination)

d) fraction of land with long term consumption restrictions

e) fraction of land with short term consumption restrictions (contamination of leaves).

In pathway "a" the following function was used to calculate the probability of early death P_a for a person receiving a dose D (in Sv):

$$P_a = 1 - 0.5^{\left(\frac{D}{4}\right)^{6.6}+\left(\frac{D}{9.5}\right)^{4}+\left(\frac{D}{15}\right)^{10}}$$

The dose D was calculated for a given event (with a specific weather situation) for each location within the social unit and the resulting probability P_a was then multiplied by the number of poeple living at that location.

By summing up over all locations within that social unit, the fraction of population suffering early death as a consequence of the assumed event could thus be evaluated.

The probability of delayed death (pathway b) was calculated with the following equation: $P_b = 0.01\ D$

For the remaining three pathways limit values have been used to determine if a given location is considered inhabitable after an event or if restrictions on the consumption of foodstuff produced at that location have to be imposed.

A location was considered inhabitable if the dose accumulated within one year due to ground contamination would exceed 0.25 Sv. It is realized that the public opinion in Germany may possibly tend to use a more restrictive limit.

For the pathways d and e if was assumed, that foodstuff exceeding 2000 Bq/kg would not be allowed to be consumed.

In this way the fraction of the population and the fraction of land within one social unit being harmed by a specific event was obtained.

The corresponding social damage was evaluated as follows for each pathway:

A sigmoidal damage function was selected with such parameters that social damage is close to zero if only a small fraction of the inhabitants of a social unit is harmed. It approaches 1 if the fraction approaches 1. The time needed for a society to regenerate after a disaster follows a curve of similar shape. The function which we selected is shown in Figure 1.

For one event (assumed to happen in a specific weather situation) the social damage caused by each pathway was calculated for each social unit. It was then multiplied by an factor representing relative importance; namely, 1 for pathway a, 0.1 for pathway b, c and d and 0.01 for pathway e and added up for all 5 pathways to obtain the relative social harm for a given event.

In order to add up the societal harm of social units with a different number of inhabitants, this relative value was multiplied by a weighting factor equal to the population of the respective social unit.

This calculation was then performed for all weather conditions which had occurred in the past in one calendar year. Thus we obtained the probability distribution of the social harm as a consequence of a specific event and additionally the average social harm per event for each social unit.

Finally this avarage social harm was summed up over all social units to obtain the total social harm caused by the event.

This calculation was repeated for different release situations by variing the percentage of core inventory released to the atmosphere, the emission height and the duration of the release.

The results show that the societal harm usually increases if the release of a given percentage of core inventory is distributed over a longer period of time as compared to a release lasting about 1 hour only. This is explained by the fact, that changing wind directions result in more social units being harmed by the emission. This is only partly compensated by the diluting effect caused by a distrubition over a larger area specifically if the contamination of land stays above the set limit in both cases.

In order to test the applicability of such a probabilistic societal harm factor one has to attribute a monetary equivalent. The order of magnitude can be evaluated by applying this methodology to systems recently ordered in some countries for the mitigation of the consequences of a core melt accident acompanied by containment failure. Evidently such installations are considered "reasonable" in the eyes of the authorities.

The impact of such an installation on the individual and societal harm can be shown by comparing different release cases as shown in Figure 2. The results (Figure 3) have been tentatively evaluated using the same monetary equivalent for societal harm as for individual harm (2 million DM per unit of harm). It turns out that an installation with an annual cost (capital + operating cost) of 1 million DM is justified if it reduces the release of iodine and aerosols by a factor of 100 or more for events with a probability of occurence of more than 1.6×10^{-5} per year.

According to probabilistic risk assessment studies this is the order of magnitude expected for such events.

Therefore the inherent monetary equivalent for societal harm reflected by these requirements was in the order of 2 million DM per unit of harm. The individual harm alone caused by such events would not justify these additional installations.

The remaining social and individual costs of probabilistic events will then have to be included in a justification evaluation of the modified installation.

Monetary equivalents of much more than 2 million DM per unit of societal harm would have to be used if one wanted to argue that the impact of probabilistic events makes a nuclear installation of the evaluated design unjustified.

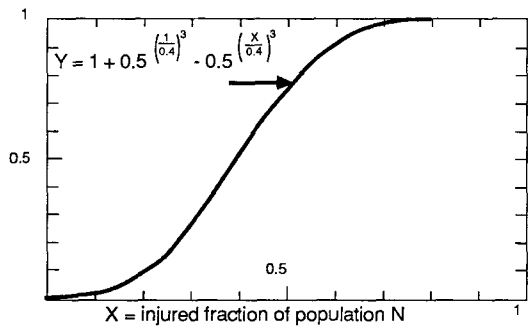

fig.1 **Societal Hazard = Y * N * I**

Case	Emission Height (m)	Duration (h)	% of Core Inventory released Noble Gases	Iodine	Cesium...
A	40	1, 8, 48	100	100	100
B	160	1, 8, 48	100	100	100
C	160	1, 8, 48	100	0.3	0.03
D	160	1, 8, 48	100	0.3	0.0003
E	160	1, 8, 48	100	0.003	0.0003

fig.2 **Analysed Release Events**

Case	Duration	Early Death	Delayed Death	Societal Damage
A	1	193 000	4 300	802 000
	8	337 000	13 400	1 357 000
	48	308 000	30 000	1 864 000
C	1	5	149	18 000
	8	0	149	37 000
	48	0	150	71 000
E	1	0	34	6 500
	8	0	34	8 300
	48	0	34	9 600

fig 3. Average Consequence per Event

FORMATION DU PERSONNEL POUR L'INFORMATION DU PUBLIC

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Après l'accident nucléaire de Tchernobyl, le public français a brusquement recherché toutes les informations qui pourraient l'aider à comprendre quels étaient les risques immédiats et différés.

L'information officielle, exacte mais froide, et souvent tardive, répondait mal aux attentes irrationnelles du public, les médias diffusaient une information rapide, sensationnelle mais confuse, émise sans contrôle de la fiabilité de ses sources.

Deux catégories d'intermédiaires privilégiées ont peu participé à l'information du public français :

- le personnel des centrales nucléaires et celui des centrales classiques, soit 24 000 personnes ;
- le corps médical et, d'une manière générale, les professions de santé, c'est-à-dire plus de 300 000 représentants en France.

Le public attend du personnel des centrales des réponses techniques sur la nature et la probabilité des risques que présente une centrale nucléaire qu'il est supposé bien connaître. Il peut également être rassuré par la confiance dont ce personnel témoigne, pour sa santé et celle de sa famille, vis-à-vis d'un accident et de ses conséquences.

Les professions de santé sont davantage interrogées sur les effets des rayonnements ainsi que sur les mesures sanitaires individuelles ou collectives à prendre en cas d'accident.

L'information dont doivent disposer les différents interlocuteurs du public est la même ; elle doit couvrir toute la phénoménologie d'un accident nucléaire depuis son origine technique jusqu'à ses conséquences sanitaires différées, mais la mise en forme de cette information doit nécessairement adopter les modes de pensée de chaque profession.

Electricité de France a pris la décision dès le mois de juin 1986 de donner une dimension exceptionnelle à son action d'information du public en assurant une véritable culture de son personnel dans le domaine accidentel, et en amplifiant sa participation dans l'information des membres des professions de santé.

Cette communication traite essentiellement de la formation du personnel car cette opération s'achève.

L'importance des effectifs impose de définir un mode de large diffusion du message puis d'adapter celui-ci en conséquence.

Comment diffuser l'information.

Deux principes ont été respectés :

- le personnel devait être formé par sa hiérarchie directe pour que le message essentiellement technique soit perçu comme une conviction acquise par les cadres en raison de leur compétence ;
- la plus grande initiative devait être laissée aux centrales pour organiser leur démultiplication interne.

Le mode de diffusion retenu a été la démultiplication à trois niveaux :

- Un premier niveau organisé au plan national en 16 sessions de 3 jours. Dans chaque session un spécialiste de chacun des

domaines abordés a rafraîchi et complété les connaissances d'environ 18 "grands formateurs". Les 288 "grands formateurs" ainsi formés étaient des cadres volontaires, sélectionnés au nombre de 4 ou 5 par centrale, pour leur compétence en radioprotection ou en sûreté ou encore en mesures dans l'environnement ou enfin des médecins du travail des centrales systématiquement associés à ces sessions du premier niveau.
- Un deuxième niveau organisé en centrale a permis aux grands formateurs de préparer 3000 cadres techniques et administratifs.
- Enfin avec le troisième niveau qui s'achève en centrale, les cadres développent pour leur personnel une "culture accidentelle", sans pouvoir utiliser la logique d'une approche scientifique.

Comment adapter le message. Les 2 premiers niveaux de démultiplication ont été organisés en fonction du troisième qui est la véritable finalité.

Les connaissances du personnel sur l'action des rayonnements étaient correctes mais limitées au milieu de travail, c'est-à-dire à l'irradiation et la contamination externe ; en effet l'interdiction de boire, de manger, de fumer, écarte le risque notable de contamination interne.

Par contre il avait en général une vision confuse des concepts de sûreté et même des idées partiellement erronées sur les conséquences d'un accident à l'extérieur de la centrale.

Le lancement de l'opération. Rapidement une brochure décrivant le déroulement général d'un accident et ses conséquences à l'extérieur de la centrale a été conçue pour l'ensemble du personnel.

Ce document visait à répondre rapidement aux attentes du personnel après Tchernobyl, à donner le temps d'organiser une "formation lourde" avec démultiplication à 3 niveaux, enfin à susciter des interrogations de la part du personnel pour enrichir le dialogue au cours du troisième niveau de démultiplication.

A posteriori cette plaquette résume la formation reçue et sert de documentation de référence.

La "formation lourde" à trois niveaux. La préparation du message à transmettre à travers cette action de formation, comme la rédaction de la brochure, s'est heurtée à plusieurs écueils :

- Au niveau mondial, la diversité des filières, les différences de conception en matière de sûreté, ne permettent pas de développer un modèle universel d'accident et de conséquences.
  Cette difficulté a été globalement résolue en distinguant :
  . La population habitant à moins de 10 kilomètres d'une centrale, pour laquelle un accident peut avoir à court terme une incidence directe mais connue car venant d'une centrale déterminée.
  . Le reste de la population, qu'un accident dans une centrale française ou étrangère atteindra indirectement, notamment par la commercialisation de denrées alimentaires contaminées.

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- La deuxième difficulté réside dans la quantification des risques. Celle-ci nécessite la connaissance des unités et des normes. Les médias utilisent des unités de l'ancien et du nouveau système. Avant tout ce sont des ordres de grandeur qu'il faut familiariser, si possible sans avoir recours à des unités sans signification pour le public.
  Il serait commode de rapporter toutes les valeurs quantifiées à une seule référence : la dose annuelle maximale admissible pour le corps entier.
  En principe cette unité qui est d'un emploi direct en cas d'irradiation externe, trouve son équivalence en contamination interne par les Limites Annuelles d'Incorporation raccordées par des limites dérivées à la contamination des aliments.
  Malheureusement l'arbitraire qui préside au plan international à l'établissement des limites commerciales, empêche toute présentation cohérente simple.

  Nous avons donc adopté une présentation à deux niveaux de complexité au choix de celui qui dispense l'information :
  - une présentation se limitant à la LAI considérée elle-même comme une unité à laquelle toutes les grandeurs citées se raccordent ;
  - un complément exprimant la LAI et les autres grandeurs en becquerels et en REM, unités les plus utilisées par les médias.

- La troisième grande difficulté est liée aux effets stochastiques sur l'organisme dus aux faibles doses de rayonnement.
  Face au grand nombre d'informations sans fondement rigoureux diffusées sur ce thème, nous avons simplement cherché à montrer le caractère purement mathématique de l'extrapolation linéaire considérée comme la limite supérieure d'un domaine d'incertitude bornée en valeur inférieure par une valeur nulle.

La forme du message. L'information à délivrer au personnel des centrales (comme celui destiné aux membres des professions de santé) visait à donner à ces interlocuteurs privilégiés du public, une culture minimale. Celle-ci leur permet de concevoir eux-mêmes et d'exprimer simplement les réponses circonstanciées à des questions infiniment variées en nature et en niveau.

La méthode consistant à préparer un jeu de questions-réponses pour en quelque sorte structurer les dialogues entre les médiateurs et le pulic, n'a pas été retenue en raison de son manque d'adaptabilité à l'auditoire et à la nature et à la variété des questions.

Les principaux thèmes traités. Les mêmes thèmes ont été traités dans la brochure et à chaque niveau de démultiplication de façon à parcourir la phénoménologie d'un accident selon un enchaînement logique partant des causes et se développant jusqu'aux conséquences tardives. L'accident de Tchernobyl a été présenté à tous les stades en insistant sur son caractère très majorant.

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- La sûreté. Tout en développant la conception française de barrières successives pour organiser une défense en profondeur, l'accent a été particulièrement mis sur le rôle du personnel pour la sûreté en exploitation.
  Par contre les approches probabilistes, trop théoriques, n'ont pas été abordées ; elles ne pourraient être un thème d'échange entre le personnel et le public.

- Les rejets. Ce thème fait apparaître la différence d'ordre de grandeur entre "l'inventaire du coeur" et le "terme source" en présentant les mécanismes spontanés de rétention des radioéléments complétés par la rétention des filtres de rejets gazeux, en cours d'installation.
  Cette information montre que des réacteurs de puissances comparables peuvent rejeter, en cas d'accident, des quantités de radioéléments très différentes. Le rôle prépondérant des rejets gazeux par rapport aux transferts de radioéléments par les eaux souterraines est également mis en évidence. Une mention particulière est accordée aux Iodes et aux Césiums.

- La dispersion atmosphérique. La situation privilégiée de la France et de l'Espagne lors de l'accident de Tchernobyl n'a pas toujours été comprise, voire admise. Une présentation des trajets de dispersion des rejets, très changeants en fonction du régime des vents, est toujours appréciée. L'importance du lavage d'un nuage radioactif par la pluie justifie les hétérogénéités de la contamination du sol entre deux régions voisines.

- L'organisation de la crise. L'importance du rôle de chaque membre du personnel en cas de crise dans une centrale nucléaire est mise en relief par la nécessaire cohérence des actions de chacun.
  Ce thème qui présente les organisations complémentaires d'Electricité de France et des Pouvoirs Publics a un effet rassurant particulièrement pour le personnel des centrales classiques.

- Tchernobyl. La présentation détaillée de l'accident permet d'en montrer le caractère exceptionnel.

Conclusions. Bien que l'opération ne soit pas terminée, elle apparaît comme une remarquable réussite qu'il faudra entretenir. Déjà des responsables régionaux d'EDF estiment nécessaire de l'étendre aux 90000 agents des centres de distribution en faisant "parrainer" chacun d'eux par une centrale nucléaire.

SUBJECT INDEX