# 4 Radiological quantities and units

In this Chapter the fundamental quantities and units for ionising radiation and in addition specific quantities used in radiological protection are described.

## 4.1 Introduction

While radiation field quantities, quantities describing radioactivity and absorbed dose quantities are based on physical phenomena only, specific dose quantities in radiation protection as e.g. effective dose, include factors which are based on judgements about the biological response of tissues, e.g. due to cancer induction. These factors have been changed in the past in view of new research results and ideas. The definitions given are mainly based on ICRU Report 51 [93I1], ICRU Report 60 [98I1], ICRP Publication 60 [91I1] and the ISO Standards Handbook, Quantities and Units [93I2].

### Stochastic and non-stochastic quantities

Physical quantities are used to describe physical phenomena or objects. Many physical processes, e. g. the decay of radionuclides, the number of interactions in a small volume irradiated or the energy transferred, are subject to inherent fluctuations. This situation is described by *stochastic quantities* the values of which follow a probability distribution. Some times this may be a Poisson distribution which is uniquely determined by its mean value. In many other cases a quantity is defined by averaging in time or over a volume which results in a single value with no inherent fluctuation. Those quantities, e. g. fluence or absorbed dose, are called *non-stochastic quantities*.

### Units

A unit is a reference sample of a quantity with which other quantities of the same kind are compared. Every quantity is expressed as a product of a numerical value and a unit. Generally the use of the International System of Units (SI) as given by the BIPM [91BI] is recommended which is based on the 7 base units meter, kilogram, second, ampere, kelvin, mole and candela. Derived SI-units are often given special names like joule, becquerel or gray. Some other units are, however, generally used which are outside of the international system, e. g. the electron volt (eV) and the atomic mass unit (u) - and the time units minute, hour, day and year are also generally permitted.

Nevertheless, other units even if not recommended are still in use in radiation measurements and radiation protection. Table 1 presents some numerical relationships between those units and the SI-units recommended.

**Table 4.1.** Former units and its relations to SI-units

| Quantity | Symbol | SI-unit | Name | Former units | |
|---|---|---|---|---|---|
| absorbed dose | $D$ | J kg$^{-1}$ | gray (Gy) | rad | 1 rad = 0.01 Gy |
| exposure | $X$ | C kg$^{-1}$ | | roentgen | 1 R = 2.58 $10^{-4}$ J kg$^{-1}$ |
| dose equivalent | $H$ | J kg$^{-1}$ | sievert (Sv) | rem | 1 rem = 0.01 Sv |
| activity | $A$ | s$^{-1}$ | becquerel (Bq) | curie | 1 Ci = 3.7 $10^{10}$ Bq |
| potential alpha energy concentration | $c_p$ | J m$^{-3}$ | | Working level | 1 WL = 2.08 $10^{-5}$ J m$^{-3}$ = 1.30 $10^{8}$ MeV m$^{-3}$ |
| potential alpha energy exposure | $E_p$ | J h m$^{-3}$ | | Working level month ($T$ = 170 h) | 1 WLM = 3.54 $10^{-3}$ J h m$^{-3}$ = 2.21 $10^{10}$ MeV h m$^{-3}$ |

# 4.2 Radiation field quantities

Radiation field quantities are non-stochastic quantities defined at any point of a radiation field. Radiation fields may consist of various types of particles and the field quantities are always related to a specific particle type. This is usually expressed by adding the particle name to the quantity, e.g. photon fluence or neutron flux. There are two classes of radiation field quantities referring either to the number of particles or to the energy transported by them.

A radiation field of a specific particle type can be fully described by the number $N$ of particles, their distribution in energy as well as their spatial, directional and temporal distribution. This needs the definition of scalar and vectorial quantities. While in radiation dosimetry mostly scalar field quantities are used, vectorial quantities are often needed and applied in radiation transport theory and calculations. The radiation field quantities are defined in specifying the field in increasing detail.

## 4.2.1 Scalar radiation field quantities

**Particle number, radiant energy**

The **particle number**, $N$, is the number of particles that are emitted, transferred, or received.
Unit: 1

The **radiant energy**, $R$, is the energy (excluding rest energy) of the particles that are emitted, transferred, or received.
Unit: joule, J

For particles of energy $E$ (excluding rest energy), the radiant energy, $R$, is equal to $N \cdot E$.

The distributions, $N_E$ and $R_E$, of the particle number and the radiant energy with respect to energy are given by

$$N_E = \mathrm{d}N/\mathrm{d}E \qquad \text{and} \quad R_E = \mathrm{d}R/\mathrm{d}E \tag{4.2.1a+b}$$

where d$N$ is the number of particles with energy between $E$ and $E$ + d$E$ and d$R$ is their radiant energy.

**Flux, energy flux**

| | |
|---|---|
| The **flux**, $\dot{N}$, is the quotient of d$N$ by d$t$, where d$N$ is the increment of the particle number in the time interval d$t$.<br>$\dot{N} = \mathrm{d}N/\mathrm{d}t$<br>Unit: $s^{-1}$ | The **energy flux**, $\dot{R}$, is the quotient of d$R$ by d$t$, where d$R$ is the increment of the radiant energy in the time interval d$t$.<br>$\dot{R} = \mathrm{d}R/\mathrm{d}t$<br>Unit: W |

The term flux has often been employed for the quantity fluence rate (see below). This usage should be avoided.

**Fluence, energy fluence**

The quantity fluence is based on the idea of counting the number of particles incident or passing a small sphere. It is defined by:

| | |
|---|---|
| The **fluence**, $\Phi$, is the quotient of d$N$ by d$a$, where d$N$ is the number of particles incident on a sphere of cross-sectional area d$a$.<br>$\Phi = \mathrm{d}N/\mathrm{d}a$<br>Unit: $m^{-2}$ | The **energy fluence**, $\Psi$, is the quotient of d$R$ by d$a$, where d$R$ is the radiant energy incident on a sphere of cross-sectional area d$a$.<br>$\Psi = \mathrm{d}R/\mathrm{d}a$<br>Unit: J $m^{-2}$ |

The fluence is independent of the directional distribution of the particles passing the sphere. In calculations, fluence is often expressed in terms of the length of trajectories of particles passing a volume d$V$. The fluence, $\Phi$, is given by

$$\Phi = \mathrm{d}l/\mathrm{d}V \tag{4.2.2}$$

where d$l$ is the sum of the lengths of trajectories through this volume.

The distributions, $\Phi_E$ and $\Psi_E$, of the fluence and energy fluence with respect to energy are given by

$$\Phi_E = \mathrm{d}\Phi/\mathrm{d}E \qquad \text{and} \quad \Psi_E = \mathrm{d}\Psi/\mathrm{d}E \tag{4.2.3a+b}$$

These quantities are often called spectral fluence and spectral energy fluence, respectively.

**Fluence rate, energy fluence rate**

The temporal distribution of the fluence and energy fluence is generally of interest. This results in the following definitions:

| | |
|---|---|
| The **fluence rate**, $\dot{\Phi}$, is the quotient of d$\Phi$ by d$t$, where d$\Phi$ is the increment of the fluence in the time interval d$t$.<br>$\dot{\Phi} = \mathrm{d}\Phi/\mathrm{d}t$<br>Unit: $m^{-2}$ $s^{-1}$ | The **energy fluence rate**, $\dot{\Psi}$, is the quotient of d$\Psi$ by d$t$, where d$\Psi$ is the increment of the energy fluence in the time interval d$t$.<br>$\dot{\Psi} = \mathrm{d}\Psi/\mathrm{d}t$<br>Unit: J $m^{-2}$ |

The *fluence rate* has often been termed *particle flux density*. Because the term *density* mostly characterises a mass density ($kg^{-1}$), it is recommended to use the term *fluence rate* and not *particle flux density*.

**Particle radiance, energy radiance**

| | |
|---|---|
| The **particle radiance**, $\dot{\Phi}_\Omega$, is the quotient of $\mathrm{d}\dot{\Phi}$ by $\mathrm{d}\Omega$, where $\mathrm{d}\dot{\Phi}$ is the fluence rate of particles propagating within a solid angle $\mathrm{d}\Omega$ around a specified direction. $\dot{\Phi}_\Omega = \mathrm{d}\dot{\Phi} / \mathrm{d}\Omega$ Unit: $m^{-2}\,s^{-1}\,sr^{-1}$ | The **energy radiance**, $\dot{\Psi}_\Omega$, is the quotient of $\mathrm{d}\dot{\Psi}$ by $\mathrm{d}\Omega$, where $\mathrm{d}\dot{\Psi}$ is the energy fluence rate of particles propagating within a solid angle $\mathrm{d}\Omega$ around a specified direction. $\dot{\Psi}_\Omega = \mathrm{d}\dot{\Psi} / \mathrm{d}\Omega$ Unit: $W\,m^{-2}\,sr^{-1}$ |

The specification of a direction $\Omega$ requires two variables. In a spherical coordinate system with a polar angle, $\theta$, and an azimuthal angle, $\varphi$, $\mathrm{d}\Omega$ is equal to $\sin\theta\,\mathrm{d}\theta\,\mathrm{d}\varphi$.

The distribution of particle radiance and energy radiance with respect to energy are given by

$$\dot{\Phi}_{\Omega,E} = \mathrm{d}\dot{\Phi} / \mathrm{d}\Omega\,\mathrm{d}E \quad \text{and} \quad \dot{\Psi}_{\Omega,E} = \mathrm{d}\dot{\Psi} / \mathrm{d}\Omega\,\mathrm{d}E \qquad (4.2.4a+b)$$

## 4.2.2 Vectorial radiation field quantities

Radiometric quantities are often used to describe the flow of radiation in specific directions. This needs the definition of vectorial quantities. For example, the scalar angular differential quantities like particle radiance and energy radiance are transferred to vectorial quantities by multiplication with the unit vector $\boldsymbol{\Omega}$ in a specific direction. Vectorial quantities are

| | | | |
|---|---|---|---|
| vectorial particle radiance, $\dot{\boldsymbol{\Phi}}_\Omega$ | with | $\dot{\boldsymbol{\Phi}}_\Omega = \boldsymbol{\Omega}\cdot\dot{\Phi}_\Omega$ | unit: $m^{-2}\,s^{-1}\,sr^{-1}$, |
| vectorial energy radiance, $\dot{\boldsymbol{\Psi}}_\Omega$ | with | $\dot{\boldsymbol{\Psi}}_\Omega = \boldsymbol{\Omega}\cdot\dot{\Psi}_\Omega$ | unit: $W\,m^{-2}\,sr^{-1}$, |
| vectorial fluence rate, $\dot{\boldsymbol{\Phi}}$ | with | $\dot{\boldsymbol{\Phi}} = \int \dot{\boldsymbol{\Phi}}_\Omega\,\mathrm{d}\Omega$ | unit: $m^{-2}\,s^{-1}$, |
| vectorial energy fluence rate, $\dot{\boldsymbol{\Psi}}$ | with | $\dot{\boldsymbol{\Psi}} = \int \dot{\boldsymbol{\Psi}}_\Omega\,\mathrm{d}\Omega$ | unit: $W\,m^{-2}$, |
| vectorial fluence, $\boldsymbol{\Phi}$ | with | $\boldsymbol{\Phi} = \int \dot{\boldsymbol{\Phi}}\,\mathrm{d}t$ | unit: $m^{-2}$, |
| vectorial energy fluence, $\boldsymbol{\Psi}$ | with | $\boldsymbol{\Psi} = \int \dot{\boldsymbol{\Psi}}\mathrm{d}t$ | unit: $J\,m^{-2}$. |

A detailed description is given in ICRU Report 60 [98I1]. The distribution of a quantity with respect to energy of the particle considered is described by an index $E$ similar to the scalar quantities. For example, the distribution of the vectorial particle radiance is given by

$$\dot{\boldsymbol{\Phi}}_{\Omega,E} = \boldsymbol{\Omega}\cdot\dot{\Phi}_{\Omega,E} \qquad \text{unit: } m^{-2}\,s^{-1}\,sr^{-1}\,MeV^{-1}.$$

## 4.3 Interaction coefficients and quantities

Ionising radiation is either charged (e.g. electrons, positrons, protons and α-particles) or uncharged (e.g. photons and neutrons). This dominates the main interaction with matter. While charged particles (called *directly ionising particles*) are mainly slowed down by electromagnetic interactions with electrons of the target atoms, the uncharged particles (*indirectly ionising particles*) interact with matter in separated events. Indirectly ionising particles are either absorbed or its energy and direction is altered.

The probabilities of specific interactions between radiation and matter are characterized by interaction coefficients. They refer to specific interaction processes, type and energy of radiation and the matter involved. The definition of those coefficients important for dosimetry and related quantities are given in this Section.

### 4.3.1 Cross section

The cross section is the most fundamental interaction coefficient. It is defined as follows.

The **cross section**, $\sigma$, of a target entity, for a particular interaction produced by incident particles is the quotient of $P$ by $\Phi$, where $P$ is the probability of that interaction for a single target entity when subjected to the particle fluence, $\Phi$. It is

$\sigma = P/\Phi$ unit: $m^2$.

A special unit often used for the cross section is the barn (b) with $1\ b = 10^{-28}\ m^2$.

Cross sections mostly vary with the energy of the incident radiation (notation: $\sigma(E)$). The distribution of a cross section with respect to the energy and direction of the emitted radiation is often called differential cross section ($\mathrm{d}\sigma/\mathrm{d}\Omega$ : angular differential cross section, $\mathrm{d}\sigma/\mathrm{d}E$: energy differential cross section, $\mathrm{d}^2\sigma/\mathrm{d}E\mathrm{d}\Omega$ : energy and angular differential cross section).

The total cross section, $\sigma_T$, is the sum of the cross sections of all possible interaction channels for an incident particle of a given type and energy and a given target material.

### 4.3.2 Mass attenuation coefficient and mass energy transfer coefficient

For an infinite small parallel beam of uncharged radiation, the interaction of radiation with matter results in an attenuation of the incident beam with depth in material. This can be described by the relation

$$\frac{\mathrm{d}N}{N} = \mu \cdot \mathrm{d}l \qquad (4.3.1)$$

where $\mathrm{d}N/N$ is the fraction of particles that experience interactions in traversing a distance $\mathrm{d}l$ in the material. $\mu$ is the *linear attenuation coefficient*. The reciprocal of $\mu$ is called the mean free path $\lambda$ of an uncharged particle. In first order $\mu$ is proportional to the density $\rho$ of a material. This leads to the definition of the *mass attenuation coefficient*, $\mu/\rho$ .

$$\frac{\mu}{\rho} = \frac{1}{\rho\,\mathrm{d}l}\frac{\mathrm{d}N}{N} \qquad \text{unit: } \mathrm{m^2\,kg^{-1}} \qquad (4.3.2)$$

The mass attenuation coefficient is related to the total cross section by

$$\frac{\mu}{\rho} = \frac{N_\mathrm{A}}{M}\sigma_T = \frac{N_\mathrm{A}}{M}\sum_\mathrm{J}\sigma_\mathrm{J} \tag{4.3.3}$$

where $N_\mathrm{A}$ is the Avogadro constant and $M$ the molar mass of the material considered. $\sigma_\mathrm{J}$ are the cross section related to the interaction of type J in this material.

For uncharged particles the transfer of energy to charged particles in the material is of high interest in dosimetry. This is expressed by the *mass energy transfer coefficient*, $\mu_\mathrm{tr}/\rho$ , which is defined by

$$\frac{\mu_\mathrm{tr}}{\rho} = \frac{1}{\rho\,\mathrm{d}l}\frac{\mathrm{d}R_\mathrm{tr}}{R} \qquad \text{unit: m}^2\,\text{kg}^{-1} \tag{4.3.4}$$

where $\mathrm{d}R_\mathrm{tr}/R$ is the fraction of incident radiant energy that is transferred to kinetic energy of charged particles by interactions when traversing a distance $\mathrm{d}l$ in the material of density $\rho$. If incident uncharged particles of a given type and energy can produce several types of interactions in a material, $\mu_\mathrm{tr}/\rho$ can be expressed in terms of the partial cross Sections, $\sigma_\mathrm{J}$, by the relation

$$\frac{\mu_\mathrm{tr}}{\rho} = \frac{N_\mathrm{A}}{M}\sum_\mathrm{J} f_\mathrm{J}\sigma_\mathrm{J} \tag{4.3.5}$$

where $f_\mathrm{J}$ is the average fraction of the incident particle energy that is transferred to kinetic energy of charged particles in an interaction of type J.

The mass energy transfer coefficient is related to the mass attenuation coefficient by

$$\frac{\mu_\mathrm{tr}}{\rho} = \frac{\mu}{\rho}f = \frac{\mu}{\rho}\frac{\sum_\mathrm{J} f_\mathrm{J}\sigma_\mathrm{J}}{\sum_\mathrm{J}\sigma_\mathrm{J}} \tag{4.3.6}$$

For $\mu_\mathrm{tr}/\rho$ of a compound material the material is usually treated as consisting of independent atoms and the contributions from the different components are summed considering their partial density.

A small part of the energy transferred to charged particles may not be locally absorbed in the material but further transferred to secondary photons (e.g. Bremsstrahlung). Therefore, an additional coefficient, the *mass energy absorption coefficient*, $\mu_\mathrm{en}/\rho$, is defined by the product of $\mu_\mathrm{tr}/\rho$ and (1-$g$) where $g$ is the fraction of the energy of charged particles that is lost in radiative processes in the material.

Data of mass energy transfer and mass energy absorption coefficients are given by Seltzer [93Se].

For neutron radiation, the *kerma coefficient* $K/\Phi$ (kerma per unit neutron fluence, often called *kerma factor*) is mostly used instead of $\mu_\mathrm{tr}/\rho$ for characterising the energy transfer (see 4.4.1). Data of kerma coefficients for biological important materials from thermal to 150 MeV neutrons are published by Chadwick et al. [99Ch] and in ICRU Report 63 [00I1].

## 4.3.3 Mass stopping power and linear energy transfer (LET)

Charged particles passing matter loose energy by collisions with electrons, by emission of bremsstrahlung in the electric fields of nuclei or atomic electrons or by elastic Coulomb scattering and inelastic nuclear processes on atoms or nuclei. This effect is characterized by the *mass stopping power* $S/\rho$ for charged particles in a material with density $\rho$. It is

$$\frac{S}{\rho} = \frac{1}{\rho}\frac{\mathrm{d}E}{\mathrm{d}l} \qquad \text{unit: J m}^2\,\text{kg}^{-1} \tag{4.3.7}$$

where $\mathrm{d}E$ is the energy lost by a charged particle in traversing a distance $\mathrm{d}l$ in the material. $S = \mathrm{d}E/\mathrm{d}l$ is called the *linear stopping power*. $E$ may be given in eV and the unit of $S/\rho$ may then expressed in eV $\mathrm{m}^2\,\mathrm{kg}^{-1}$ or other multiples like MeV $\mathrm{cm}^2\,\mathrm{g}^{-1}$, for example.

The transfer of energy from the primary charged particle to secondary electrons is of specific interest in dosimetry, especially to those electrons receiving a kinetic energy less than a given value only. They will locally deposit their energy near to the track of the primary particle. This led to the definition of the quantity *linear energy transfer* (LET) or *restricted linear electronic stopping power* $L_\Delta$ given by

$$L_\Delta = \frac{\mathrm{d}E_\Delta}{\mathrm{d}l} \qquad \text{unit: J m}^{-1}\text{, often used keV }\mu\text{m}^{-1} \qquad (4.3.8)$$

where $\mathrm{d}E_\Delta$ is the energy lost by a charged particle due to electronic collisions when traversing a distance $\mathrm{d}l$ minus the sum of the kinetic energies of all electrons released with kinetic energies in excess of $\Delta$. This definition given in ICRU Report 60 [98I1] differs from earlier ones [80I1] in a way that $L_\Delta$ now includes the binding energies for all collisions and the threshold of the kinetic energy of the released electrons is now $\Delta$ instead of $\Delta$ minus the binding energy.

$\Delta$ is often given in eV and then the notation $L_{100}$ means an energy cutoff of 100 eV. $L_\infty$ is often called *unrestricted linear energy transfer* $L$ and is equal to $S_{\mathrm{el}}$, the *electronic stopping power* due to collisions with electrons.

### 4.3.4 Mean energy expended in a gas per ion pair formed

In dosimetry, where often charge measurements due to ionisation in gases are the basis of dose determinations, the kinetic particle energy necessary to create an ion pair is of general interest. This led to the definition of the *mean energy expended in a gas per ion pair formed* $W$. It is

$$W = \frac{E}{N} \qquad \text{unit: J} \qquad (4.3.9)$$

where $N$ is the number of ion pairs when the initial kinetic energy $E$ of the charged particle is completely dissipated in the gas considered. This definition includes also the ions produced by secondary electrons or bremsstrahlung.

## 4.4 Quantities related to energy transfer

### 4.4.1 Stochastic quantities

The energy transfer from incident particles to a target material is a stochastic process. For example, the energy deposition along a track of a charged particle is randomly distributed. The values of stochastic quantities are, therefore, subject to inherent fluctuations. They generally follow a probability distribution and mean values may be given. For example, a Poisson distribution is already uniquely determined by its mean value. Stochastic quantities are often used in microdosimetry in order to describe the energy transfer to very small volumes.

#### 4.4.1.1 Energy deposit and energy imparted

The *energy deposit* $\varepsilon_i$, is the energy deposited in a single interaction i, thus

$$\varepsilon_i = \varepsilon_{in} - \varepsilon_{out} + Q \qquad \text{unit: J, often used eV}$$

where $\varepsilon_{in}$ is the energy of the incident ionising particle (excluding rest energy), $\varepsilon_{out}$ the sum of energies of all ionising particles leaving the interaction (excluding rest energy) and $Q$ the change in the rest energies of the nucleus and all particles involved in the interaction. $Q > 0$ means a decrease of rest energy, $Q < 0$ an increase.

The total energy transferred to matter in a given volume is often of interest. The *energy imparted* $\varepsilon$ to the matter in a given volume is the sum of all energy deposits in the volume

$$\varepsilon = \sum \varepsilon_i \qquad \text{unit: J, often used eV}$$

The *mean energy imparted* $\bar{\varepsilon}$ to the matter in a given volume is a non-stochastic quantity and can be expressed in terms of the radiant energy $R_{in}$ (sum of all radiant energies of the incoming particles) and $R_{out}$ (sum of the radiant energies of all outgoing particles). It is

$$\bar{\varepsilon} = R_{in} - R_{out} + \sum Q\,, \qquad \text{unit: J, often used eV}$$

#### 4.4.1.2 Lineal energy and specific energy

Corresponding to the non-stochastic quantity LET the stochastic quantity *lineal energy y* is defined by the quotient of $\varepsilon_s$ by $\bar{l}$, where $\varepsilon_s$ is the energy imparted to the matter in a given volume by a single energy deposition event and $\bar{l}$ is the mean chord length of that volume, thus

$$y = \frac{\varepsilon_s}{\bar{l}} \qquad \text{unit: J m}^{-1}\text{, mostly used keV µm}^{-1}$$

This quantity is mainly used in microdosimetry, especially in measurements with low-pressure tissue-equivalent proportional counters where single event distributions in terms of $y$ are measured.

The *specific energy* (imparted) $z$ is the quotient of $\varepsilon$ by $m$, where $\varepsilon$ is the energy imparted to matter of mass $m$. It is

$$z = \frac{\varepsilon}{m} \qquad \text{unit: gray (Gy), } 1\text{ Gy} = 1\text{ J kg}^{-1}$$

The specific energy includes the energy transferred to the matter $m$ from all events involved.

### 4.4.2 Non-stochastic quantities

#### 4.4.2.1 Kerma, kerma rate

The transfer of energy from uncharged particles (indirectly ionising particles, e.g. photons or neutrons) to matter is performed by the liberation and slowing down of secondary charged particles in this matter. This led to the definition of the quantity kerma. The *kerma K* is the quotient of $dE_{tr}$ by $dm$, where $dE_{tr}$ is the sum of the kinetic energies of all charged particles liberated by uncharged particles in a mass $dm$ of material. It is given by

$$K = \frac{dE_{tr}}{dm}\,, \qquad \text{unit: gray (Gy), } 1\text{ Gy} = 1\text{ J kg}^{-1}$$

Kerma is a non-stochastic quantity. For a very small mass element, however, the energy transfer $\mathrm{d}E_{\mathrm{tr}}$ underlies in principle stochastic fluctuations. In this case a non-stochastic quantity means that $\mathrm{d}E_{\mathrm{tr}}$ is seen to be the expectation value of the sum of energies of liberated charged particles.

For monoenergetic uncharged particles of energy $E$ the kerma is related to the fluence by

$$K = \Phi E\,(\mu_{\mathrm{tr}}/\rho) \tag{4.4.1}$$

For a given energy distribution $\Phi_E$ of the uncharged particles the kerma can be calculated by

$$K = \int \Phi_E\, E\,(\mu_{\mathrm{tr}}/\rho)\,\mathrm{d}E \tag{4.4.2}$$

For neutrons, the quotient of $K$ by $\Phi$, is called kerma coefficient (often also called kerma factor) where $\Phi$ is the neutron fluence (see 4.3.2).

The kerma rate, $\dot{K}$ , is the quotient of d$K$ by d$t$, where d$K$ is the increment of $K$ in the time interval d$t$.

$$\dot{K} = \frac{\mathrm{d}K}{\mathrm{d}t}, \qquad \text{unit: Gy s}^{-1}.$$

### 4.4.2.2 Absorbed dose, absorbed dose rate

The quantity absorbed dose is a basic quantity in radiation dosimetry and relevant to all types of ionising radiation whether directly or indirectly ionising. It is a non-stochastic quantity and defined by:

$$D = \frac{\bar{\varepsilon}}{\mathrm{d}m} \qquad \text{unit: gray (Gy), 1 Gy = 1 J kg}^{-1}$$

where $\bar{\varepsilon}$ is the mean energy imparted to the matter of mass d$m$.

While kerma is related only to those secondary charged particles produced in d$m$ but transferring their energy to matter partially also outside d$m$, absorbed dose includes all energy transferred to d$m$ partially also from secondary charged particles produced outside but entering d$m$. Only under *charged particle equilibrium* and negligible radiation losses, however, the values of absorbed dose and kerma are equal in a homogeneous material.

The absorbed dose rate, $\dot{D}$ , is the quotient of d$D$ by d$t$, where d$D$ is the increment of the absorbed dose in the time interval d$t$. It is

$$\dot{D} = \frac{\mathrm{d}D}{\mathrm{d}t} \qquad \text{unit: Gy s}^{-1}$$

### 4.4.2.3 Exposure, exposure rate

The quantity exposure is related to the production of charges in gas by ionising radiation. Historically its definition is elder than kerma or absorbed dose. Its use, however, is restricted to photons only. The exposure $X$ is the quotient of d$Q$ by d$m$, where d$Q$ is the absolute value of the total charge of the ions of one sign produced in air when all the electrons and positrons liberated or created by photons in air of mass d$m$ are stopped in air.

$$X = \frac{\mathrm{d}Q}{\mathrm{d}m} \qquad \text{unit: C kg}^{-1}\text{ (former: roentgen, R)}$$

It should be noted that in this definition the charges due to ionisation arising from the absorption of bremsstrahlung emitted by the electrons is not included in d$Q$.

The exposure rate, $\dot{X}$ , is the quotient of d$X$ by d$t$, where d$X$ is the increment of the exposure in the time interval d$t$. It is

$$\dot{X} = \frac{\mathrm{d}X}{\mathrm{d}t} \qquad \text{unit: C kg}^{-1}\text{ s}^{-1}$$

# 4.5 Dose quantities in radiation protection

## 4.5.1 Concept of radiation protection quantities

The development of dosimetric concepts and the definition of specific quantities for use in radiation protection have 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 [91I1], the ICRP has published its most recent general recommendations for radiation protection including a system of quantities.

The ICRP and ICRU have developed a hierarchy of quantities for radiation protection applications comprising primary limiting dose quantities (called "protection quantities") taking account of human body properties and operational dose quantities for 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 and the exposure risk, were not explicitly considered in the definition of these quantities.

*Operational quantities* are dose equivalent quantities defined for use in radiation protection measurements related to external exposure (area or individual monitoring). They are needed for monitoring external exposures because

- protection quantities are generally not measurable,
- for area monitoring a point quantity is needed,
- a non-isotropic human-body related quantity like effective dose is not appropriate in area monitoring,
- instruments for radiation monitoring need to be calibrated in terms of an operational quantity.

Operational quantities usually provide an estimate or upper limit for the value of the limiting quantities due to an exposed, or potentially exposed, person under most irradiation conditions. They are often used in practical regulations instead of the primary limiting quantities.

For internal exposure, however, other methods are used and no similar dose quantities have been defined. In this case organ doses or effective dose are estimated from the information on intake or excretion of radioactive substances. Model based conversion coefficients exist for a large number of radionuclides relating the intake to organ doses and effective dose (see 4.6 and Chapter 7).

Both, protection and operational quantities can be related to "radiation field quantities" (see Sect. 4.2) or air kerma (see Sect. 4.3) which are point quantities defined in any point of a radiation field and whose units are directly realised through primary standards at national standards laboratories since long time. The numerical relations (conversion coefficients) between those quantities and the protection or operational dose quantities are given in Chapter 6.

## 4.5.2 Protection quantities

In 1977 the ICRP [77I1] introduced the tissue (or organ) 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 by introducing tissue weighting factors [73Jac]. Although in general this concept was not changed by ICRP 60 [91I1] in 1990, important modifications, however, were introduced e.g. replacing dose equivalent quantities by equivalent dose quantities. The present system of quantities is summarised in the following.

### 4.5.2.1 Absorbed dose and equivalent dose in a tissue or organ

The *absorbed dose in a tissue or organ* $D_T$ 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 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 range.

The *equivalent dose in a tissue or organ* is defined by

$$H_T = \sum_R w_R D_{T,R} \qquad \text{unit: sievert (Sv) (1 Sv = 1 J kg}^{-1}\text{)}$$

where $D_{T,R}$ is the mean organ dose in the tissue or organ 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 [77I1]. The sum is taken over all types of radiation involved.

### 4.5.2.2 Radiation weighting factors

For external irradiation, the values of the radiation weighting factors $w_R$ are given by the parameters of the external radiation field only (type and energy 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 and 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 body, is not explicitely considered. This effect is mainly important in the case of incident neutrons where at low energies 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 4.2). 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. For this case there exists no ICRP recommendation until now.

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. 4.1). In practice, neutron fields contain neutrons with a broad energy distribution. Because the use of a continuous $w_R$-function for effective dose estimation is more appropriate in these cases it is recommended to apply the continuous function in any case to avoid ambiguities. Then the weighting factor for neutrons ranges from 5 to 22 depending on neutron energy with its maximum value at 500 keV. All conversion coefficients for neutrons published in ICRP 74 [96I1] and ICRU 54 [98I2] are based on the continuous function only (see Chapt. 6).

**Table 4.2**. Radiation weighting factors $w_R$

| Radiation | | $w_R$ |
|---|---|---|
| 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 introduced for neutrons 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.

The radiation weighting factor for incident external protons with energies above 2 MeV has been set to 5. It is, however, questioned if this value is appropriate for protons of all energies above 2 MeV. There exists a general opinion that a weighting factor of about 2 seems to be more realistic for high energy protons above about 5 to 10 MeV. External protons of lower energies have a small range in tissue and contribute to the skin dose only.

#### 4.5.2.3 Effective dose

The effective dose $E$ is the weighted sum of the equivalent doses in tissues and organs T:

$$E = \sum_{\mathrm{T}} w_{\mathrm{T}} H_{\mathrm{T}} \qquad \text{with } \sum_{\mathrm{T}} w_{\mathrm{T}} = 1 \qquad \text{unit: sievert (Sv)} \tag{4.5.1}$$

where $w_T$ are *tissue weighting factors* characterising the relative sensitivity of the various tissues with respect to stochastic effects resulting from ionising radiation exposure and $H_T$ is the equivalent dose in one of the 13 specified tissues and organs (see Sect. 4.5.2.4).

The effective dose is a quantity which is not sex specific or dependent on age of a person. In principle, the effective dose is determined by taking the dose values in all tissues and organs of an individual person. Those data, however, are never measurable. For external exposure, therefore, always calculated conversion coefficients are used which relate the external radiation field to the doses in the tissues and organs (see Chapt. 6).

Following ICRP Report 74 [96I1], the effective dose is then calculated by

$$E = w_{\mathrm{breast}} H_{\mathrm{breast,\,female}} + \sum_{\mathrm{T}\neq\mathrm{breast}} w_{\mathrm{T}} \frac{H_{\mathrm{T,\,male}} + H_{\mathrm{T,\,female}}}{2} \tag{4.5.2}$$

Under a given exposure condition (radiation field, direction of radiation incidence, exposure period), therefore, all persons are given the same effective dose value independent of sex and age.

#### 4.5.2.4 Tissue Weighting Factors

The definition of effective dose takes care of the different radiosensitivity of the various organs and tissues in the human body with respect to cancer induction and mortality by introducing tissue weighting factors. Twelve tissues and organs are specified with individual weighting factors $w_T$. The values have been developed from a reference population of equal numbers of both sexes and a wide range of ages. They are applied to workers, to the whole population, and to either sex including children and the unborn child (foetus).

An additional "remainder" tissue with a weighting factor of 0.05 is also defined [91I1]. Its dose is given by the mean value from ten specified tissues and organs (see Table 4.3). The upper large intestine formerly included in the remainder, is now considered as part of the colon and has been replaced by the extrathoracic airways [93I3, 94I2]. While in the calculation of conversion coefficients for the intake of radionuclides the remainder dose is obtained from the mass-weighted doses to the single tissues and organs involved, the coefficients for external exposure are calculated giving identical weights to each of the remainder tissues [96I1].

**Table 4.3.** Tissue weighting factors $w_T$

| Organ or tissue | $w_T$ |
|---|---|
| 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[1)] | 0.05 |

1) "Remainder" tissues are adrenals, brain, extrathoracic airways, 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.

#### 4.5.2.5 Committed or collective equivalent dose and effective dose

Several subsidiary dosimetric quantities have been additionally defined. After an intake of radionuclides to a body these nuclides may give rise to equivalent doses in different tissues and organs of the body spread over long time depending on the physical and biological half-life of the radionuclides and their biokinetic behaviour in the body.

The time integral of the equivalent dose rate is called the *committed equivalent dose* $H_T(\tau)$, in a tissue or organ T, where $\tau$ is the integration time (in years) following the intake at time $t_0$.

$$H_T(\tau) = \int_{t_0}^{t_0+\tau} \dot{H}_T(t)\mathrm{d}t \qquad (4.5.3)$$

If $\tau$ is not specified, it is implied that its value is 50 y for workers and from intake up to age 70 years for members of the public including children. For patients in nuclear medicine, the integration may run from $t_0$ to $\infty$ because the biological and physical half-life of the radionuclides applied is much less than 10 y.

The same specification holds for the quantity *committed effective dose* $E(\tau)$ defined by the weighted sum of $H_T(\tau)$ over all specified tissues and organs T.

$$E(\tau) = \sum_T w_T \cdot H_T(\tau) \tag{4.5.4}$$

All dosimetric quantities referred before are related to a single tissue or organ of a single individual. Often it may be of interest to quantify the total dose a number of people received from one source or one release of radioactive material. The relevant quantity is called *collective equivalent dose* $S_T$ in a tissue or organ T and is defined by

$$S_T = \int_0^\infty H_T \cdot \frac{dN}{dH_T} dH_T \qquad \text{unit: man sievert (man Sv)} \tag{4.5.5}$$

where $(dN/dH_T)dH_T$ is the number of individuals receiving an equivalent dose between $H_T$ and $H_T$+$dH_T$ or by

$$S_T = \sum_i \overline{H}_{T,i} N_i \tag{4.5.6}$$

where $N_i$ is the number of individuals in a subgroup i receiving a mean tissue equivalent dose $\overline{H}_{T,i}$. The summed effective doses of all members of a group or population is called the *collective effective dose S* defined in a similar way by

$$S = \int_0^\infty E \cdot \frac{dN}{dE} dE \qquad \text{or} \qquad S = \sum_i \overline{E}_i \cdot N_i \qquad \text{unit: man Sv} \tag{4.5.7a+b}$$

where $N_i$ is a subgroup i receiving a mean equivalent dose $\overline{E}_i$.

## 4.5.3 Operational quantities

### 4.5.3.1 Dose equivalent and quality factor

The radiation protection quantity dose equivalent $H$ is defined by

$$H = Q\,D \qquad \text{unit: Sv (1 Sv = 1 J kg}^{-1}\text{)}$$

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 [77I1]. In principle, this concept of $Q$ has not been changed by ICRP 60 [91I1], 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:

$$Q(L) = \begin{cases} 1 & \text{for } L < 10\ \text{keV/}\mu\text{m} \\ 0.32\,L - 2.2 & \text{for } 10 \le L \le 100\ \text{keV/}\mu\text{m} \\ 300/\sqrt{L} & \text{for } L > 100\ \text{keV/}\mu\text{m} \end{cases} \tag{4.5.8}$$

The quality factor $Q$ at a point in tissue is then given by [86I1]:

$$Q = \frac{1}{D} \int_{L=0}^{\infty} Q(L) D_L \mathrm{d}L \qquad (4.5.9)$$

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.

### 4.5.3.2 The concept of operational quantities

The basic concept of the operational quantities is described in the ICRU Reports 39 and 43 [85I1, 88I1]. They have been introduced linking the external irradiation to the effective dose and the equivalent dose of the skin and eye lens in order to control their limits. The present definitions are given in ICRU Report 51 [93I1]. The operational quantities for radiation protection are dose equivalent quantities defined either for *strongly penetrating* or for *weakly penetrating* radiation incident on the human body (sometimes also the expressions *penetrating* and *low penetrating* are used instead of *strongly* and *weakly penetrating* radiation).

The radiation is characterised as either *weakly*- or *strongly penetrating* depending on which dose (effective dose or skin equivalent dose) is closer to the corresponding limit. Weakly penetrating radiations are $\alpha$-particles, $\beta$-particles with energies below 2 MeV and photons with mean energies below about 12 keV. Photons above this energy, electrons above 2 MeV and all neutrons are strongly 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** | **Operational quantities for** | |
|---|---|---|
| | **area monitoring** | **individual monitoring** |
| **Strongly penetrating radiation** | ambient dose equivalent | personal dose equivalent |
| **Weakly penetrating radiation** | directional dose equivalent | personal dose equivalent |

### 4.5.3.3 Operational quantities for area monitoring

#### ICRU sphere phantom

For all types of radiation the operational quantities for area monitoring are defined on the basis of a dose equivalent value at a point in a simple phantom, the ICRU sphere. It is a sphere of tissue-equivalent material (30 cm in diameter, 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.

**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. 4.1).

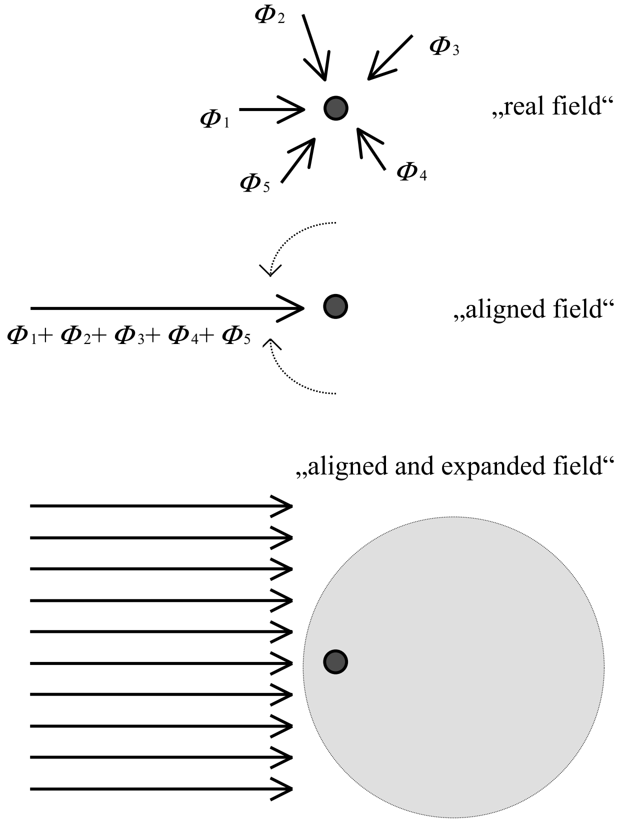


**Fig. 4.1.** Aligned and expanded field concept.

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 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 $\boldsymbol{\Omega}$ 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 the point of interest 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.

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

For area monitoring of strongly penetrating radiation the operational quantity is the ambient dose equivalent $H^*(10)$ defined by:

> 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 strongly penetrating radiation it is $d$ = 10 mm and $H^*(d)$ is written $H^*(10)$.

While this definition with the parameter $d$ is given in ICRU [93I1] and ICRP [91I1] the most recent ICRU Report [01I1] **dealing also with operational quantities** defines *ambient dose equivalent* by $H^*(10)$, thus restricting its definition to strongly penetrating radiation only. In practice, however, this has already been realised because other values have never been used.

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.

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

For area monitoring of weakly penetrating radiation the operational quantity is the directional dose equivalent $H'(0.07,\boldsymbol{\Omega})$ or, in rare cases, $H'(3,\boldsymbol{\Omega})$ defined by.

> The *directional dose equivalent* $H'(d,\boldsymbol{\Omega})$ 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 $\boldsymbol{\Omega}$.
>
> For weakly penetrating radiation it is $d$ = 0.07 mm and $H'(d,\boldsymbol{\Omega})$ is written $H'(0.07,\boldsymbol{\Omega})$.
> In case of monitoring the dose to the eye lens $H'(3,\boldsymbol{\Omega})$ with $d$ = 3 mm may be chosen.

In practice $H'(0.07,\boldsymbol{\Omega})$ is almost exclusively used in area monitoring for weakly penetrating radiation. For unidirectional radiation incidence the quantity may be written $H'(0.07,\alpha)$, where $\alpha$ is the angle between the direction $\boldsymbol{\Omega}$ and the direction opposite to radiation incidence.

The value of the directional dose equivalent can strongly depend on the direction $\boldsymbol{\Omega}$. The same is true for instruments for measuring weakly 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 $H'(0.07,\boldsymbol{\Omega})$ 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.

### 4.5.3.4 Operational quantities for individual monitoring

Individual monitoring is usually performed with individual dosemeters worn on the body and the operational quantity defined for this application takes into account this situation. The true value of the operational quantity is determined by the irradiation situation near the point where the dosemeter is worn. 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 strongly penetrating radiation a depth $d$ = 10 mm is recommended.
>
> For weakly 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 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 measured with a dosemeter worn on the body. The 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. 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, however, requires that the personal dosemeter must be worn at a position on the body which is representative with respect to the exposure. For the usual dosemeter position in front of the trunk the quantity $H_p(10)$ mostly furnishes a conservative estimate of $E$ even in cases of lateral or isotropic radiation incidence on the body. In cases of exposure from the back, however, a dosemeter worn at the front side and correctly measuring $H_p(10)$, will not provide a conservative estimate of $E$.

A further requirement for an operational quantity is that it allows dosemeters to be calibrated under reference conditions in terms of that quantity. The personal dose equivalent is defined in the individual human body and obviously individual dosemeters cannot be calibrated in front of a real human body. For calibration, 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 be defined not only in the human body but also in three phantoms of ICRU tissue (see Fig. 4.2) – a slab phantom (30 cm × 30 cm × 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, $H_{p,slab}(10)$ and $H_{p,slab}(0.07)$ etc., are defined as the true values of the corresponding $H_p$-quantities (see also Sect. 6.1.2 and Sect. 10.2.1).

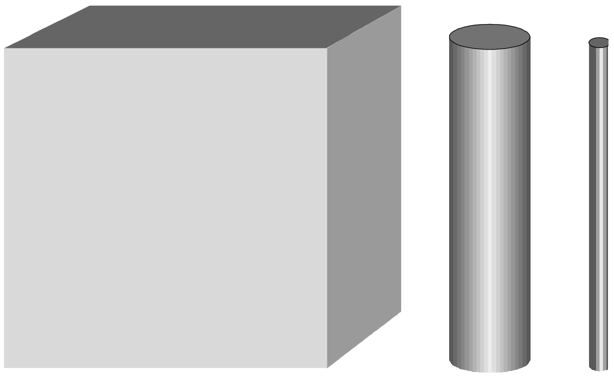

**Fig. 4.2.** Phantoms of ICRU tissue for the definition of $H_p$-quantities for calibration of individual dosemeters.
a) slab phantom
b) wrist phantom
c) finger phantom

## 4.6 Radioactivity quantities

The decay of a radionuclide is a stochastic process which means that the number of decays within a fixed time interval is described by a probability distribution. The expectation value of the number of decays is determined by the decay constant which is specific for each radionuclide and energy state (mostly the decay constant for the ground state is given).

The *decay constant* $\lambda$ of a radionuclide in a particular energy state is the quotient of d$P$ by d$t$, where d$P$ is the probability that a given nucleus undergoes a spontaneous transition from that energy state in the time interval d$t$. It is

$$\lambda = \frac{\mathrm{d}P}{\mathrm{d}t} \qquad \text{unit: s}^{-1}$$

The *half-life* $T_{1/2}$ of a radionuclide in a particular energy state is the mean time of the radionuclide in that state to decrease to one half of their initial number of nuclei. It is $T_{1/2} = (\ln 2)/\lambda$.

## 4.6.1 Activity, specific activity, activity concentration, activity per area

The activity $A$ of an amount of a radionuclide in a particular energy state at a given time is the quotient of d$N$ by d$t$, where d$N$ is the expectation value of the number of spontaneous nuclear transitions from that energy state in the time interval d$t$. It is

$$A = \frac{\mathrm{d}N}{\mathrm{d}t} \qquad \text{unit: becquerel (Bq), 1 Bq = 1 s}^{-1}$$

Radionuclides are mostly included in other solid, liquid or gaseous material and the amount is quantified by the quantities specific activity and activity concentration.

The *specific activity* $a_s$ is given by the quotient of the activity $A$ by the mass $m$, where $A$ is the activity of the radionuclide in the mass $m$.

$$a_s = \frac{A}{m} \qquad \text{unit: Bq kg}^{-1}$$

The *activity concentration* $c_{\text{nuclide}}$ is given by the quotient of the activity $A$ by the volume $V$, where $A$ is the activity of the radionuclide in the volume $V$.

$$c_{\text{nuclide}} = \frac{A}{V}, \qquad \text{unit: Bq m}^{-3}$$

For the determination of contaminations the distribution of radionuclides on surfaces is of interest. The related quantity is the *activity per unit area* $a_a$ defined by the quotient of the activity $A$ by the area $F$, where $A$ is the activity of a radionuclide distributed on the surface area $F$.

$$a_a = \frac{A}{F} \qquad \text{unit: Bq m}^{-2}\text{, often Bq cm}^{-2}$$

For decontamination of a surface from deposited radionuclides it is usually important if the radionuclides are removable or if they are diffused into the surface region of the material and are fixed near the surface in the material. If an $a_a$-value is given it should be specified if this value is related to the removable part only or to the total activity at the surface.

## 4.6.2 Specific quantities for radon, thoron and their progeny

Radon ($^{222}$Rn) and thoron ($^{220}$Rn) are gaseous radionuclides in the U- and Th-decay chain, respectively, occurring naturally (see 3.4.3). Their decay products are also radionuclides but metallic. While for radon the short-lived progeny $^{218}$Po, $^{214}$Pb, $^{214}$Bi and $^{214}$Po (see Table 4.4) are important in radiation protection, the important thoron progeny are $^{216}$Po, $^{212}$Pb, $^{212}$Bi and $^{212}$Po (see Table 4.5). In air there is usually a mixture of radon/thoron and short-lived radon/thoron progeny. These progeny are mostly attached to aerosols. A few percentages of them, however, are non-attached. The progeny may be deposited in the lung where its decay by alpha-particle emission is seen to be most important for lung cancer induction. Specific quantities have been defined taking care of this situation.

**Table 4.4.** Data of radon ($^{222}$Ra) progeny (nuclear data are from [NN98])

| Radionuclide | i | half-life $T_{1/2}$ | number of atoms per Bq $A/\lambda$ | potential alpha energy per atom $\varepsilon$ [MeV] | [$10^{-12}$ J] | per Bq $\varepsilon/\lambda$ [MeV] | [$10^{-12}$ J] | $k$[2] |
|---|---|---|---|---|---|---|---|---|
| $^{218}$Po | 1 | 3.10 min | 268 | 13.69 | 2.19 | 3 670 | 589 | 0.106 |
| $^{214}$Pb | 2 | 26.8 min | 2 320 | 7.69 | 1.23 | 17 800 | 2 860 | 0.513 |
| $^{214}$Bi | 3 | 19.9 min | 1 723 | 7.69 | 1.23 | 13 100 | 2 100 | 0.381 |
| $^{214}$Po | 4 | 164 µs | [1] | 7.69 | 1.23 | $2\times10^{-3}$ | $2.9\times10^{-4}$ | $6\times10^{-8}$ |

[1] no number is given because all atoms decay in less than 1 s and a calculated number would be much less than 1.
[2] factor $k$ is defined in Eq. (4.6.1)

**Table 4.5.** Data of thoron ($^{220}$Ra) progeny (nuclear data are from [NN98])

| Radionuclide | i | half-life $T_{1/2}$ | number of atoms per Bq $A/\lambda$ | potential alpha energy per atom $\varepsilon$ [MeV] | [$10^{-12}$ J] | per Bq $\varepsilon/\lambda$ [MeV] | [$10^{-12}$ J] | $k$[3] |
|---|---|---|---|---|---|---|---|---|
| $^{216}$Po | 1 | 0.15 s | [1] | 14.6 | 2.34 | 3.32 | 0.51 | $6\times10^{-6}$ |
| $^{212}$Pb | 2 | 10.6 h | 55 056 | 7.8 | 1.25 | 429 000 | 68 710 | 0.913 |
| $^{212}$Bi | 3 | 60.6 min | 5 246 | 7.8[2] | 1.25 | 40 900 | 6 554 | 0.087 |
| $^{212}$Po | 4 | 304 ns | [1] | 8.8 | 1.25 | $3.8\times10^{-6}$ | $6\times10^{-7}$ | $8\times10^{-12}$ |

[1] no number is given because all atoms decay in less than 1 s and a calculated number would be much less than 1.
[2] mean value from decay of $^{212}$Bi and $^{212}$Po by α-particle emission.
[3] factor $k$ is defined in Eq. (4.6.1)

### Potential alpha energy

The potential alpha energy $\varepsilon_i$ of an atom i in the decay chain of radon or thoron is the total energy of alpha-particles emitted during the decay of this atom to the long-living $^{210}$Pb or stable $^{208}$Pb, respectively.

The potential alpha energy of $N$ atoms of type i is $N\cdot\varepsilon_i$. The number of atoms $N$ per Bq is equal to $A/\lambda$, where $A$ is the activity of this radionuclide and $\lambda$ its decay constant. The potential alpha energy per Bq is then given by $\varepsilon/\lambda$ (unit: J Bq$^{-1}$, often used MeV Bq$^{-1}$).

### Concentration in air

The *potential alpha energy concentration* $c_{p,i}$ of a short-lived radon (or thoron) progeny in air is the sum of the potential alpha energy $\varepsilon_i$ of all atoms of this progeny present in a volume $V$ divided by this volume. It is

$$c_{p,i} = \frac{N_i \varepsilon_i}{V} = c_i \frac{\varepsilon_i}{\lambda_i} \qquad \text{unit: J m}^{-3} \text{ often MeV m}^{-3}$$

where $N_i$ is the number of atoms of this progeny in the volume $V$, $c_i$ the corresponding activity concentration and $\lambda_i$ the decay constant. The units are related by 1 J m$^{-3}$ = $6.242 \times 10^{12}$ MeV m$^{-3}$.

The *potential alpha energy concentration* (PAEC) $c_p$ of any mixture of short-lived radon (or thoron) progeny in air is the sum of the potential alpha energy concentrations of all progeny in the volume considered.

$$c_\mathrm{p} = \sum_\mathrm{i} c_\mathrm{p,i} = \sum_\mathrm{i} \varepsilon_\mathrm{i} \cdot c_\mathrm{i} / \lambda_\mathrm{i} \qquad \text{unit: J m}^{-3} \text{ often MeV m}^{-3}$$

Historically, for the potential alpha energy concentration the unit working level (WL) has widely been used. While originally defined as the potential alpha energy concentration associated with the radon progeny in equilibrium with 100 pCi $\mathrm{l}^{-1}$, 1 WL is now accurately fixed equal to $1.300 \times 10^8$ MeV $\mathrm{m}^{-3}$ which equals $2.08 \times 10^{-5}$ J $\mathrm{m}^{-3}$.

**Equilibrium equivalent concentration, equilibrium factor**

In case of radioactive equilibrium the activity concentration of radon $c_\mathrm{Rn}$ and of its progeny are equal. This, however, is usually not the case in air. For a non-equilibrium mixture a quantity equilibrium equivalent concentration $c_\mathrm{e}$ has been defined.

The *equilibrium equivalent concentration* (EEC) $c_\mathrm{e}$ corresponding to a non-equilibrium mixture of progeny in air is the fictitious activity concentration of radon in radioactive equilibrium with its short-lived progeny that has the same potential alpha energy concentration $c_\mathrm{p}$ as the actual non-equilibrium mixture. It is always $c_\mathrm{e} \le c_\mathrm{Rn}$. The SI-unit for both quantities, $c_\mathrm{e}$ and $c_\mathrm{Rn}$, is Bq $\mathrm{m}^{-3}$. In order to avoid confusion, the values of $c_\mathrm{e}$ are often marked Bq $\mathrm{m}^{-3}$ (EEC).

The equilibrium equivalent concentration $c_\mathrm{e}$ can be calculated from the activity concentrations of the progeny by the equation

$$c_\mathrm{e} = \sum_\mathrm{i} k_\mathrm{i} \cdot c_\mathrm{i} \qquad \text{with } k_\mathrm{i} = (\varepsilon_\mathrm{i}/\lambda_\mathrm{i}) \Big/ \sum (\varepsilon_\mathrm{i}/\lambda_\mathrm{i}) \tag{4.6.1}$$

The factors $k_\mathrm{i}$ are given in Tables 4 and 5 and it is

for radon progeny: $c_\mathrm{e} = 0.106\, c_\text{Po-218} + 0.513\, c_\text{Pb-214} + 0.381\, c_\text{Bi-214} + 6 \times 10^{-8}\, c_\text{Po-214}$ (4.6.2)

and for thoron progenies: $c_\mathrm{e} = 7 \times 10^{-6}\, c_\text{Po-216} + 0.913\, c_\text{Pb-212} + 0.087\, c_\text{Bi-212} + 8 \times 10^{-12}\, c_\text{Po-212}$ (4.6.3)

Obviously, the radionuclides $^{216}$Po, $^{214}$Po and $^{212}$Po can be ignored when calculating $c_\mathrm{e}$ because of their very low $k_\mathrm{i}$–values.

The *equilibrium factor F* is defined as the quotient of the equilibrium equivalent concentration and the activity concentration of the parent nuclide, radon, in air.

$$F = c_\mathrm{e} / c_\mathrm{Rn} \tag{4.6.4}$$

The value of $F$ ranges from 0 to 1 and is a measure to what extent radioactive equilibrium between radon and its progeny is obtained. Mostly this is not the case and often a mean value of 0.4 is convenient for the situation in homes.

The unattached progeny in air which are not attached to aerosols is also of special interest. The unattached fraction $f_\mathrm{p}$ is defined by the relative fraction of the total potential alpha energy concentration which stems from progeny in air which are not attached to aerosols. It is

$$f_\mathrm{p} = \frac{c_\mathrm{p}^\mathrm{f}}{c_\mathrm{p}} = \frac{c_\mathrm{p}^\mathrm{f}}{c_\mathrm{p}^\mathrm{a} + c_\mathrm{p}^\mathrm{f}} \tag{4.6.5}$$

where $c_\mathrm{p}^\mathrm{a}$ is the potential alpha energy concentration of the progeny attached to aerosols, $c_\mathrm{p}^\mathrm{f}$ is that of the unattached fraction and $c_\mathrm{p}$ is the sum of both parts.

**Inhalation exposure of individuals**

The exposure of an individual to radon progeny $P_p$ is defined as the time integral of the potential alpha energy concentration $c_p$ in air to which the individual is exposed.

$$P_p(T) = \int_0^T c_p(t)\,dt \tag{4.6.6}$$

where $T$ is the period of the exposure. A similar integral is given if the equilibrium concentration $c_e(t)$ is taken for integration. It is then called the *equilibrium equivalent exposure* $P_e(T)$:

$$P_e(T) = \int_0^T c_e(t)\,dt \tag{4.6.7}$$

The potential alpha energy exposure $P_p$ is often expressed in terms of working level month (WLM), even if not recommended for further use. This quantity has been introduced especially for specifying occupational exposure and a fixed time period $T$ of 170 hours has therefore been chosen equal to a mean monthly working time. The relation to SI-units (see Table 4.1) is given by 1 WLM = $3.54 \times 10^{-3}$ J h m$^{-3}$ = $2.21 \times 10^{10}$ MeV h m$^{-3}$.

## 4.7 Quantities for internal dosimetry

Internal exposure means an exposure by ionising radiation emitted from radionuclides incorporated and distributed in the body. A direct measurement of doses in a human body is not possible. For internal exposure there are, therefore, no specific operational dose quantities defined. In contrast to external monitoring usually the committed tissue or organ equivalent doses and committed effective dose are determined (period of 50 y for workers and a period up to the 70$^{th}$ year of life for members of the public including children) and complex compartment models are used to describe the long term biokinetic behaviour of the radionuclides in the human body (see Chapt. 7). The committed tissue and organ equivalent doses of an individual are usually determined by external measurements, e.g. activity concentration of specific radionuclides in the air, specific activity of food and water or contamination of the skin, and the application of calculated dose conversion coefficients (often called *dose coefficients*) which have been published for inhalation, ingestion and intake through the skin for a large number of radionuclides (see Chapter 7). Measured excretion data are usually used to estimate the intake of radionuclides subsequently and then conversion coefficients are applied to evaluate doses.

The intake of radionuclides by inhalation, ingestion or through intact or wounded skin or the excretion by exhalation, urine, faeces etc, is determined in terms of measurable quantities. These are often activity concentrations in air , $c_{nuclide}$ (in terms of Bq m$^{-3}$), and inhalation frequencies or inhaled activities, the specific activity of solids and liquids, $a_s$ (in terms of Bq kg$^{-1}$), the amount of ingested radioactive substances or their specific activity in excretions. Further details are given in Chapter 7.

In cases where radionuclides emit high energy γ-rays their distribution in the body may be determined by external measurements of γ-rays with a whole-body counter (large γ-detectors well shielded against radiation from the environment) in combination with computer codes simulating the photon absorption in the body.

## 4.8 Limits, constraints, action levels

The system of radiation protection as recommended by the ICRP [91I1] is based on the following principles (see also Chapter 1):

a) *No practice involving exposure to radiation should be adopted unless it produces sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes (principle of justification).*
b) *In relation to any particular source within a practice, the magnitude of individual doses, the number of people exposed, and the likelihood of incurring exposures where these are not certain to be received should all be kept as low as reasonably achievable, economic and social factors being taken into account. This procedure should be constrained by restrictions on the doses to individuals (dose constraints), or the risks to individuals in the case of potential exposures (risk constraints), so as to limit the inequity likely to result from the inherent economic and social judgements (the optimisation of protection).*
c) *The exposure of individuals resulting from the combination of all the relevant practices should be subject to dose limits, or to some control of risk in the case of potential exposures. These are aimed at ensuring that no individual is exposed to radiation risks that are judged to be unacceptable from these practices in any normal circumstances. Not all sources are susceptible of control by action at the source and it is necessary to specify the sources to be included as relevant before selecting a dose limit (individual dose and risk limits).*

Generally, radiation protection takes care of exposure situations and doses which are relevant for the health of the persons involved or may not be ignored compared to the normal exposure from natural radiation sources. This means that there should exist a dose level below which exposures from radionuclides or other radiation sources may not be taken care of and where no regulations are necessary, independent of the fact that, in principle, any radiation may induce cancer or genetic defects. The ICRP sees such a dose range below a few tens of μSv (committed dose or dose per year) for a single individual which is about 1/100 of the normal exposure from natural sources in the environment. Often an upper boundary of 10 μSv (committed dose or dose per year) is used to decide if further investigations or actions are necessary.

Usually the human exposures are classified in three different categories.

The first is called *occupational exposure* which means any exposure incurred at work and principally as a result of situations which can be reasonably regarded as being in the responsibility of the operating management. It also includes potential exposures where the probability of a future exposure due to planned work forces may be estimated [97I1].

*Medical exposures* describe the exposure of patients during diagnostic and treatment. While medical exposures are intended to provide a direct benefit to the patient, the practice should be justified and optimized with respect to applied doses and the medical benefit.

*Public exposures* are all exposures other than occupational or medical exposures. Public exposures include environmental exposures due to natural sources in the environment, e.g. natural actinides, radon, potassium-40 and cosmic radiation, but also those exposures due to artificial sources where the target group is the general population (details are given in Chapter 11). Examples are the broadly distributed radionuclides from the nuclear bomb test in 1950 to 1970 and the contamination due to the Chernobyl accident.

## Dose limits

Dose limits have been recommended by the ICRP [91I1] for occupational exposure and for exposures to the public. They are given in terms of effective dose and few organ equivalent doses, always summed over a given period. The limits apply to the sum of the relevant doses from external exposure in the specified period (often one year) and from intakes of radionuclides in the same period. The corresponding internal dose is the 50-year committed dose (for occupational exposure) or the committed dose up to the age of 70 years (for members of the public). For public exposures the scope of these limits are restricted to doses incurred as the result of practices. Doses incurred in situations where the only protective action takes the form of an intervention are not included in system of dose limits (see action levels). Radon, thoron and their progeny in open air or in houses, natural radionuclides already in the environment and cosmic radiation on ground, are examples of those situations.

While these dose limits are called primary limits, for practical reasons further limits (secondary or derived limits) are specified which are given in terms of operational or other quantities and derived from the primary limits. They are applied, for example, to define control or prohibited areas or annual limits of intake (ALI) of radionuclides. The ALI values which are specific for each radionuclide considered are also based on the committed dose for the same periods as mentioned above.

The primary dose limits internationally recommended by ICRP [91I1] and the IAEA [96IA] are given below. Many, but not all countries have transferred these values into their national legislation and regulations. Some countries have either less or more restrictive regulations. As a consequence, the legal dose limits may, therefore, be different in different countries.

For occupational exposure the effective dose is limited to 20 mSv per year averaged over 5 years (100 mSv in 5 years) with the further provision that the effective dose should not exceed 50 mSv within any single year. This limit avoids any deterministic effects of exposures and limits the stochastic effects to a risk level of about $10^{-3}$. For internal exposure the committed dose limit is restricted to 20 mSv in each year and the annual limit of intake (ALI) is related to this value.

For women, when pregnancy has been declared, the embryo and foetus should be protected by applying for external exposure an additional equivalent dose limit to the surface of its abdomen of 2 mSv for the remaining period of the pregnancy and limiting the intake of radionuclides to 1/20 of the annual limit of intake.

The detriment due to external weakly penetrating radiation mainly concerns the skin or the eye lens. In order to avoid deterministic effects the skin dose is, therefore, additionally limited to 500 mSv per year (averaged over any 1 $cm^2$, regardless of the area exposed) and the dose to the eye lens to 150 mSv per year. For the same reason, the annual equivalent dose to the extremities (hands, feet) is also limited to 500 mSv.

The approach for choosing dose limits to the public may be either based on the same ideas as for occupational exposure considering, however, the fact of the large number of persons involved or on the judgement on the existing dose level from natural radiation sources and its variation in different places where no influence on the health detriment of the population has been observed.

For public exposure from sources given in practices, the ICRP has recommended a limitation of the effective dose to 1 mSv per year. In special circumstances, however, a higher value may be allowed in a single year if the average over 5 years does not exceed 1 mSv per year. The ICRP has also defined additional annual limits for the skin and the eye lenses which are 1/10 of the value recommended for workers (50 mSv averaged over any 1 $cm^2$ of the skin and 15 mSv for the lens of the eye).

## Dose constraints

The control of public exposure in normal situations is usually performed by the application of controls at the different sources applying procedures of constrained optimisation and the use of prescriptive limits. A dose constraint, which is a value of individual dose from a defined source, should be used in the optimisation of protection to exclude protection options that would result in individual doses exceeding the constraint. Dose constraints are an integral part of the optimisation of protection and are thus prospectively. They are not, however, limits to be applied retrospectively.

The above mentioned annual limit for occupational exposure means implicitly that the dose constraint for optimisation should not exceed 20 mSv per year. It is often convenient to define a homogeneous group of persons – a critical group – which are assumed to be most highly exposed by the single source considered and to apply the dose constraint to the mean dose in that critical group.

For medical exposures no dose limits have been recommended because a radiotherapy or diagnostic treatment should always provide a direct benefit to the patient. The choice of the practice and its performance should be optimized with respect to applied doses and the medical benefit.

In order to characterize good medical practice and to enable quality assurance programs for use in these cases, it is helpful to define constraints or reference values based on the actual state of the art of the various investigations and procedures. Such values are especially given in the various practices of diagnostics with X-rays and radionuclides.

## Action levels

While for the situation of occupational exposures generally dose limits are defined, there are other situations where the only protective action takes the form of an intervention, e.g. in cases of public exposure in areas of high level of natural radiation or in areas contaminated because of former human activities or accidents like e.g. nuclear bomb testing or the Chernobyl accident.

For intervention situations action levels may be defined which specify dose levels or an activity concentration in air or the specific activity in materials of the environment which are of concern with respect to public exposure. If such a level is exceeded, this should initiate measures for a reduction of the exposure and different action levels may define different measures characterising the strength of the necessary intervention.

While for public exposure from sources given in practices, a dose limit of 1 mSv per year is recommended which is in the order of the natural background exposure excluding radon, action levels for initiating protection measures to the public are mostly higher, depending on the strength of the recommended measures.

A typical case for the definition of action levels is the exposure by radon and its progenies. Radon is always present in the environment and may appear in higher concentrations at specific work places or in homes. For radon, however, the specification of a dose level for actions is relatively complex because the dose coefficient relating a mean radon concentration at a place to an effective dose value depends on the mean equilibrium factor $F$ (see 4.6.2) and the mean annual time people stay at this place. Furthermore, the coefficient is mainly based on modelling and includes a large uncertainty (see Chapt. 7). Mostly the radon action levels are given in terms of Bq m$^{-3}$ or Bq m$^{-3}$ h.

For occupational exposure with the assumption of 2000 h at work and a mean equilibrium factor of 0.4 a conversion factor of 156 Bq m$^{-3}$ mSv$^{-1}$ or 62 Bq m$^{-3}$(EEC)mSv$^{-1}$, respectively, is given [94I1]. Actions for reducing the radon concentration at a work place may be performed if the mean annual radon concentration is in the range from 500 Bq m$^{-3}$ to 1500 Bq m$^{-3}$ (1000 Bq m$^{-3}$ corresponds to about 6 mSv per year for 2000 h at work and $F$ = 0.4). A mean annual concentration of 3000 Bq m$^{-3}$ corresponds to about 20 mSv per year which is equal to the dose limit for occupational exposure.

For public exposure in dwellings a conversion factor of 58 Bq m$^{-3}$ mSv$^{-1}$ or 23 Bq m$^{-3}$(EEC)mSv$^{-1}$, respectively, is given [94I1] under the assumption that a person stays 7000 h per year in the house (with $F$ = 0.4 in the house) and the other time in free air (with a low radon concentration). Action levels are recommended by the ICRP also for this case. They are based on the following ideas. There exists a range of normal mean radon concentrations in dwellings where no actions are necessary or useful. For existing houses with higher mean concentrations actions for reducing such values should be considered. Future houses should be designed to stay within the normal range. Because of the very different situation in the various regions regarding the natural radon concentration in the ground and hence in houses, local recommendations or regulations may differ strongly in the different countries. The ICRP has specified an upper boundary of the normal mean radon concentration in dwellings with 200-400 Bq m$^{-3}$ depending on the regional situation.

Another situations which may occur are emergencies where people or the government needs advice what type of actions are necessary under given or expected exposures, e.g. staying at home, avoiding to eat fresh vegetables or drink fresh milk or leaving a defined area for some time. Such a situation is also a type of intervention where action levels may be defined in national regulations like those mentioned above. Principles and more detailed information are given in ICRP Publication 63 [93I4]. For immediate emergency situations there may also dose values be given for fireman and other rescue personnel in order to restrict their risks due to an exposure. Dose levels may be defined which should not be exceeded in one case or annually or in life time.

## 4.9 References


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