Dosimetric Concepts and Calibration of Instruments

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1. Introduction

The development of dosimetric concepts and the definition of specific quantities for use in radiation protection has a long history. An important basis for the present concepts was already provided in the 60's and 70's by both the International Commission on Radiological Protection (ICRP) and the International Commission on Radiation Units and Measurements (ICRU). In 1991 in its Publication 60 [1], the ICRP has published its most recent general recommendations for radiation protection.

These recommendations include basic concepts and guidelines for radiation protection, a modified concept for radiation protection quantities and reduced limits for occupational and public exposure. In the following this concept for radiation protection quantities and the operational quantities introduced by ICRU for monitoring of external exposure will be described and some problems discussed. A necessary basis for implementation into practice is that internationally agreed data on conversion coefficients for photon, neutron and electron radiation exist. This will also be mentioned. In addition, ISO and IEC have considered the new quantities and published standards on radiation fields used for calibration and on calibration procedures. On the basis of all these recommendations legal bodies in many countries have already introduced, or are at present introducing, new regulations for radiation protection. Obviously, this process from the formulation of recommendations by international committees based on scientific findings to the adoption of regulations and the practical realisation of a new concept takes a long time, i. e. ten years or more.

2. Concept of radiation protection quantities

The international commissions ICRP and ICRU have developed a hierarchy of quantities for radiation protection applications which can be described by primary limiting dose quantities (called "protection quantities") taking account of human body properties and additional operational dose quantities for the monitoring of external exposure. For monitoring of internal exposure other quantities than dose quantities are used.

The basic idea of a *primary limiting quantity* is to relate the "risk" of exposure to ionising radiation (exposure by internal and external radiation sources) to a single (dose) quantity which takes account of the man as a receptor, the different radiation sensitivities of various organs and tissues and the different radiation qualities. Other influence parameters, however, e.g. the influence of dose and dose rate or sex and age of a person exposed on the biological response to the exposure risk, were not considered in the definition of these quantities. These effects are to some extent taken into account by the various limitations of exposure.

Operational quantities are dose quantities defined for use in radiation protection measurements for external exposure (area or individual monitoring). They usually should provide an estimate of or upper limit for the value of the limiting quantities due to an exposed, or potentially exposed, person and are often used in the place of those quantities in practical regulations. Operational quantities are needed for monitoring external exposures because

- protection quantities generally are not measurable,
- for area monitoring a point quantity is needed,
- a non-isotropic human-body related quantity like the effective dose is not appropriate.

For internal exposure, however, other methods are used and no similar type of quantities has been defined yet. In this case organ doses or effective dose are directly estimated from the information on intake or excretion of radioactive substances.

Both, protection quantities and operational quantities can be related to "basic physical quantities" as specified in ICRU Report 33 [2], which are generally used in radiation metrology, and in radiation dosimetry in particular, and are defined without considering any specific aspect of

radiation protection. Basic physical quantities are quantities whose units are directly obtained through primary standards at national standards laboratories, e. g. the fluence of neutrons, exposure or air kerma for photons and absorbed dose for beta radiation (see chapter 7). They have been well known for a long time. They are point quantities defined in any point of a radiation field.

In the following, the notes are restricted to the situation of external radiation exposure. Most of the radiation protection quantities mentioned above either are not directly measurable or their values are not directly traceable to primary standards. For this reason, knowledge of the numerical relations between the basic physical quantities and these dose quantities is a very important condition for the practical implementation of the whole system of radiation protection quantities. It is, therefore, vital that an agreed set of data accepted by national and international authorities is available so that confusion is avoided. Following ICRP Publication 60, conversion coefficients for photon, neutron and electron radiation were evaluated and published in ICRP Publication 74 [3] in 1996.

3. Limiting quantities

As early as in 1977 the ICRP [4] introduced the organ (or tissue) dose equivalent, H_T, and the effective dose equivalent, H_E, whose definition takes care of the relative variation of the tissue response with different types of radiation and different tissues or organs in the human body. Although in general this concept was not changed by ICRP 60 [1], important modifications, however, were introduced. The present system of quantities is summarised in the following.

Absorbed dose and equivalent dose in an organ or tissue

The **absorbed dose in an organ or tissue, D_T**, is the absorbed dose averaged over the volume of a tissue or organ T (rather than at a point). While the absorbed dose at a point generally is the fundamental dose quantity, in radiation protection, the mean dose in an organ becomes the basic protection quantity correlated with the exposure risk. This concept therefore is obviously based on the linear dose-effect relationship and the additivity of doses for risk assessment as an appropriate approximation in the low dose region. Otherwise, averaging of doses and adding of doses over long periods would not be an acceptable procedure.

The **equivalent dose in an organ or tissue** is defined by:

$$H_T = \sum_R w_R D_{T,R} ,$$

where $D_{T,R}$ is the mean organ dose in the organ or tissue T from radiation of type R incident on the human body and w_R are radiation weighting factors characterising the biological effectiveness of the specific radiation R relative to photons. These factors have replaced the mean quality factors used in the concept of organ dose equivalent before [4]. The sum is taken over all types of radiation involved.

Radiation weighting factors

For external irradiation, the values of the radiation weighting factors are given by the parameters of the external radiation field only (by the type and spectral distribution of the radiation incident on the body). This means that w_R is a body-averaged value representing a mean value for the relative biological effectiveness of all tissues of the body. Because the w_R value is the same for all organs in a body, this procedure obviously ignores any local variation of the radiation quality in the human body which may result from the generation of secondary radiation of different types in the human body. This effect is important mainly in the case of incident neutrons where secondary photons strongly contribute to the absorbed doses of various organs.

The w_R values for various types of radiation are specified in ICRP 60 in a table (see Table 1). For photons, electrons and muons of all energies a value of one is fixed with the exception of Auger electrons emitted from nuclei bound to DNA, but this is of no importance for external irradiation. The radiation weighting factor for neutrons depends on the neutron energy. Different w_R values are given by either a step function or a continuous function as an approximation (see Fig. 1). In practice, neutron fields mostly contain neutrons with a broad energy distribution. The use of a continuous w_R-function for effective dose estimation is much more appropriate in these

situations and it is, therefore, strongly recommended to apply the continuous function in any case to avoid additional ambiguities. In this case the weighting factor for neutrons ranges from 5 to 22 depending on the neutron energy with its maximum value at 500 keV. All conversion coefficients for neutrons published in ICRP 74 are based on the continuous function only [3].

Discussion on radiation weighting factors for protons and neutrons and especially on their energy dependence are still underway. For high energy protons (e.g. above about 10 MeV) a weighting factor of less than 5 seems to be more realistic because the ionisation density of protons decreases with energy and nuclear reactions of protons contribute only partially to the total dose. A value of about 2 is assumed to better describe its relative biological effectiveness, while external protons of much lower energy are of lesser importance because they mainly contribute to the skin dose.

The weighting factors for neutrons of two energy ranges are under discussion. At energies below about 10 keV the neutrons are strongly moderated in the human body and finally absorbed via the reaction H(n,γ)D, thus producing secondary photons in the body which then contribute up to 90 % to the total absorbed dose in a human body. Therefore, a weighting factor value smaller than 5 might be more appropriate in this neutron energy range (see Fig. 2).

For neutrons with energies above about 30 MeV calculations of mean quality factors in an anthropomorphic phantom show that these values are much smaller than the weighting factor of 5. Obviously, the description of the energy dependence either based on $Q(L)$ or given by w_R are still inconsistent for neutrons.

Effective dose

The effective dose, E, is the weighted sum of organ equivalent doses:

$$E = \sum_T w_T H_T \qquad \text{with} \qquad \sum w_T = 1$$

where w_T are tissue weighting factors characterising the relative sensitivity of the various tissues with respect to cancer induction and mortality. Twelve tissues and organs are specified with individual weights w_T and an additional "remainder" tissue is defined (with a weight of 5 %) the dose of which is given by the mean value from ten specified organs and tissues (see Table 2).

4 Operational quantities

4.1 Dose equivalent and quality factor

For a long time, the radiation protection quantity dose equivalent, H, has already been defined by

$$H = Q\ D,$$

where D is the absorbed dose at the point of interest and Q a quality factor weighting the relative biological effectiveness of radiation. Q is defined as a function of the linear energy transfer, L, of a charged particle in water (see e.g. [4]). In principle, this has not been changed by ICRP 60, but the dose equivalent is now restricted to the definition of operational radiation protection quantities and the quality factor function $Q(L)$ was modified in 1991 according to the following equation:

$$Q(L) = \begin{cases} 1 & \text{for} \quad L < 10 \text{ keV/µm} \\ 0{,}32\ L - 2{,}2 & \text{for} \quad 10 \le L \le 100 \text{ keV/µm} \\ 300/\sqrt{L} & \text{for} \quad L > 100 \text{ keV/µm} \end{cases}$$

The quality factor Q at a point in tissue is then given by [5]:

$$Q = 1/D \int_0^\infty Q(L)\ D_L\ \mathrm{d}L$$

where D_L is the distribution of D in L at the point of interest. This function is most important for neutrons because various types of secondary charged particles are produced in tissue in this case.

4.2 The concept of operational quantities

The basic concept of the operational quantities are described in the ICRU Reports 39 and 43 [6, 7]. The present definitions are given in ICRU Report 51 [5]. The operational quantities for radiation protection are dose equivalent quantities defined either for penetrating or for low-penetrating radiation.

The radiation incident on a human body is characterised as *penetrating radiation* or *low-penetrating radiation*, depending on the ratio of the skin dose to effective dose. Radiation is considered to be low-penetrating when the dose equivalent received by the skin (dose received at a depth of 0,07 mm) in the case of normal incidence of a broad radiation beam is higher than ten times the effective dose - otherwise it is considered to be penetrating. Low-penetrating radiations are α-particles, β-particles with energies below 2 MeV and photons with energies below about 12 keV. Neutrons always are penetrating radiation.

Due to the different tasks in radiation protection monitoring - area monitoring for controlling the radiation at work places and definition of controlled or forbidden areas or individual monitoring for the control and limitation of individual exposures - different operational quantities were defined. While measurements with an area monitor are mostly performed free in air, an individual dosemeter is usually worn on the front of the body. As a consequence, in a given situation, the radiation field "seen" by an area monitor free in air differs from that "seen" by an individual dosemeter worn on a body where the radiation field is strongly influenced by the backscatter and absorption of radiation in the body. The operational quantities allows for this effect. They may be presented as follows:

Radiation type	**Quantities for area monitoring**	**Quantities for individual monitoring**
Penetrating radiation	ambient dose equivalent, $H^*(10)$	personal dose equivalent, $H_p(10)$
Low-penetrating radiation	directional dose equivalent, $H'(0{,}07,\Omega)$	personal dose equivalent, $H_p(0{,}07)$

In rare cases, if a limitation of the dose of the eye lens becomes significant, further quantities for low-penetrating radiation, $H'(3,\Omega)$ for area monitoring and $H_p(3)$ for individual monitoring are recommended. For photon and neutron radiation, they are not of importance.

4.3 Operational quantities for area monitoring

4.3.1 The ICRU sphere phantom

For all types of radiation the operational quantities for area monitoring are defined on the basis of a phantom, the ICRU sphere. It is a sphere of tissue-equivalent material (diameter: 30 cm, density: 1 g cm^{-3}, mass composition: 76,2 % oxygen, 11,1 % carbon, 10,1 % hydrogen and 2,6 % nitrogen). It adequately approximates the human body as regards the scattering and attenuation of the radiation fields under consideration.

4.3.2 Aligned and expanded radiation field

The operational quantities for area monitoring defined in the ICRU sphere should retain their character of a point quantity and the property of additivity. This is achieved by introducing the terms expanded and aligned radiation field in the definition of these quantities (see Fig. 3):

An *expanded* radiation field is a radiation field in which the spectral and the angular fluence have the same values in all points of a sufficiently large volume equal to the values in the actual field at the point of interest. The expansion of the radiation field ensures that the whole ICRU

sphere is thought to be exposed to a homogeneous radiation field with the same fluence, energy distribution and directional distribution as in the point of interest P of the real radiation field.

If all radiation is (thought to be) aligned in the expanded radiation field so that it is opposed to a radius vector Ω specified for the ICRU sphere, the aligned and expanded radiation field is obtained. In this fictitious radiation field, the ICRU sphere is homogeneously irradiated from one direction, and the fluence of the field is the integral of the angular differential fluence at point P in the real radiation field over all directions. In the expanded and aligned radiation field, the value of the dose equivalent at any point in the ICRU sphere is independent of the directional distribution of the radiation of the real radiation field.

4.3.3 Ambient dose equivalent, $H^*(d)$

For area monitoring of penetrating radiation the operational quantity is the ambient dose equivalent, $H^*(d)$, with d = 10 mm.

> The ambient dose equivalent, $H^*(d)$, at a point of interest in the real radiation field, is the dose equivalent that would be produced by the corresponding aligned and expanded radiation field, in the ICRU sphere at a depth d, on the radius vector opposing the direction of radiation incidence.
>
> For penetrating radiation it is d = 10 mm and $H^*(d)$ is written $H^*(10)$.

As a result of the imaginary alignment and expansion of the radiation field, the contributions of radiation from all directions add up. The value of $H^*(10)$ is therefore independent of the directional distribution of the radiation in the actual field. This means that the reading of an area dosemeter for the measurement of $H^*(10)$ should be independent of the directional distribution of the radiation - an ideal detector should have an isotropic fluence response.

$H^*(10)$ should give a conservative estimate of the effective dose a person would receive when staying at this point. This is always the case for photons below 10 MeV in contrast to the formerly used free-in-air quantities air kerma or exposure which are non-conservative in the photon energy range near 80 keV. For neutrons the situation is different. $H^*(10)$ is not conservative with respect to E under AP irradiation conditions in the energy range from 1eV to about 50 keV. In realistic neutron fields with a broad neutron energy distribution, however, this energy range mostly is of small importance and in practice $H^*(10)$ therefore remains in most cases conservative with respect to E.

4.3.4 Directional dose equivalent, $H'(d,\Omega)$

For area monitoring of low-penetrating radiation the operational quantity is the directional dose equivalent, $H'(d,\Omega)$ with d = 0,07 mm or, in rare cases, d = 3 mm.

> The directional dose equivalent, $H'(d,\Omega)$, at a point of interest in the actual radiation field, is the dose equivalent that would be produced by the corresponding expanded radiation field, in the ICRU sphere at a depth d, on a radius in a specified direction Ω.
>
> For low-penetrating radiation it is d = 0,07 mm and $H'(d,\Omega)$ is written $H'(0,07,\Omega)$.
> In case of monitoring the dose to the eye lens $H'(3,\Omega)$ with d = 3 mm may be chosen.

For unidirectional radiation incidence the quantity may be written $H'(0,07,\alpha)$, where α is the angle between the direction Ω and the direction opposite to radiation incidence. The value of $H'(10,0°)$ is equal to $H^*(10)$.

In practice, $H'(0,07,\Omega)$ is almost exclusively used in area monitoring for low-penetrating radiation, even if irradiation of the eye lens cannot be precluded.

The value of the directional dose equivalent can strongly depend on the direction Ω, i.e. on how the ICRU sphere is oriented in the expanded radiation field. The same is true of instruments for measuring low-penetrating radiation - e.g. beta- or alpha-particle radiation —the reading of which can strongly depend on the orientation in space. In radiation protection practice, however, it is always the maximum value of $H'(0,07,\Omega)$ at the point of interest which is of importance. It is usually obtained by rotating the dose rate meter during the measurement and looking for the maximum reading.

4.4 Operational quantities for individual monitoring

Individual monitoring is usually performed with dosemeters worn on the body and the operational quantity defined for this application takes this situation into account. For individual monitoring the operational quantity is the personal dose equivalent, $H_p(d)$.

The personal dose equivalent, $H_p(d)$, is the dose equivalent in ICRU tissue at a depth d in a human body below the position where an individual dosemeter is worn.

For penetrating radiation a depth $d = 10$ mm is recommended.

For low-penetrating radiation a depth $d = 0,07$ mm is recommended.
In special cases of monitoring the dose to the eye lens a depth $d = 3$ mm may be appropriate.

The operational quantities for individual monitoring meet several criteria. They are equally defined for all types of radiation, additive with respect to various directions of radiation incidence, take into account the backscattering from the body and can be approximately measured with a dosemeter worn on the body. The new personal dose equivalent quantities, $H_p(10)$ and $H_p(0,07)$, are defined in the person, in the actually existing radiation field, and are measured directly on the person.

Other requirements the quantities should satisfy can, however, be fulfilled only with additional specifications.

Obviously, the person influences the radiation field by scattering and attenuating the radiation. Since $H_p(10)$ and $H_p(0,07)$ are defined in the body of each person considered, their values vary from one person to another and also depend on the location on the body where the dosemeter is worn. In a non-isotropic radiation field the value of the personal dose equivalent, $H_p(10)$, also depends on the orientation of the person in this field.

An operational quantity for individual monitoring should allow the effective dose to be assessed or should provide a conservative estimate under nearly all irradiation conditions. This obviously is not always possible. For example, if a dosemeter is worn at the front side of the body and the person is exposed from the back, this condition cannot be fulfilled because most of the radiation will already be absorbed within the body and not reach the front where the dosemeter is positioned. Even if the dosemeter correctly measures $H_p(10)$ in this case, this value is not a conservative estimate of the effective dose, E. It is, therefore, an additional requirement in individual dosimetry that the personal dosemeter must be worn at a position on the body which is representative of body exposure. For a dosemeter position in front of the trunk, however, the quantity $H_p(10)$ mostly furnishes a conservative estimate of E even in cases of lateral or isotropic radiation incidence on the body.

A further requirement for an operational quantity is that it allow dosemeters to be calibrated under reference conditions in terms of that quantity. The personal dose equivalent is defined in the individual human body and it is obvious that individual dosemeters cannot be calibrated in front of a real human body. For a calibration procedure, the human body must therefore be replaced by an appropriate phantom. Three standard phantoms have been defined by ISO for this purpose and the definition of $H_p(10)$ and $H_p(0,07)$ is extended to define positions and doses not only in the human body but also in three phantoms of ICRU tissue (see Fig. 4a) —a slab phantom

(30 cm x 30 cm x 15 cm), a wrist phantom (a cylinder of 73 mm in diameter and 300 mm in length) and a finger phantom (a cylinder of 19 mm in diameter and 300 mm in length). In reference radiation fields used for calibration, the values of the quantities in these phantoms are defined as the true values of the corresponding H_p-quantities. More information is given in chapter 11.

4.5 Further comments

The definition of *penetrating* and *low-penetrating* radiation as advice for the application of operational quantities is rather complex. For this purpose, the values of protection quantities (effective dose and skin dose) must be known and, in addition, it has not been taken into consideration that the effective dose depends on the direction of radiation incidence on the human body [3]. The choice of a factor of 10 between skin equivalent dose and effective dose as a selection criterion was logical when the skin dose and effective dose equivalent limit also differed by a factor of 10 (500 mSv per year to 50 mSv per year) and the radiation was then characterised depending on what dose limit was relevant. This, however, is no longer the case because following ICRP 60 the ratio of the annual limits of skin equivalent dose to effective dose now is 25. In practice, it seems possible to totally avoid the complex specification of radiation types in the definition of the operational quantities by stating only for what objectives and tasks the quantities are used [8]. For example, $H^*(10)$ and $H_p(10)$ will be applied if a control of effective doses is considered; $H'(0{,}07,\Omega)$ and $H_p(0{,}07)$ will be used if the skin dose needs to be controlled.

A second remark deals with the quantities $H'(3,\Omega)$ and $H_p(3)$ which are proposed for use if the equivalent dose to the eye lens needs to be separately controlled. The eye lens is not defined as a specific organ and even considering that a separate limit has been defined for the equivalent dose of the eye lens, it is questioned whether a separate operational quantity is really needed. It appears that for area monitoring $H^*(10)$ and $H'(0{,}07,\Omega)$ are sufficient in rare cases to control also external exposures to the eye lens. This may also be the case in individual monitoring. To reduce the number of operational quantities to what is really necessary I would suggest omitting these quantities.

A further remark deals with the definition of the directional dose equivalent, $H'(d,\Omega)$, for which a direction Ω must always be specified. This quantity is applied only in case of monitoring low-penetrating radiation. As mentioned before, in practice, only the maximum value of $H'(0{,}07,\Omega)$ at a given point is of interest in radiation monitoring. This value is usually determined by rotating the instrument and thus varying its orientation. It is proposed for simplification that this value should be written $H'(0{,}07)$.

All these comments may result in a proposal for a slightly modified scheme of operational quantities which simplifies their definition and directly specifies their application. These changes obviously will not lead to changes in the present practical use of the operational quantities.

Objective	**Quantities for**	
	area monitoring	**individual monitoring**
control of effective dose	ambient dose equivalent, $H^*(10)$	personal dose equivalent, $H_p(10)$
control of skin equivalent dose	directional dose equivalent, $H'(0{,}07)$	personal dose equivalent, $H_p(0{,}07)$

However, the question remains whether $H^*(10)$ and $H_p(10)$ can be used as operational quantities in very high energy radiation fields, e.g. fields at flight altitudes or in space or close to

high-energy accelerators. This needs to be further investigated and discussed on the basis of calculated conversion coefficients which have recently become available.

A final remark is concerns the problem of extremity monitoring. Extremities may be often exposed in parts only. They are not, however, specifically defined and the phantoms used for the calculation of conversion coefficients have modelled the distribution of the muscle tissue and hence the extremities without any detail. For extremities specific conversion coefficients thus do not exist and hence a decision which operational quantity should be applied to measure extremity doses in addition to those for the skin has not yet been achieved on an international level. Usually personal extremity dosemeters are worn on the fingers and the operational quantity to be measured is $H_p(0,07)$. This will be sufficient if only low-penetrating radiation is involved. It may, however, be questioned whether this quantity will be appropriate if other radiations are involved.

5. Conversion coefficients

Protection quantities

For external exposure to photons, neutrons or electrons conversion coefficients were calculated for monoenergetic radiation relating the physical quantities air kerma K_a (for photons), neutron fluence, Φ_n, or electron fluence, Φ_e, to the dose quantities defined in the human body. For this purpose extensive Monte Carlo calculations were performed by various groups using geometrically defined anthropomorphic phantoms - either hermaphrodite or ADAM and EVA models [9,10]. A detailed set of evaluated data of coefficients for the three types of radiation was prepared by an international ICRU/ICRP Report Committee and then published by ICRP [3] and ICRU [11]. These data are now recommended for general use. They are also included in ISO standards and have already become the basis for many regulations in various countries.

The change from the formerly used quantity effective dose equivalent, H_E, to effective dose, E, has resulted in very small quantitative changes only for external photon and beta radiation (see Fig. 5). In photon fields, E generally is a few percent smaller than H_E due to the increase in the number and the difference in the weighting of the organs considered. A photon dosemeter the reading of which provided a conservative estimate of H_E, behaves similarly with respect to E and it, therefore, is not necessary to change anything in the record kept for individual photon doses when changing from H_E to E.

For neutrons, however, the situation is different (see Fig 6). Due to the change in the weighting of neutrons and the use of w_R instead of the $Q(L)$ concept, the conversion coefficients for the organ and effective dose equivalent are much higher than for the former quantity effective dose equivalent, H_E. Data [3] show that for neutrons below about 1 MeV, under all irradiation conditions, the ratio E/H_E is always higher than 2 with a maximum of about 7 for PA irradiation at about 50 keV neutrons.

Operational quantities

For the calibration of measuring devices conversion coefficients are needed for all types of radiation and all quantities of interest. While for the ambient and directional dose equivalent conversion coefficients have been calculated in the ICRU-sphere phantom, the coefficients for the personal dose equivalent quantities are given for the slab phantom (for $H_p(10)$ and $H_p(0,07)$) and the wrist and finger phantoms (only for $H_p(0,07)$) (see Fig. 4a). Evaluated and recommended data for photons, neutrons and beta radiation are available in the ICRP and ICRU Publications (3,11) and in some relevant ISO standards (14,15,16).

The value of the dose equivalent at a depth of 0,07 mm in the ICRU sphere —and therefore also $H'(0,07,\alpha)/K_a$ and $H'(0,07,\alpha)/\Phi_e$ - depends on the direction of radiation incidence on the sphere in relation to the reference direction selected (specified by angle α). The reading of an area dosemeter for the measurement of $H'(0,07,\Omega)$ should exhibit an appropriate directional dependence. The same holds for the quantity $H_p(0,07)$ and the response of an individual dosemeter.

For photons, there are only small changes in the values of the conversion coefficients compared to the ICRU Report 47 [12] at low photon energies. While the ICRP and ICRU

publications give data only for monoenergetic radiation fields, mean values for the different series of x-rays are available from ISO standard 4037 (14). Photon radiation from radionuclides mostly is not monoenergetic. Table 3 shows calculated mean conversion coefficients for some radionuclides frequently used.

The conversion coefficients $H^*(10)/\Phi_n$ for monoenergetic neutrons published in ICRP 74 are generally recommended for use. They are different from those published in ICRP Publication 51 [13] because of the change in the $Q(L)$ function (see Fig. 6). The evaluated data are based on a continuous curve. The sharp peak at 20 MeV obviously is an artefact as below 20 MeV 7 sets with calculated data were available for the evaluation, while the data above 20 MeV rely mainly on one set only. Obviously, $H^*(10)$ is not always a conservative estimate of E, but underestimates E at intermediate energies and at energies above about 30 MeV. In real neutron fields with broad energy spectra, however, the energy range between thermal and 10 keV neutrons contributes very little to the total dose equivalent and hence $H^*(10)$ remains conservative compared to E for nearly all practical neutron spectra. This may not be the case for spectra containing many high energy neutrons above about 30 MeV. Having, however, in mind the large uncertainties of the conversion coefficients for neutrons above 20 MeV as mentioned in ICRU Report 57 (11), does not seem to be justified at this moment to choose a separate operational quantity in this energy range. During recent years, however, more calculated high energy data have become available and a new evaluation is strongly recommended. with a view to obtain a more reliable data record for high energies.

With the quantities $H'(0,07,\Omega)$ and $H'(3,\Omega)$, clearly defined dose quantities for beta radiation are now available for area monitoring. As extrapolation chambers are used for direct measurements of absorbed dose and dose equivalent, conversion coefficients for $H'(0,07, 0°)/\Phi_e$ or $H'(3, 0°)/\Phi_e$ are rarely needed.

6. Calibration principles

Calibration can be defined as a set of operations performed under specified conditions to establish the relationship between values indicated by a measuring instrument or system and the corresponding known true values of a quantity to be measured. In the field of radiation protection, the measuring instruments are usually area survey meters or personal dose and dose rate meters.

The calibration of personal dosemeters or area survey meters used for radiation protection purposes is mostly a three step process. First, the value of a physical quantity such as air kerma or particle fluence for which primary standards usually exist, is determined by a reference instrument at a reference point in the radiation field used for calibration. Second, the value of the appropriate radiation protection quantity is determined by application of a conversion coefficient relating the physical quantity to the radiation protection quantity. Conversion coefficients used to determine operational quantities for neutrons and photons were evaluated by international committees and finally accepted for general use by international agreements. Third, the device being calibrated is placed at this reference point to determine the response of the instrument to the radiation protection quantity, e. g. the personal, ambient or directional dose equivalent.

The calibration methods described in this part closely follow the recommendations of the International Organization for Standardization (ISO) dealing with reference radiations [14,15,16,17, 18]. These methods are applicable only to the determination of dose equivalents from external radiation sources.

7. Physical quantities

The radiation types used for the calibration of dosemeters are mainly photons, neutrons and beta particles. It would in principle be desirable to perform the calibrations for all radiation types in exactly the same way using the same equipment. The physical nature of the different types of radiationdictates, however, that calibrations for each of these types are performed differently using different instrumentation and techniques.

The primary physical quantity used to specify a photon radiation field is exposure or air kerma, and the primary standard instruments used for its measurement are air-filled ionization

chambers. For photon energies up to about 150 keV, mostly a free-air chamber is used as a standard instrument to measure air kerma. For higher photon energies, air-equivalent walled cavity chambers are generally employed. Properties of radiation fields used for the calibration of photon dosemeters are described in ISO 4037-1 [14].

Calibrations of dosemeters and survey instruments for the measurement of beta radiation are performed using standard reference beta sources as specified in ISO standard 6980-1 [15]. Determination of the conventional true value of the absorbed dose, and hence the directional dose equivalent, is achieved with a thin-window extrapolation ionisation chamber.

The primary quantity measured for neutrons is fluence. In monoenergetic neutron fields the fluence is measured either directly by a reference instrument (e. g. proton recoil telescope, proportional counter or Long Counter) or by applying the associated particle method. As regards radioactive neutron sources, the neutron fluence is determined from the source emission rate which is usually determined from comparative activation measurements performed by a national standards laboratory. The emission rate is then used to compute the neutron fluence or fluence rate. In addition, the neutron energy spectrum must be known. With the known spectral fluence mean conversion coefficients can be calculated and applied to determine the neutron dose equivalent [16].

8. Calibration procedures

8.1 General procedures applicable to all calibrations

All radiation qualities should be chosen in accordance with the relevant ISO standards 4037-1, 6980 and 8529-1 [14-16], and produced by the methods described in these standards. It generally is useful to select an appropriate radiation quality taking into account the specified energy range and range of dose equivalent or dose equivalent rate of the device to be calibrated. In addition, it is necessary to take account of contaminating radiation in the calibration field such as scattered radiation or photons in a neutron field.

The three aforementioned ISO standards are at present extended to also include methods for the implementation of the new operational quantities. This is done by the development of six additional standards in ISO 4037, 6980 and 8529 referred to as Part 2 and Part 3 [14-16].

9. Procedures for reference calibrations

The procedures in each of the following sections apply to calibrations using photon, beta or neutron reference radiation. In many cases, each type of calibration follows basic principles, but if specific requirements are to be met, these will be stated.

9.1 Photons

For photon radiation, it is expected that the reference calibration laboratory will have a constant potential x-ray generator at its disposal which is appropriate to produce various filtered x-ray beams and if possible also fluorescence x-ray spectra. The characteristics of the x-ray machine such as tube voltage, tube current as well as the stability of these parameters must be known. The inherent filtration of the x-ray tube must be determined. The materials used for the construction of the filter sets and fluorescence radiators must also be well-known in terms of composition, thickness and uniformity. ISO Standard 4037-1 [14] gives specifications for these items.

The verification of the quality of the filtered x-ray beams should at least include the determination of the half-value layer and the homogeneity coefficient. At energies below 50 keV care must be taken because of the strong energy dependence of conversion coefficients for $H^*(10)$ and $H_p(10)$. Mean conversion coefficients for x-ray spectra below about 30 keV as given in ISO Standard 4037-1 may not be appropriate if the photon spectrum differs from that assumed in the standard. It is recommended at these energies either to measure the energy spectrum or to

determine $H_p(10)$ directly using an $H_p(10)$-reference instrument [19]. Also, an evaluation should be performed to determine the degree of scattered radiation present.

Since the output of an x-ray machine may be subject to variations as a function of time, it is necessary to control this output. This can be achieved either by measurement with the reference chamber before and after the calibration measurement or by monitoring using a thin-window transmission-type ionisation chamber. The filtered x-ray beam is normally allowed to pass through the transmission monitor before reaching the device being calibrated.

Photon reference radiations can also be produced by various radionuclide sources. ISO 4037-1 recommends the use of ^{241}Am, ^{137}Cs and ^{60}Co, with energies of about 59,5 keV, 662 keV and 1252 keV (mean of 1173 keV and 1332 keV), respectively. Recommendations for collimation and physical characterisation of these photon sources are similar to those given for x-ray beams. Continuous monitoring of the intensity of such sources is usually not necessary.

Special high-energy photon beams ranging from 4 to 9 MeV can be produced via nuclear reactions using an ion accelerator ($^{19}F(p,\alpha\gamma)^{16}O$ for about 6 —7 MeV and $^{12}C(p,p\langle\gamma)^{12}C$ for 4,44MeV gamma radiation). Furthermore, activation in a reactor can be used to produce high energy gamma ray sources using $^{16}O(n,p)^{16}N\rightarrow^{16}O^*$, Ti(n,γ) or Ni(n,γ). Details of the production and characterisation are also given in ISO 4037-1 [14]. While the accelerator-produced fields show a low background, a much higher source strength can be achieved using activated Ti- or Ni-sources.

The primary quantity that must be determined for the calibration of photon-measuring devices is the air kerma. The air-kerma or air-kerma rate at the point of tes normally is determined using an air-equivalent walled ionisation chamber calibrated by a national primary standards laboratory (or which is traceable to such a chamber). The chamber is positioned with the centre of its collecting volume at the point of test, without a phantom in place. The charge collected by the chamber is measured with an electrometer, and corrections are applied to account for the effects of air temperature, air pressure, ionic recombination and other influence parameters.

Finally, an air-kerma-to-dose equivalent conversion coefficient appropriate for the radiation must be applied to specify the conventional true value of the operational quantity used for calibration. Values for these coefficients are given in ISO 4037 Part III [14].

9.2 Beta particles

Methods of production and characterisation of reference beta fields are found in ISO Standard 6980 [15]. Two general types of reference sources are recommended for use. The first type consists of small (nearly point) sources mounted in a special irradiation apparatus that includes beam flattening filters to increase the field uniformity. The second type comprises those sources which do not use flattening filters, and includes several planar sources that can be used for the calibration of surface contamination measuring instruments.

The ISO Standard recommends five sources of the first type (point sources): ^{14}C, ^{147}Pm, ^{204}Tl, $^{90}Sr+^{90}Y$ and $^{106}Ru+^{106}Rh$, producing beta spectra with maximum energies of approximately 0,16 MeV, 0,24 MeV, 0,76MeV, 2,27 MeV and 3,54 MeV, respectively. Instead of ^{204}Tl which is not always available today, ^{85}Kr may also be used because the mean beta energy is nearly equal, the half-life time is much longer (3915 d compared to 1381 d) and sources with much higher activities are now available (e.g. 3,7 GBq compared to 19 MBq) which leads to dose rates higher by a factor of about 100. Point sources should be used only at specific distances and together with beam flattening filters to achieve a beam profile with a constant dose rate. Sources of the second type also include the aforementioned radionuclides, but they can be used at various distances without flattening filters. For the characterisation of these sources, it is recommended that the energy spectrum be measured and the residual energy be determined. This determination is based on a measurement of the signal from an appropriate detector, as a function of the thickness of absorbers placed between source and detector. The uniformity of the dose over the calibration area should also be determined to be within the specifications as described in the standard. Calibration distances are also specified for each source.

The determination of the absorbed dose or dose rate at the point of test is to be made by direct measurement by a national standards laboratory or using a suitable transfer standard instrument such as an extrapolation ionisation chamber.

9.3 Neutrons

Neutron reference radiations for calibration are given in ISO Standard 8529-1 [16]. This standard describes neutron reference radiations from four radionuclide-based neutron sources recommended for the calibration of neutron dosemeters. In addition further neutron fields are specified which can be produced using reactors and charged-particle accelerators. The radionuclide sources include D_2O-moderated ^{252}Cf, ^{252}Cf, ^{241}Am-B and ^{241}Am-Be. The dose equivalent average energy of these sources are, 2,2 MeV, 2,4 MeV, 2,8 MeV and 4,4 MeV, respectively. Such radionuclide sources normally are initially calibrated by a national primary standards laboratory. The calibration certificate will provide a value for the neutron source strength. The value of this quantity can then be used to compute the neutron fluence rate at the point of test. An anisotropy of the source must, however, be considered. Care must also be taken to correct for secondary neutrons scattered from the air or the walls of the room. This is usually done either by performing measurements at different distances from the source or by an additional measurement with an absorber (shadow shield) between source and detector. The fluence or fluence rate can then be multiplied by the appropriate fluence-to-dose equivalent conversion coefficient given in ISO 8529 [16] to obtain the value of the dose equivalent used for calibration. With broad neutron spectra a mean value must be applied. In this case, however, not a fluence-averaged but a dose-averaged value must be calculated.

The response of a neutron detector mostly depends on the neutron energy and therefore the calibration factor of a dosemeter or survey instrument will generally depend on the neutron fluence spectrum of the source used for calibration. Furthermore, the response depends on the angle of incidence of the neutrons. Measurements in neutron fields other than that used for calibration mostly require some information about the energy dependence of the response so that it can be assessed whether a correction factor has to be applied. The energy dependence of a detector response can be determined by measurements in monoenergetic neutron fields with a broad range of different energies, in special cases in part also by calculations. Monoenergetic neutron fields can either be produced using an accelerator facility and specific nuclear reactions or by providing a filtered neutron beam from a reactor. Details are given in ISO 8529-1.

10. Procedures for the calibration of area survey meters

Instruments used for area monitoring are calibrated free in air. This condition results from the definitions of ambient dose equivalent, $H^*(d)$, and directional dose equivalent, $H'(d,\Omega)$, which require that the design of the instruments used for measuring these quantities take account of backscatter. Most area survey meters measure and therefore have to be calibrated in terms of dose equivalent rate (although some devices also measure accumulated dose equivalent).

The survey meter is placed with its reference point coinciding with the point of test. The quantity of interest, such as the air-kerma rate, will have been measured previously. If the reference radiation source is an x-ray generator, the reference calibration can be performed with a primary or secondary standard ionisation chamber and maintained using of a monitor ionisation chamber. This can be calibrated to indicate a conventional true value of the air-kerma rate at the point of test for the survey meter. Alternatively, the primary reference standard and the survey meter can both be mounted on a remotely controlled platform that allows for alternative placing of the reference points of the two instruments at the point of test. First the standard chamber is used to determine the air-kerma rate, and then the reading of the survey meter is determined. As a further check of the stability of the air-kerma rate, the standard chamber can then be returned to the point of test for measurement.

The calibration of neutron survey meters also requires placing of the instrument's reference point at the point of test, however, the determination of the location of the reference point may be more complex. The effective point of measurement for a neutron survey meter is defined as the position within the device at which the device may be considered to be located if it could be replaced by an equivalent point detector. In practice, this means that the free field response of this device, as a function of distance between the effective point of measurement and the source centre, follows the inverse-square law for a given neutron energy or energy spectrum. The location of the effective point of measurement within the device generally depends on the geometrical design of the instrument and the energy and directional distribution of the incident neutron field.

Some brief guidelines are given here, but the reader is referred to ISO Standard 10647 [16] for complete details. For spherical detectors, the effective point of measurement is the geometrical centre. For cylindrical devices irradiated perpendicular to the cylindrical axis, the effective point of measurement is at the intersection of the cylindrical axis and the line from the source, perpendicular to the axis. For irradiation parallel to the cylindrical axis, the effective point of measurement is located along the axis, but its position will be dependent on the incident neutron energy, and its location must be determined by measurement.

The calibration of beta survey instruments basically is straightforward. The reference point of the instrument is placed at the point of test and the calibration in terms of the operational quantity can be carried out directly. It is however necessary to exercise extreme care in the use of the reference radiation sources specified in ISO 6980 [15]. The calibration distances are relatively small so that positioning errors can result in large measurement uncertainties. In addition, the relatively low energy of some of these beta sources requires that corrections be applied for a number of effects including those due to air density and humidity.

11. Procedures for the calibration of personal dosemeters

While the calibration of survey meters is generally carried out free in air, the calibration of personal dosemeters should be performed with the dosemeters mounted on an appropriate phantom. Three phantoms have been defined by ISO for calibrations, corresponding to the positions on which personal dosemeters are worn (on the body, on the arm or on a finger). Their shapes are the same as those of the ICRU-tissue phantoms used for the calculation of the conversion coefficients (see chapter 5).

The quantity to be measured for individual monitoring is the personal dose equivalent, $H_p(10)$ or $H_p(0,07)$, respectively, in the body of the person wearing the dosemeter. For the calibration of personal dosemeters worn on the body, the true value of the quantity is given by the dose equivalent in an ICRU-tissue slab phantom at the depth specified by the quantity. In order to determine the value of $H_p(d)$ at the point of test, it is necessary to first use the reference calibration techniques briefly described in the preceding section for the type of radiation under consideration. When the physical quantity of interest has been determined, the appropriate conversion coefficient is used to calculate the value of the operational quantity.

Ideally, personal dosemeters (if fixed on the appropriate phantom) should have a dose equivalent response with an energy and angular dependence similar to those of the air kerma- or fluence-to-personal dose equivalent conversion coefficient. It is then assumed that the device measures personal dose equivalent correctly when it is fixed to the body.

Personal dosemeters are very often integrating devices measuring the accumulated dose equivalent. In calibrations the dose rate and the irradiation time must, therefore, be controlled to obtain the dose equivalent value of interest.

Calibrations of personal dosemeters as well as measurements of their response as a function of energy and direction of radiation incidence, should be carried out on the ISO water slab phantom [14], a water-filled slab (30 cm x 30 cm x 15 cm) and walls made of PMMA (see Fig. 4b). The front wall should be 2,5 mm and the other walls 10 mm thick. When this phantom is used, no corrections are applied for possible differences in backscatter between this phantom and the ICRU tissue slab phantom used to define the true value of the quantity.

The personal dosemeter is fixed to the front face of the phantom so that the reference direction of the dosemeter coincides with the normal to the front face of the phantom. The reference point of the dosemeter is placed at the point of test. When angular studies are performed, the dosemeter, together with the phantom, is rotated about an axis through the reference point.

If several dosemeters are irradiated simultaneously, they should be fixed to the front face of the phantom in a circular pattern around the centre of the front face so that no sensitive element of a dosemeter is positioned outside a circle 15 cm in diameter.

For dosemeters worn on the fingers, the ISO rod phantom should be used. This phantom is a PMMA cylinder of 19 mm in diameter and 300 mm in length. For dosemeters worn on the wrist or ankle, the ISO pillar phantom should be used. It is a water-filled hollow cylinder with PMMA walls, an outer diameter of 73 mm and a length of 300 mm. The cylinder walls are 2,5 mm and the end faces 10 mm thick [14,15,18]. If several dosemeters are irradiated simultaneously, they should be fixed to these phantoms so that they remain within a band 15 cm in length, centred on the long axis of the phantoms. At present, extremity dosemeters are used only to measure low-penetrating radiation (skin dose) and therefore they were calibrated on these phantoms in terms of $H_p(0,07)$ only. Conversion coefficients for $H_p(10)$ are not available for the extremity phantoms.

12. Reporting of results

The calibration of a personal dosemeter or area survey meter is not complete without the calibration being documented. National regulations often specify the details and format of both calibration records and certificates, as well as the frequency of calibration and the period of time for which calibration records are to be kept. The following list gives a general guideline for calibration records or certificates. A certificate should include:

1. Date and place of calibration,
2. Description of dosemeter or instrument (type and serial number),
3 Owner of device,
4. Descriptions of reference radiation sources and standard instruments,
5. Reference conditions, calibration conditions or standard test conditions,
6. Results with statement of uncertainties,
7. Names of the person who performed the calibration and of the reviewer,
8. Any special observations.

13. Statement of uncertainties

The results of a measurement yield only an approximation to the true value of the quantity being determined, therefore the results must be stated along with an estimation of the uncertainty. The methods used for analysing and reporting uncertainties are given in the joint BIPM/IEC/ISO/OIML Document "Guide to the Expression of Uncertainty in Measurement" [20]. On principle, all contributions to the total uncertainty are to be handled in the same way. This includes the uncertainties arising from random effects and those from systematic effects. The method for estimating the total uncertainty is to express each uncertainty component by a standard deviation equal to the positive square root of the estimated variance. The combined total uncertainty is then also expressed as a standard deviation and corresponds to the confidence interval with the coverage factor k=1 (confidence interval of about 66 %). Often a coverage factor of 2 is applied which corresponds to a confidence interval of about 95 %. It should always be noted which coverage factor k is applied. For details, the above-mentioned guide is to be referred to.

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21. Appendix: Definitions used in calibrations

Some definitions are given which are important for describing calibration procedures.

Influence Quantity

An influence quantity (parameter) is a quantity which may have an effect on the result of a measurement, without being the objective of the measurement itself.

Reference Conditions

The reference conditions represent the set of influence quantities for which the calibration factor is valid without any correction. The quantity to be measured is not an influence quantity and its value may be chosen freely depending on the properties of the device to be calibrated.

Standard Test Conditions

The standard test conditions represent the range of values of a set of influence quantities under which a calibration or determination of response is carried out.

Note: Ideally, calibrations should be carried out under reference conditions. As this is not always achievable (i.e. for ambient air pressure or temperature), a small interval around the reference values can be used.

Calibration Conditions

The calibration conditions are those within the range of standard test conditions actually prevailing during the calibration.

Point of Test

This is the point in the radiation field at which the reference point of a device being calibrated is placed and at which the conventional true value of the quantity to be measured is known.

Reference Point

The reference point of a device is the point which is placed at the point of test for calibrating or testing purposes.

Note: The distance of measurement refers to the distance between the radiation source and the reference point of the device being calibrated.

Reference Direction

The reference direction is the direction, in the co-ordinate system of the device being calibrated, with respect to which the incidence angle of the radiation is measured in unidirectional fields.

Reference Orientation

The reference orientation of the device is that for which the direction of the incident radiation coincides with the reference direction of the device.

Conventional True Value of a Quantity

The conventional true value of a quantity is the best estimate of the value of a quantity to be measured. It is determined by a primary or secondary standard, or by a reference instrument that has
been calibrated against a primary or secondary standard.

Backscatter Factor (Photons)

The backscatter factor is the ratio of air-kerma in front of a phantom to the air-kerma at the same position, free in air without a phantom. The field is considered to be unidirectional with a direction of incidence perpendicular to the phantom surface.

Note: The value of the backscatter factor depends on the point of test (distance from the surface and from the beam axis), on the beam diameter, the phantom size and material, and on the radiation energy.

Table 1: Radiation weighting factors, w_R, for different types of radiation.

Radiation	**Radiation weighting factor w_R**
Photons	1
Electrons[1)], muons	1
Neutrons: $E_n <$ 10 keV	5
$E_n =$ 10 keV to 100 keV	10
$E_n >$ 100 keV to 2 MeV	20
$E_n >$ 2 MeV to 20 MeV	10
$E_n >$ 20 MeV	5
Protons $E_p >$ 2 MeV (unless recoil protons)	5
α-particles, fission fragments, heavy nuclei	20

1) With the exception of Auger electrons from atoms bound to DNA

As an approximation to the step function ICRP has specified a smooth w_R function:

$$w_R = 5 + 17 \exp(-[\ln(2 E_n)]2/6)$$

with E_n neutron energy in MeV.

Table 2: Tissue weighting factors ,w_T, for various organs and tissues.

Organ or tissue	**Tissue weighting factor, w_T**
Gonads	0,20
Bone marrow (red)	0,12
Colon	0,12
Lung	0,12
Stomach	0,12
Bladder	0,05
Breast	0,05
Liver	0,05
Oesophagus	0,05
Thyroid	0,05
Skin	0,01
Bone surface	0,01
Remainder	0,05

1) Remainder tissues are adrenals, brain, upper large intestine, small intestine, kidney, muscle, pancreas, spleen, thymus and uterus.

The mean value of the equivalent doses of the ten remainder organs and tissues is to be multiplied by 0,05. If in a special case a single tissue or organ has an equivalent dose higher than each of the 12 individually defined organs and tissues, then this organ or tissue should get a weighting factor of 0,025 and the other 9 remainder tissues together a weighting factor of 0,025.

Table 3: Quotient of the ambient dose equivalent, $H^*(10)$, and the air kerma, K_a, for the photon radiation of some radionuclides. Only photon energies above 20 keV have been taken into consideration.

Nuclide	Half-life	Important photon energies in MeV	$H^*(10)/K_a$
^{24}Na	15,0 h	1,37 2,75	1.04
^{60}Co	5,27 a	1,17 1,33	1.02
^{124}Sb	60 d	0,60 — 2,09	1.04
^{131}I	8,02 d	0,08 - 0,72	1,11
^{137}Cs	30 a	0,66	1,06
^{182}Ta	114 d	0,06 — 1,23	1,03
^{192}Ir	74 d	0,30 - 0,61	1,12
^{224}Ra and progenies	1600 a	0,19 — 2,4	1,05
^{241}Am	458 a	0,06	1,48

Table 4: Quotient $H'(0,07,\alpha)/H'(0,07,0°)$ for three sources of the PTB's secondary beta-ray standard with compensation filter.

Angle of radiation incidence	$H'(0,07, \alpha)/H'(0,07, 0°)$ for beta radiation		
α in degrees	$^{90}Sr + ^{90}Y$	^{204}Tl	^{147}Pm
0	1,00	1,00	1,00
15	1,01	0,99	0,98
30	1,05	0,96	0,84
45	1,12	0,90	0,69
60	1,15	0,73	-
75	0,89	0,50	-

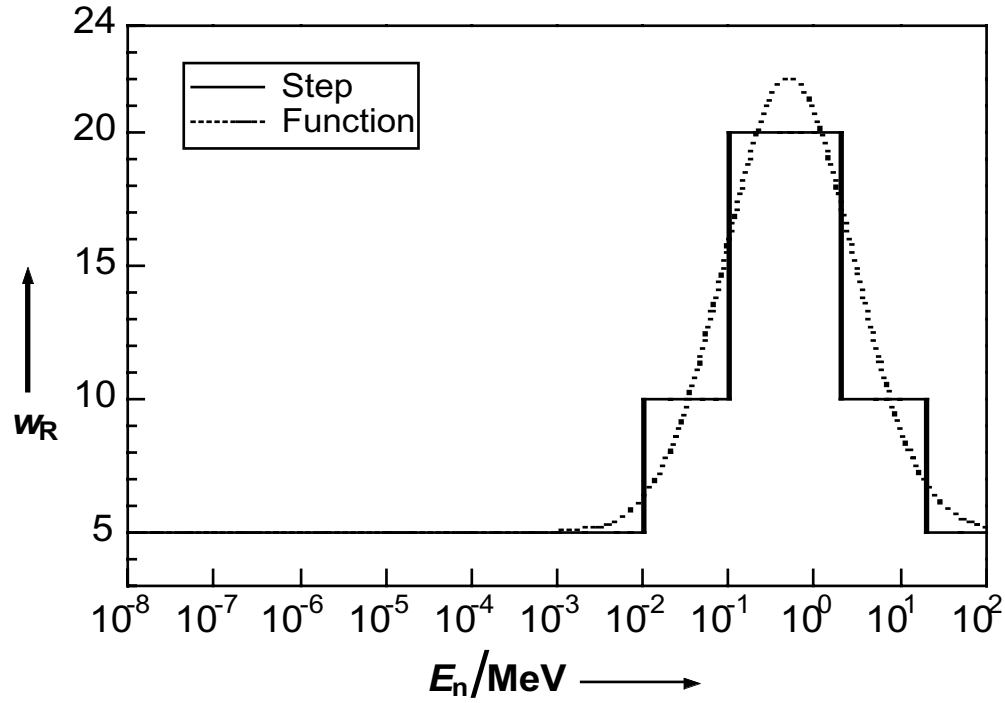


Fig. 1: Radiation weighting factor, w_R, for external neutron exposure versus neutron energy.

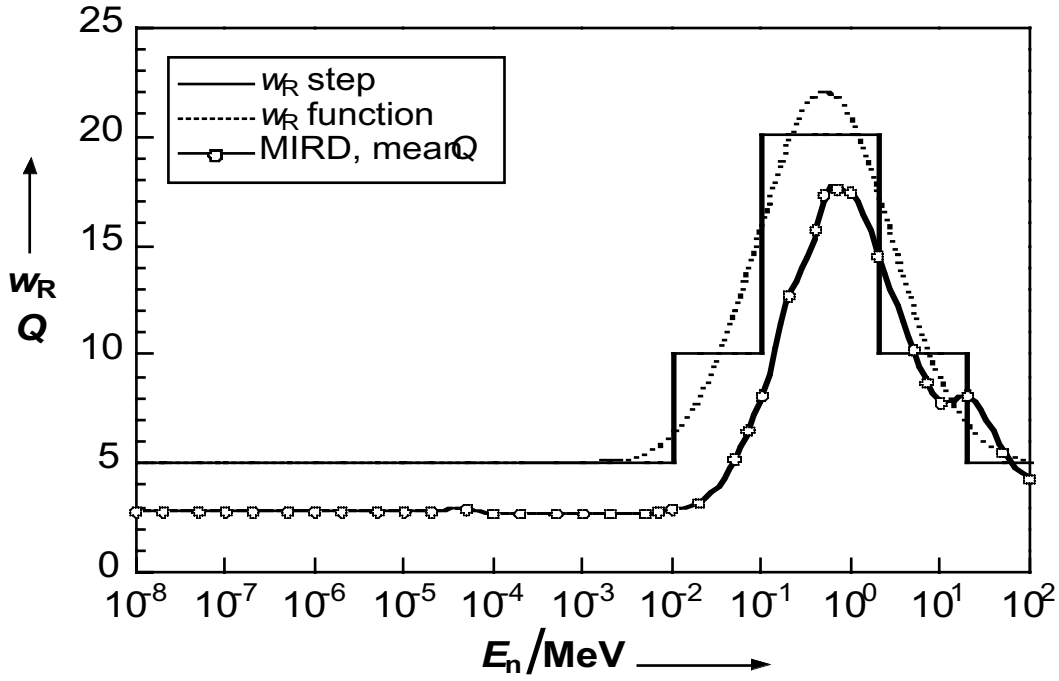


Fig. 2: Radiation weighting factor, w_R, for neutrons and a mean quality factor Q for neutron radiation versus neutron energy. The Q-values are based on MIRD phantom calculations using the actual $Q(L)$-relationship and calculating mean organ dose equivalent values.

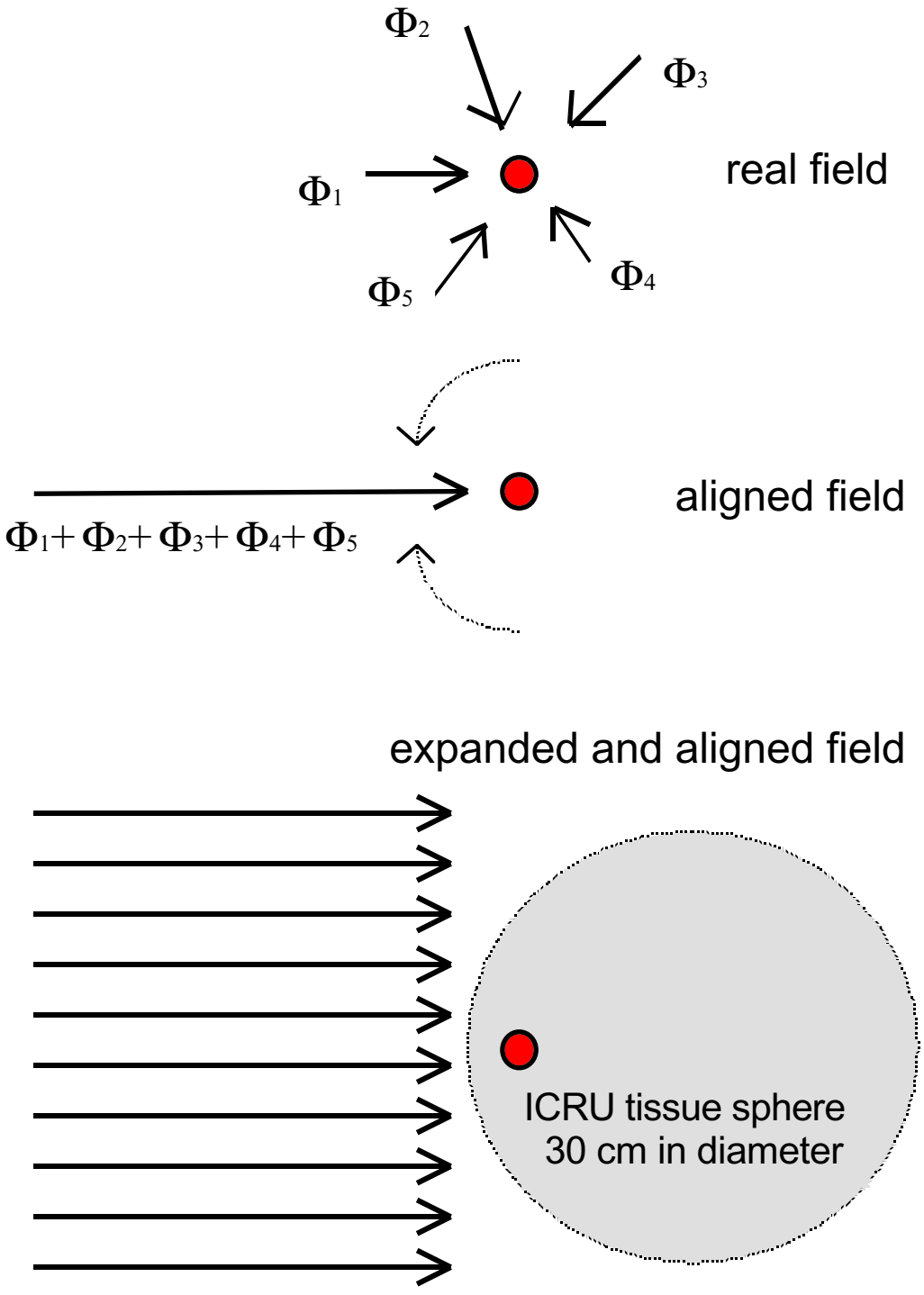
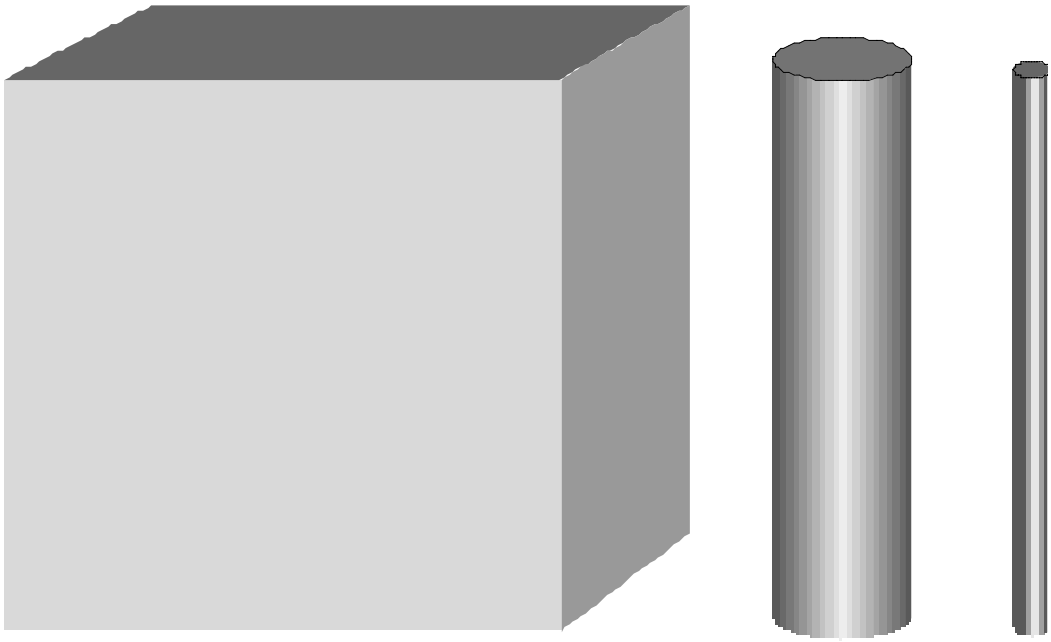
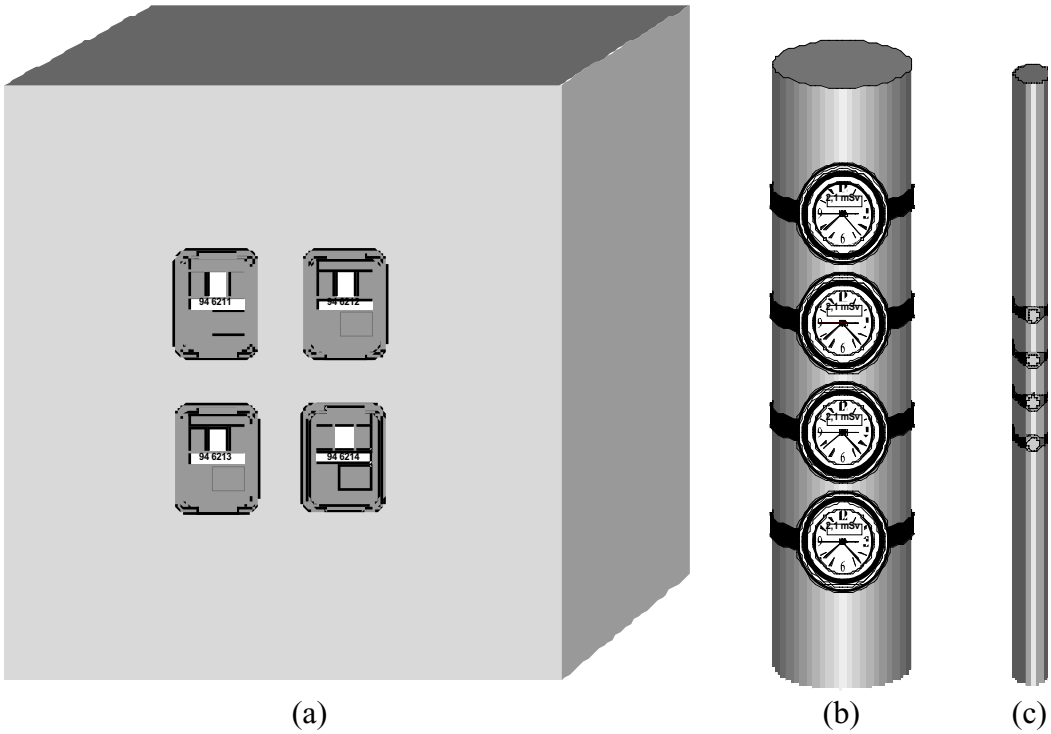


Fig. 3: Schematic diagram demonstrating the meaning of the expressions aligned and expanded .



(a) (b) (c)

Fig. 4a: ICRU-tissue phantoms for the calculation of conversion coefficients for $H_p(10)$ and $H_p(0,07)$.
(a) slab phantom (b) wrist phantom (c) finger phantom



(a) (b) (c)

Fig. 4b: ISO-phantoms used for the calibration of individual dosemeters.
(a) water slab phantom (b) water wrist phantom (c) PMMA finger phantom

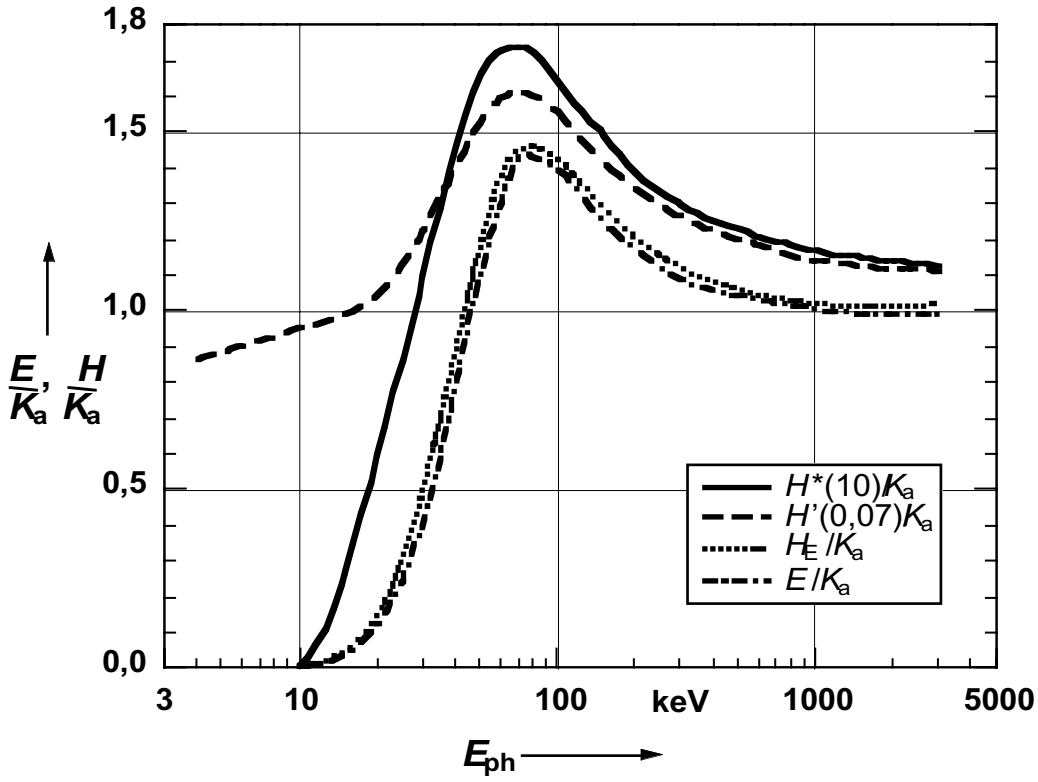


Fig. 5: Conversion coefficients for photon radiation versus photon energy.

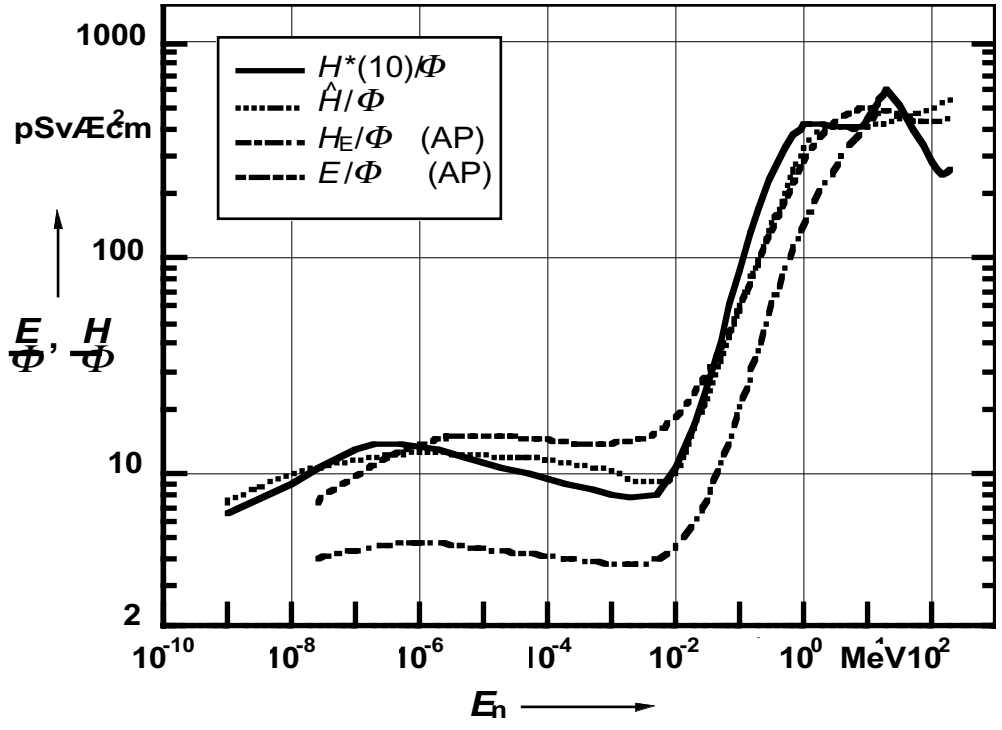


Fig. 6: Conversion coefficients for neutrons radiation versus neutron energy.