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OCCUPATIONAL EXPOSURE TO IONIZING RADIATION

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Abstract

An overview of occupational exposure is presented. Concepts and quantities used for radiation protection are explained as well as the ICRP system of dose limitation. The risks correlated to the limits are discussed. However, the actual exposures are often much lower than the limits and the average risk in radiation work is comparable with the average risk in other safe occupations. Actual exposures in various occupations are presented and discussed.

Keywords (chosen by the author)

Occupational exposures, risks, concepts and quantities, ICRP

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OCCUPATIONAL EXPOSURE TO IONIZING RADIATION

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INTRODUCTION

Occupational exposure comprises all the dose equivalents and intakes of radioactive nuclides incurred by a worker during periods of work.

This definition should in principle include all workers because of their permanent exposure to the natural radiation. However, from a regulatory point of view there are reasons for not including exposure to "normal" natural radiation, but only those components of natural radiation that result from man-made activities or that arise in special environments. There is no sharp dividing line, however, between levels of "normal" natural radiation and modified and enhanced natural radiation caused by human activities. Examples of the latter are aviation, mining and other underground work, use of some phosphate fertilizers etc. In aviation the aircraft crew is exposed to the increased cosmic radiation with height, in underground work the natural radiation is enhanced by uncovering uranium or thorium and/or increasing the concentration of the natural radionuclides in air by the underground work. When using phospate fertilizers with a higher than normal content of natural radionuclides the workers are exposed to enhanced external and internal irradiation.

Irrespective of the exact definition of occupational exposure, it is easily concluded that the number of workers exposed to ionizing radiation increases continually in the world. The exact number is difficult to determine because of the different routines used for recording occupational exposures and accounting workers as radiation workers. This difficulty is caused by different approaches

to international recommendations on classification of workers and their registrations.

The relative number of radiation workers partly reflects the level of technical development and economic and social welfare of a country and in developed countries the number is at present of the order of 2 per 1 000 inhabitants and in developing countries of the order of 0.1 per 1 000 of population. The distribution of workers in different fields of occupation varies also but a major area of interest is always the medical field including doctors, nurses, laboratory personnel and dentists. Other fields are research institutions of universities and hospitals, industries where radiation is used as a tool in the industrial processes, radioisotope production etc. A field of increasing interest is the uranium fuel cycle. Even if there are national fluctuations in the economical and political interest in the nuclear field there is a general and global expansive trend of nuclear power. The installed world electrical generating capacity in 1983 was about 170 GW(e) with about 200 GW(e) under construction. In the year 2 000 an installed capacity of around 1 000 GW(e) is expected. At present about 10 % of the total electrical energy is produced by nuclear power and this is expected to increase to about 25 % by the year 2 000. The number of reactors in operation in 31 December 1982 was 298, and the number of workers involved was of the order of 100.000.

THE REGULATORY SYSTEM

The occupational exposure to ionizing radiation is regulated on the basis of international recommendations issued by ICRP. In order to understand and put the levels of exposure into perspective and possibly use the information for epidemiological purposes it is necessary to be familiar with the concepts and quantities used, the assumptions on the detrimental effects of radiation and the principles of the dose limitation system as recommended by ICRP. It is also necessary to be familiar with the monitoring and dose recording practice.

CONCEPTS AND QUANTITIES

The interaction of radiation with matter is described quantitatively by the energy deposited per unit mass. The energy deposition, the absorbed dose D, from all types of radiation is defined as

$$D = \frac{d\bar{\varepsilon}}{dm}$$

where $d\bar{\varepsilon}$ is the mean energy imparted by radiation in a volume element and dm is the mass of the volume element. The SI unit for absorbed dose is joule per kg and its special name is gray (Gy).

The quantity exposure has been used in the measurement of X- and gamma radiation. It is now used only as a quantity for reference standards. It is replaced by air kerma, K, defined by the relationship

$$K = \frac{dE_{tr}}{dm}$$

where dE_{tr} is the sum of the initial kinetic energies of all the charged ionizing particles liberated by uncharged ionizing particles in a material mass dm.

The biological effects do not depend solely on the energy deposition per unit mass, or absorbed dose, but also on the type of radiation. The correlation between dose and biological effect is also dependent on the type of biological effect and the distribution of the absorbed dose rate in time.

Effects of high absorbed doses that have a threshold below which they do not occur are called non-stochastic. For those effects for which there is no evidence of a threshold dose, such as cancer induction, it is known that if a group of people are irradiated then a proportion will show the effect, but there is no possibility of predicting which individuals will be affected. Therefore, an increase in the irradiation will only increase the probability for each individual that the effect will occur. These effects are called stochastic. For protection purposes it is assumed that the risk of stochastic effects from irradiation of a tissue is directly proportional to the absorbed dose in the tissue, although various other dose-response relationships have been observed.

With the assumption of a linear relationship between dose and its corresponding probability of effect the addition of doses implies corresponding addition of probabilities. In case of a non-linear relationship, e.g. a curvilinear, this will not be the case. For incremental doses, the incremental probability of a health effect corresponding to a small incremental dose will, for the curvilinear case, depend on the previous level of dose, whereas for the proportional case it is independent of other doses.

If the additional doses are not large in comparison with the pre-existing dose, the relevant portion of the curvilinear response can be approximated by a linear relationship. Small additional doses can then be treated independently of the pre-existing dose and of each other. This is the basis which underlies the use of risk factors to relate the incremental probability of a health effect to the incremental dose and these factors will be independent of the absolute level of dose.

To meet the need in radiation protection for a numerical relationship between the radiation exposure and its biological effects,

the quantity dose equivalent, H, is used. H is defined by

$$H = DQN$$

where Q is the quality factor and N is the product of all other modifying factors specified by ICRP. For the present the ICRP has assigned a value of unity to N. Both Q and N are dimensionless. The SI unit of dose equivalent is the same as for absorbed dose, namely Jkg^{-1}, and has been given the special name sievert (Sv). The earlier unit for dose equivalent was rem (1 rem = 0.01 Jkg^{-1}).

The quality factor Q allows for the different effectiveness of different types of radiation and its value has been defined as a function of the collision stopping power, L_∞ in water at the point of interest. ICRP has specified the relationship at a number of values of L_∞ as shown in Table 1. Other values can be obtained by linear interpolation.

Table 1. Specified relationship between Q and L_∞

L_∞ in water (kev μm^{-1})	Q
≤ 3.5	1
7	2
23	5
53	10
≥ 175	20

ICRP has also recommended approximate values for all common types of ionizing radiation; these are given in Table 2.

It is important to know that the dose equivalent should not be used to assess the consequences of accidental exposures in man which may involve high doses causing severe non-stochastic effects. For that purpose, absorbed dose is the appropriate quantity after weighting for the relative biological effectiveness (RBE) of each type of radiation. It is defined as the ratio of the absorbed dose of a reference radiation to the absorbed dose of a test radiation to produce the same level of biological effect of the same extent and/or nature, other conditions being equal.

Table 2. Recommended permissible approximation of quality factor for various types of radiation

Type of radiation	Approximate value of Q
X rays, gamma rays and electrons	1
Thermal neutrons	2.3
Neutrons, protons and singly charged particles of rest mass greater than one atomic mass unit of unknown energy	10*)
Alpha particles and multiply-charged particles (and particles of unknown charge) of unknown energy.	20

The probability of occurrence of a stochastic effect in an organ is assumed to be proportional to the dose equivalent in that particular organ. The proportionality factor differs for the various organs of the body. However, in assessing health detriment the total risk is usually required. If all the body is uniformly irradiated a single overall risk coefficient can be used. If the irradiation of different organs is non-uniform - as for instance in case of irradiation from many internally deposited radionuclides - a further quantity is necessary to represent the total risk.

ICRP has recommended a quantity for allowing for the different mortality risks associated with irradiation of different organs, together with a proportion of the hereditary effects. This quantity is defined by the sum:

$$\sum_T w_T H_T$$

where w_T is the weighting factor to represent the proportion of the stochastic risk resulting from irradiation of tissue T to the total risk when the whole body is irradiated uniformly and H_T is the mean dose equivalent in tissue T. This sum is called effective dose equivalent, H_E.

To assess the effective dose equivalent, the dose equivalent in each tissue from all sources is assessed and multiplied by the appropriate weighting factor and the resulting products are then summed. If all the tissues in the body were uniformly irradiated

*) This value has recently (in 1985) been increased by ICRP to 20 for neutrons.

the result would be numerically equivalent to the wholebody dose equivalent.

The values of w_T recommended by ICRP are shown in Table 3; they are appropriate for protection for individuals of all ages and both sexes, i.e. for workers and members of the public. The value for gonads includes serious hereditary effects, as expressed in the first two generations.

Table 3. Weighting factors recommended by ICRP for calculation of effective dose equivalent

Organ or tissue	Weighting Factor w_T
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder	0.30

In practice the "remainder" organs or tissues are taken to be the five not specifically listed in Table 3 that receive the highest dose equivalents;

The hands and forearms, the feet and ankles, the skin and the lens of the eye are not included in the "remainder", and these organs should therefore be excluded from the computation of effective dose equivalent.

The absorbed dose from external irradiation is delivered at the same time as the tissue is exposed to the radiation field. To take account of the time distribution of absorbed dose to internal organs caused by incorporated nuclides ICRP has defined the committed dose equivalent, which is the time integral of the dose equivalent rate in a particular organ following an intake of radioactive material into the body. The integration time is 50 years after intake, taken to correspond to a working lifetime. The formal

definition of committed dose equivalent is:

$$H_{50} = \int_{t_0}^{t_0+50y} \dot{H}(t)dt$$

for a single intake of activity at time t_0 where $\dot{H}(t)$ is the relevant dose equivalent rate in an organ or tissue at time t.

With the assumption that the effect is directly proportional to dose equivalent it is sometimes useful to define a quantity to measure the total radiation exposure of a group of individuals. This quantity is called the collective dose equivalent, and is given by

$$S = \int_0^\infty H\ N(H)\ dH$$

where N(H) dH is the number of individuals receiving a dose equivalent between H and H + dH; or by

$$S = \sum_i \bar{H}_i\ N(\bar{H})_i$$

where $N(\bar{H})_i$ is the number of individuals in a population subgroup receiving an average dose equivalent of $\bar{H}_i$.

If the dose is expressed in terms of effective dose equivalents the resultant definitions give the collective effective dose equivalent, S_E.

In most practical situations the collective dose equivalent is obtained by summing doses received over a specified time period, often 1 year. However, the collective committed effective dose equivalent from intakes of radionuclides in that year includes the 50 year integration of dose equivalent rates in the relevant organs resulting from the intakes.

The per caput effective dose equivalent is the average of a range of actual dose equivalents to individuals. It is obtained by dividing the collective dose equivalent over a given time in a specified population by the number of individuals in the population at the time, or, more directly, by calculating the average absorbed dose rate, or intake of radionuclides, from the source, and hence the average dose equivalent or committed dose equivalent.

The integration of the per caput dose from a practice given as a function of time gives the dose equivalent commitment, H_c, defined by

$$H_c = \int_0^\infty \dot{\bar{H}}\ (t)\ dt$$

where $\bar{\dot{H}}(t)$ is the per caput dose equivalent rate as a function of time. If the upper time limit of integration is infinity the resultant quantity is the dose equivalent commitment; if the integration is terminated at a time T, then the resultant quantity is named the truncated or incomplete dose equivalent commitment.

If the practice continues at the same rate and all other relevant factors are constant it can be shown that the maximum future per caput dose equivalent rate per unit practice will, at equilibrium, be numerically equal to the dose equivalent commitment per unit practice. This provides a means for estimation of the maximum future annual per caput dose equivalent from a continued practice.

Similarly, if a practice continues at the same rate for T years, the maximum future per caput dose equivalent rate is equal to the truncated per caput dose equivalent commitment.

A source or practice will also give rise to a collective effective dose equivalent rate which varies with time. By integration the collective effective dose equivalent commitment is achieved.

$$S_{E.c} = \int_0^\infty \dot{S}_E(t)\, dt$$

The collective effective dose equivalent commitment is associated with the total health detriment caused by a given decision or a practice. It can therefore by used for justification or optimization studies.

ICRP SYSTEM OF DOSE LIMITATION

ICRP has recommended a system of dose limitation in terms of three components:

1. The justification of the practice

   No practice shall be adopted unless its introduction produces a positive net benefit.

2. The optimization of radiation protection

   All exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account.

3. The dose limits for individuals

   The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances by ICRP.

Justification

The expression "positive net benefit" invokes the idea of cost-benefit analysis. However, the choice between practices will depend on many factors, only some of which will be associated with radiation protection. Therefore, more general decision-making methodologies would need to be applied to decisions on the justification of practices. In any case there is a need to ensure that the total detriment from a practice is appropriately small in relation to the expected benefit of the practice.

Optimization

Since any exposure to radiation is assumed to involve some degree of risk, all exposures should be kept as low as reasonably achievable, i.e. the protection should be optimized.

The optimization applies for doses below the dose limits and therefore non-stochastic effects are precluded. For the stochastic effects the mathematical expectation of the amount of harm in an exposed group of people is proportional to the collective effective dose equivalent.

The selection of a level of protection that meets the optimization requirements includes consideration of several factors. Several methods can be used for this decision-making, such as multicriteria methods, which compare by pairs the various options, aggregative methods which combine the values of the criteria in each option into a single value ranking the results from different options in order to select the best. The most normal aggregative methods are based on utility functions to quantify the criteria. An important method based on utility functions is the cost-benefit analysis which expresses the criteria in monetary terms.

The basis of cost-benefit procedures for radiation protection optimization is that an option is selected if the resulting net benefit exceeds that of the next best alternative and not otherwise. The net benefit can be expressed as $B = V - P - (X + Y)$, where B is the net benefit from the practice, V is the gross benefit, P is the production cost excluding the cost of protection, X is the cost of achieving the selected level of radiation protection and Y is the cost of the detriment. The optimization requirement means minimizing the sum of the cost of protection X and the cost of detriment Y.

The cost of protection and the detriment is dependent on the level of protection which can be characterized by a protection parameter, W. A protection parameter can be ventilation rate, thickness of lead etc. The objective of optimization is to find the "optimized" value W when $X(W) + Y(W)$ is minimum i.e.

$$\left.\frac{dX(W)}{dW}\right|_{W_o} = - \left.\frac{dY(W)}{dW}\right|_{W_o}$$

This procedure is called differential cost-benefit analysis. The level of exposure in the optimized situation as defined by this equation is such that a marginal increase in the cost of radiation protection is exactly balanced by a marginal reduction in the cost of detriment.

If X(W) and Y(W) have only discrete values, the optimized level is reached by a step-by-step improvement of the protection. The decision to go from one level of control, A, to a more costly one, B, would then be taken if

$$\frac{\left[X(W_A) - X(W_B)\right]}{\left[Y(W_A) - Y(W_B)\right]} < 1$$

In that case, optimization can be achieved through an iteration process, testing increasingly higher levels of radiation protection and stopping when the expression above no longer holds.

The optimization procedure necessitates expression of cost of protection and detriment in the same unit. Since the health detriment is proportional to the collective effective dose commitment S, it is also assumed that the cost of the health detriment also is proportional to that quantity so that

$$Y = \alpha \cdot S$$

where α is a dimensional constant expressing the cost assigned to the unit collective dose for radiation protection purposes. If other detriments such as anxiety are quantified and corresponding costs are taken into account, they should be added.

Thus the optimization may be expressed as

$$X(W) + \alpha \cdot S(W) = \text{minimum}$$

which at the optimum W corresponds to the expression

$$\left.\frac{dX}{dW}\right|_{W_o} = - \alpha \left.\frac{dS}{dW}\right|_{W_o}$$

or

$$\left.\frac{dX}{dS}\right|_{S_o} = - \alpha$$

If the functions are discontinuous, the expression takes the form

$$- \frac{X_B - X_A}{S_B - S_A} \leq \alpha$$

where A and B stand for different levels of protection.

There are some limiting factors in the optimization. If the optimization procedure leads to excessively high individual doses, the maximum acceptable individual dose is the limiting condition. If the optimization will be too expensive, available resources will be limiting. In these cases, a cost-effectiveness analysis may be appropriate, this is an analysis of how a given level of protection can be achieved at the minimum cost or how the maximum protection can be obtained at a given cost-level.

The cost of health detriment in monetary units is the cost the society is willing to pay to avoid this detriment. This cost is expressed as $\alpha \cdot S$. The value of $\alpha \cdot S$ is a controversial matter. The collective dose multiplied by the risk factor gives the statistical prediction of the number of cancers and hereditary effects in a population as a result of the exposure. If a given value could be assigned to each statistical death, the value of alpha could be estimated by division of that value by the relevant risk factor.

Several attempts have been made to define a value for the collective dose (expressed as man Sv) and the values have ranged from \$ 1,000 $manSv^{-1}$ to \$ 100,000 $manSv^{-1}$.

Dose limits

A limit is the value of a quantity which must not be exceeded.

The limits used in radiation protection are:

(1) Primary dose equivalent limits. The primary limits relate to the dose equivalent, effective dose equivalent, committed dose equivalent or committed effective dose equivalent, depending on the exposure circumstances. These limits apply to an individual or, in the case of exposure of the public, to the critical group, i.e. the most exposed group.

(2) Secondary limits are needed when the primary dose limits cannot be applied directly. In the case of external exposure, secondary limits may be expressed in terms of the so-called dose equivalent index. In the case of internal exposure, secondary limits may be expressed in terms of annual limits on intake.

(3) Derived limits are related to the primary limits by a defined model such that if the derived limits are observed, it is likely that the primary limits would also be observed.

(4) Authorized limits are limits for any quantity specified by the competent authority or by the management of an installation (operational limits). These should generally be lower than the primary or derived limits.

Furthermore, reference levels are used in radiation protection. A reference level is the value of a quantity which is used to determine a particular course of action. It is not a limit. Examples of reference levels are:

(1) Recording level is a level of dose or intake above which the information is of sufficient interest from a radiation protection point of view to be worth recording and keeping.

(2) Investigation level is a level of dose or intake above which the results are considered sufficiently important to justify further investigations.

(3) Intervention levels are usually specified for use in abnormal situations. If the value of a quantity does not exceed or is not predicted to exceed the intervention level, then it is highly improbable that intervention will be warranted.

The limit for the annual effective dose equivalent for workers is 50 mSv. In addition, the annual dose equivalent limit for individual organs and tissues of workers is 500 mSv except in the case of the lens of the eye for which the limit is 150 mSv. For women of reproductive capacity any necessary exposure should be as uniformly distributed in time as is practicable. The purpose of this is the protection of the embryo before a pregnancy is known. When a woman is known to be pregnant she should work only in such working conditions where it is most unlikely that the annual exposures will exceed three-tenths of the dose equivalent limits.

The dose equivalent from external radiation is delivered at the same time as the radiation is received and the dose equivalent from external radiation received during a year can therefore be compared with the limit. However, the dose equivalent from an intake of radioactive materials may be spread over future years, and in this case it is the committed effective dose equivalent that must be compared with the limit.

For comparison of the committed effective dose equivalent from intakes with the dose equivalent limits, secondary limits for individual radionuclides giving the maximum intake in a year can be used. The secondary limits are known as annual limits on intake

(ALI); they correspond to the committed effective dose equivalent from an intake of a given radionuclide equal to the dose equivalent limit for workers. Keeping the intakes in each year less than the ALI ensures that the maximum annual dose equivalent from that radionuclide will always be less than the dose equivalent limit even if intake continues every year for 50 years.

When both external and internal exposures are received in a year, the annual dose limit will not be exceeded if the following condition is met:

$$\frac{H_E}{50\ (mSv)} + \Sigma \frac{I_j}{I_{j,L}} \leq 1$$

where H_E is the actual annual effective dose equivalent, I_j is the annual intake of radionuclide j and I_{jL} is the annual limit on intake for radionuclide j. Sometimes it is necessary to explicitly require that the condition

$$\frac{H_{sk}}{500\ (mSv)} \leq 1$$

shall also be met where H_{sk} is the shallow dose equivalent index (∿ dose to the skin).

In some situations derived air concentrations (DAC) are of practical use. These are obtained by dividing the ALI by a standard volume of air inhaled in a working year of 2000 hours at a breathing rate of 1.2 m^3h^{-1}.

THE RISKS CORRELATED TO THE LIMITS

The objective of radiation protection is to prevent the non-stochastic effects and to limit the probability of stochastic effects to an acceptable level. That means that even if the dose is very small there is still a small probability of a detrimental health effect. This effect can be genetic and somatic. The average risk factor for hereditary effects, as expressed in the first two generations, is for radiation protection purposes given by ICRP to be about $4 \cdot 10^{-3}$ Sv^{-1}. The additional damage to later generations is of the same magnitude.

The somatic risks are distributed in time after each exposure. After the time of exposure the probability of a detrimental effect increases from approximately zero for a few years (= latent period) up to a maximum value and then declines to zero again. The cancers can be divided into leukemia and all other cancers because of their different latent periods. For leukemia the average annual risk is about $0.8 \cdot 10^{-4}$/year per Sv during 25 years which leads to a lifetime risk of mortality from leukemia of $2 \cdot 10^{-3}$ per Sv. For other

cancers the shape of the risk curve is less certain but if the average annual risk is taken to be $2 \cdot 10^{-4}$ per Sv during 40 years this corresponds to a lifetime risk of $8 \cdot 10^{-3}$ per Sv. The total risk of leukemia and other cancers is therefore about 10^{-2} per Sv.

If there is a continuous exposure of 10^{-2} Sv/year the annual risk will continuously increase to reach a level of 10^{-4}/year after several tens of years. The accumulated risk after 50 years of exposure will be 0.24 % eventually rising to 0.5 %.

The limit for the effective dose equivalent for workers is 50 mSv/year. A continuous exposure of 50 mSv/year during 50 years will accordingly result in an annual risk of $5 \cdot 10^{-4}$/year and a lifetime accumulated risk of 1-3 %.

In the ICRP recommendations it is said that a valid method for judging the acceptability of the level of risk in radiation work is to compare this risk with the risks in other occupations known to have a high standard of safety.

The risk for radiation workers and other workers can be compared in various ways. One way is to compare the average risks, another to compare the risk to maximum exposed groups. This is illustrated in Table 4.

Table 4. Comparison of risks

Causes	Average risk		Maximum risk	
	Radiation workers	Other workers	Radiation workers	Other workers
Accidents	A_1	A_2	A_3	A_4
Occupational diseases	O_1	O_2	O_3	O_4
Total	$T_1=A_1+O_1$	$T_2=A_2+O_2$	$T_3 \neq A_3+O_3$	$T_4 \neq A_4+O_4$

In a comparison between two occupations the average accident risks for radiation workers should be compared with average accident risk for workers in another occupation, i.e. A_1 and A_2 respectively. Similarly O_1 should be compared with O_2 and T_1 with T_2. In radiation work A_1 is small and O_1 is well indentified and quantifiable. In other work there is good statistical data on the accidental risks, A_2, but the knowledge of cause and effect relationship for

occupational diseases is very limited. The ideal comparison mentioned above can therefore hardly be carried out and in its publication 26 ICRP has therefore compared the risk of occupational disease caused by radiation with the risk of fatal accidents in other occupations. This risk comparison relates to O_1 and A_2.

Occupations recognized as having a high standard of safety are those in which the average annual mortality risk due to occupational hazards is less than 10^{-4}. In these occupations fatalities are accompanied by a much larger number of less severe consequences and in a comparison account should be taken of all components of harm, both fatal and non-fatal effects.

In radiation work it is very unlikely that workers will get an annual dose of 50 mSv, year after year. Because of optimization and good radiation protection practice the actual average doses to workers are less, and often much less. Considering this it is concluded that the average risk in radiation work is comparable with the average risk in other safe occupations. The actual exposures in various occupations are presented and discussed below.

Another comparison of interest would be to compare the risk profile of various occupations, i.e. to include a comparison of the maximum risk groups, identified in Table 4 by A_3, O_3, T_3 and A_4, O_4, T_4. The maximum total risk T_3 and T_4 is not the sum of the two components because they probably do not represent the same individuals. In radiation work there is a great deal of information on the dose distribution to workers and those groups which are regularly more exposed than others are easily identified. In other work, however, the corresponding information is very poor and often non-existent and it is therefore difficult to make any comparison between radiation workers and other workers on this basis. On the other hand, comparisons can be made between different kinds of radiation occupations.

ICRP's comparison with average doses and risks has occasionally been misunderstood to mean a recommended limit of the average dose to a worker over a working lifetime, in other words a recommendation of a limit for the lifetime dose. That was not the intention of ICRP. It is also to be observed that if such a limit should be introduced it would interfere with the rights of an individual to follow the career of his choice. Statement of that meaning was made by the Committee on Radiation Protection and Public Health, NEA, OECD, in 1982.

TRAINING AND ADMINISTRATIVE CONTROL

The control of occupational exposure is maintained by optimization of the radiation protection in the design and the operation. This also includes education and training of the workers in radia-

tion protection. The workers are thereby aware of the potential health hazards in their work and how these can be reduced by safe working methods and techniques. Even if the main safety should be built into the design, the importance of well educated and trained personnel should not be underestimated. Failures of the protection system can occasionally be caused by so-called "human factor", i.e. unplanned, irrational or undisciplined actions against everything learnt and regulated. This is minimized by applying good ergonomic principles in the planning of design and equipment. However, even when this has been done there are several examples of incidents or accidents caused by the human factor and some of them have led to very serious consequencies.

An education and training program could also, hopefully, increase the workers awareness of the limited reliability of all technical safety systems. Also here there are several examples of severe accidents when there has been a failure in a safety system, e.g. when a failure of the interlock system to a gamma radiography exposure room has not been observed by the worker.

Other methods to improve the radiation protection are more of an administrative nature. Examples are classification and marking of working areas with varying need for supervision and control, record-keeping of workplace monitoring and persondosimetry results to follow up the routine working conditions and special events or doses of interest etc. All these results would be used for improvements of the radiation protection by optimization procedures. Investigation levels are used to initiate special investigations. These can be triggered by measurements or predicted doses. For instance, there might be an investigation level for the expected collective dose caused by planned work. If the predicted collective dose is higher than the investigation level the employer must report and discuss the radiation protection problem with the responsible authority. Examples of such levels of the order of 0.1 manSv exist.

MANAGEMENT OF OVEREXPOSED WORKERS

ICRP's recommended dose limits in publ. 26 do not apply to accidents and emergencies but only to those conditions where the source of exposure is under control. However, ICRP gives some recommendations and guidance on the planning of actions or countermeasures after an accident. These recommendations and guides only state general principles because of the individual nature of the decisions to be made for worker involved in the accident. Recently, the Committee on Radiation Protection and Public Health, NEA, OECD, made a study among member states on their regulations and/or experience of measures taken in case of over-exposure of workers. In their answers the following common elements could be identified.

(a) Notification

Provisions are made for the notification of the worker, immediate responsible person, employer, medical advisor, and responsible authority under specified conditions, when a suspected overexposure occurs.

(b) Assessment of dose and immediate medical attention

Immediate action is required to be taken to assess the dose incurred and communicate this information to the medical advisor.

(c) Investigation

A detailed investigation is required to identify and record the circumstances causing the overexposure and recommendations are made for corrective action if indicated, to ensure that recurrence of the circumstances causing the accident is unlikely. An important element in the corrective action is to ensure that information developed in the investigationis widely disseminated so that other activities can profit from the experience.

(d) Recording of exposures

Requirements are included for the recording of the dose by the employer, the medical advisor and, in some cases, a central state authority.

(e) Medical followup and future employment restrictions

Medical followup and future employment restrictions are handled on a case-by-case basis through consultations between the employer, worker, medical advisor and competent authority. In one case there would be generic requirements in regulations that following an overexposure the individual must not be assigned to tasks which are likely to result in the dose equivalent exceeding one percent of the annual limit during the remainder of the calendar year in which the exposure occurred.

MONITORING AND DOSE ASSESSMENTS

The radiation protection principles are in general applied by the following steps.

1. An annual individual limit (50 mSv) is established.

2. A derived limit corresponding to the annual individual limit is established. That could be a dose rate, a concentration of radionuclides in air or water, a level of contaminations of surfaces etc.

3. Optimization of the protection is carried out.

4. The optimized level of protection is expressed in terms of quality of design, conditions on operation, external dose rate, activity concentration in air etc. If these levels are expressed as limits they are called authorized limits.

5. An operational level might be established as a reference level for special actions.

Monitoring is carried out to check the compliance with given regulations. The results of monitoring are also sometimes directly or indirectly used for dose assessment. The monitoring includes measurements on air, water surfaces and man and concerns external and internal irradiation. The measurements are made continuously or occasionally on a regular or irregular basis. The results of the measurements are impaired by inaccuracies of varying significance. The external exposure of man is normally measured by a personal dosemeter. TLD and film dosemeters are used for external dosimetry. The film dosemeter has the advantage of preserving its information on dose and can be examined repeatedly afterwards. TLD is easier and more rapid to handle in an automatic system for dose measurements and dose data statistics in computers. In nuclear power plants, maintenance personnel work at several plants in a year and therefore it is essential to have a rapid and reliable system for continuous and automatic information on the accumulated dose.

Standards on quality assurance tests are given by the international standard organisation (ISO). They include guidance on how tests should be made and how the results should be understood. The uncertainties and errors in the results from dosemeter measurements depend on several factors associated with the standards and reference values used, reproduceability, linearity with dose, energy dependence. The overall uncertainty for a given practice with occupational exposure may be ± 50 % or more. If it is less than ± 20 % it is certainly to be considered very good.

The primary assessment of external exposures is in terms of effective dose equivalent or the dose equivalent to special parts of the body. In many cases the body can be considered to be rather homogenously irradiated and the assessments be made of the deep dose equivalent (depth 1 cm) and the skin dose equivalent (depth 0.007 cm). The deep dose equivalent assumed to be taken as the effective dose equivalent is measured by the dosemeter on the trunk of the body in the most exposed position. In case of exposure to soft x-rays the dosemeter is placed underneath the apron and often supplemented by a dosemeter on the collar or the head. These two dosemeters can be used to assess the effective dose equivalent and the dose equivalent to the eye. Special hand and finger dosemeters are used if those extremities are particularly exposed.

The measurement of skin dose in case of beta radiation is often uncertain because of inadequate dosemeters and difficulties in interpreting the significance of the results. The thickness of the sensitive layers of the skin varies on the body and for different persons.

If the external exposure of man is measured by fixed monitors in a room additional uncertainties are introduced, like the varying representativity of the measured result for the real exposure of man. The representativity depends on how the location of the source varies in relation to the monitor and how the worker is exposed in relation to the measured exposure. Because of these uncertainties fixed monitors are used only to supplement individual monitoring and as a warning device. They can be used alone in working places where the source is fixed and the normal exposure and potential risks of high exposures are small.

The internal exposure is mostly controlled by indirect measurements on the concentration of radionuclides in air, water or on surfaces. The compliance to given regulations is shown by comparison with derived limits associated with the primary limit (50 mSv/year) or secondary limit like ALI or with the authorized limit as is required by the competent authority. For this purpose it is therefore not necessary to assess any dose caused by internal radiation. However, sometimes the assessment of the relationship between, for instance, an air concentration of a radionuclide and the corresponding intake in the body or dose is very difficult and unreliable. In these cases biological monitoring is made including external body measurements and measurements on excreta or exhaled air. By appropriate metabolic models the activity concentrations in relevant organs or tissues of the body can be assessed and, if the time of intake is known, also the amount of intake. However, the reliability of the result is dependent on the pattern of intake, whether the intake is multiple or single, the metabolic differences among various persons, the kind of nuclide etc. Some nuclides with long physical and biological halflives in the body, such as actinides, present difficult problems to measure in excreta (only a small part is released from the body) and to assess the annual intake (because some part of the body burden depends on earlier intake). Others like ^{3}H are easier to determine by biological monitoring. In many occupations the intakes of radionuclides are very small in comparison with ALI or any authorized limit and the corresponding effective dose equivalents are also very small compared to the external doses and are therefore neglected in the dose records.

The relative number of workers subject to monitoring in various occupations and working places is not consistent in relation to the actual and potential risk of exposures. ICRP considers that in cases where it is most unlikely that annual doses will exceed three-tenths of the dose limit, individual monitoring is not necessary. However,

personal dosemeters are often used also in these occupations and in general monitoring is more common than ICRP considers to be justified. The reasons are of social and personnel-political character and are intended to confirm that the conditions are satisfactory.

These circumstances cause special problems in calculating average doses to workers, because an increased number of dosemeters to unirradiated workers will decrease the average.

Another problem is the differences in the procedures used for reporting dosemeter results less than the minimum detectable level. These results are reported as zero or the minimum level of detection or something in between. However, it is generally agreed that the recording level below which the dose is recorded as zero should not exceed one-twentieth of the annual limit pro-rata for the monitoring period.

A third problem is the differences in the procedures used for reporting missing dosemeter results. The recorded dose is either an estimated dose or a pro-rata proportion of the annual authorized limit. This can distort the records and make any assessment of average and collective doses uncertain.

PRESENTATION OF DOSE DISTRIBUTIONS

The results of monitoring expressed as an annual effective dose equivalent, a fraction of ALI or a derived limit, or as other quantities for the purpose of radiation protection, are periodically collected, assessed and presented. The purpose is to show the workers, management, authorities and the public whether given limits are complied with, the distribution of doses and the average as a measure of average individual risks, the collective dose as a measure of detriment, the changes and tendencies as compared with earlier years etc.

The individual doses can also be grouped in different types of work to illustrate the various degrees of radiation protection problems. These results might also indicate the fields in which the need for radiation protection improvement is greatest although this would eventually be decided on the basis of optimization.

The collective dose is a measure of the radiation detriment caused by the practice. It is of interest to follow the change of the collective dose to judge the justification of continued practice.

It might also be of interest to group the collective doses delivered at various individual doses, e.g. those from individual doses, less than 5 $mSva^{-1}$, those between 5 and 15 $mSva^{-1}$ and those delivered at doses exceeding 15 $mSva^{-1}$. The purpose of that is to

study the relative contribution to the radiation detriment from various levels of individual risk.

It is also of interest to make comparative studies of the collective dose from various practices or work places and for that purpose it is sometimes preferable to normalize the occupational collective dose per unit of practice, e.g. per GW-year for nuclear power production, per film used for X-ray medical examination per year at hospitals. The normalized collective dose can also be used for comparison with the collective dose from exposure of the public, caused by the practice by its radioactive releases into the environment, by use of the produced radioactive material as a consumer product, by exposure of people as patients etc. Sometimes it occurs that the collective occupational dose per unit of practice is also used for expressing a level of ambition for the radiation protection of workers.

The annual dose distribution is often found to be log-normal, especially for doses much below the dose limit. A variable x is said to have a log-normal distribution if $y = \ln(x)$ has a normal distribution. A log-normal distribution can be identified by plotting the cumulative frequency on a probability axis against the logarithm of dose. By use of the log-probability plot technique, deficiences in the data (like missing data, data less than the limit of detection limit etc.) are compensated for by extrapolation.

AVAILABLE DATA ON OCCUPATIONAL EXPOSURES AND THEIR USE FOR EPIDEMIOLOGICAL STUDIES

As described above there is a stringent and logical system of dose limitation for radiation workers with special quantities, formalistic and practical protection procedures, monitoring and methods for assessments and control. A lot of monitoring and dosimetric data are available. As has already been pointed out, the data are impaired by errors and uncertainties but neverthless, used as a part of the radiation protection system, serve their purpose to ensure good radiation protection practice. Whether the data can be used for epidemiological studies is much more uncertain. The information given here might help in that judgement.

Successful epidemiological studies on occupational radiation exposures are sparse. The reason for that is less due to bad epidemiology than to low radiation doses, which of course is a very satisfactory situation. Assuming that the average annual effective dose equivalent for workers is 5 mSv, the expected number of cancers in a group of 10^4 workers would be of the order of 1 per year at equilibrium. That number of cases would not make the epidemiology study easy.

However, there are special groups of workers with higher risks

than others and they can be identified by the monitoring and dosimetry system. One such group, the uranium miners, has been examined by epidemiologists and correlations have been found between radon daughter exposure in mines and the excess rate of lung cancer. This correlation has also been found for non-uranium miners with high radon daughter exposures.

The United Nations Scientific Committee on Atomic Radiation (UNSCEAR) has recently given extensive information in its 1982 Report on occupational exposure for various occupations, countries and time. As expected, there are great variations in average doses and collective doses depending on varying standard of radiation protection, numbers of workers and facilities, reporting system, etc. Table 5 is a rough summary table indicating the orders of magnitude.

Table 5. Average individual doses and collective doses to workers in various occupations.

Occupation	Average dose (range) mSv a^{-1}	Collective dose
Nuclear		
Mining	10-20	1 manSv $(GW_{(e)}a)^{-1}$
Milling	2-4	0.1 -"-
Fuel manufacture	1-3	1 -"-
Reactors	3-10	10 -"-
Reprocessing	3-10	10 -"-
Research	1-5	5 -"-
Total		30 -"-
Medical	0.3-3	1 manSv per 10^6 population
Industrial radiography	1-3	0.5 -"-
Research	0.3-5	

Some comments on the table. The total collective occupational dose equivalent from nuclear power production can be estimated by multiplying by the total energy generated. In 1979 that was 70 $GW_{(e)}a$ which gives about 2000 manSv. This corresponds to 0.5 manSv per 10^6 population as a comparison with the other occupations.

In the nuclear field a general tendency of the occupational doses with time is that the average doses decrease, the collective doses increase and the normalized collective doses per $GW_{(e)}a$ are decreasing slightly. Groups with higher-than-average occupational radiological risks are uranium miners and health physicists and some maintenance workers at reactor plants.

In medicine, industry and research the average doses are generally low. In industrial radiography with radioisotopes, the potential risk of serious accidents is a significant risk factor.

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