Chapter

Dosimetry for Use in Preparedness and Response to Radiological and Nuclear Emergency

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Abstract

Lessons learned from responses to past radiological and nuclear emergencies have shown that more guidance is needed for assessing doses to those who were affected in emergency exposure situation. The chapter introduces system of dosimetric quantities for use in emergency preparedness and response to nuclear or radiological emergency, which includes RBE-weighted absorbed dose in tissue or organ for evaluation of the risk of severe deterministic effects, equivalent dose in tissue or organ for evaluation of the risk of stochastic effects, and effective dose for evaluation of the detriment due to undetectable stochastic effects. The chapter also provides internationally proved criteria for protection of individual in emergency exposure situation and framework of dose and risk assessment in an emergency. The special attention has been put on evaluation of available sources of dosimetric data needed for dose and risk assessment in an emergency.

Keywords: emergency, dosimetry, internal exposure, external exposure, deterministic effects, stochastic effects

1. Introduction

In event of nuclear or radiological emergency, uncontrolled exposure to ionizing radiation can cause fatal or threatening health effects. Dose received by individual determines the nature of such effects and their severity. To protect individual in emergency exposure situation, one needs to assess doses, which have been received, or would be received in emergency exposure situation, to be able to take an informed decision on protective and other response actions, including medical treatment of overexposed person as demanded in the IAEA General Safety Requirements (GSR) Part 7 [1]. The chapter presents a basis for dose assessment in emergency exposure situation and includes three parts:

1. Explanation of dosimetric quantities, such as basic, protection, and operational quantities, for use in emergency preparedness and response.

2. Criteria for assessment of doses received or expected to be received in emergency exposure situation.

3. Framework of estimation of protection quantities from monitoring results in event of nuclear or radiological emergency.

2. Dosimetric quantities

The International Commission on Radiation Units and Measurements (ICRU) in Report 30 [2] formulates a quantitative dosimetric concept in radiobiology as follows:

1. Biological effects of radiation are correlated with the energy absorbed by ionization and excitation in unit mass of tissue.

2. Biological effects of radiation are modified by microscopic spatial distribution of energy of radiation imparted to matter.

Dosimetric concept considers two-step assessment of a dose to characterize health consequences of exposure to ionizing radiation. On the first step, an energy of radiation absorbed in a tissue or organ has to be evaluated. On the next step, a human exposure has to be evaluated in term of protection quantities which could be used in models of developing radiation health effects to assess health consequences (risks) associated with irradiation. At this step, quality of radiation has to be considered taking into account its dependence on properties of radiation, properties of tissue or organ, and expected health effect.

2.1 Basic physical quantities

The basic dosimetric quantities include the particle fluence Φ_R, the kerma K_R, and the absorbed dose D_R of radiation R. Basic quantities characterize the field of radiation and its interaction with a medium where the human body could be present:

1. The fluence, Φ_R, is the quotient of dN_R by ds, where dN_R is the number of particles R incident on a sphere of cross-sectional area ds. Unit of fluence is m^{-2}.

2. The kerma, K_R, for ionizing uncharged particles R, is the quotient of dE_{tr} by dm, where dE_{tr} is the mean sum of the initial kinetic energies of all the charged particles liberated in a mass dm of a material by the uncharged particles incident on dm. Unit of kerma is J/kg. The special name for the unit of kerma is gray (Gy).

3. The absorbed dose, D_R, is the quotient of $d\bar{\varepsilon}$ by dm, where $d\bar{\varepsilon}$ is the mean energy imparted by ionizing radiation to matter of mass dm. Unit of absorbed dose is J/kg. The special name for the unit of absorbed dose is gray (Gy).

The ICRU provides exact definition of these quantities in [3].

2.2 Protection quantities

Protection quantities are the dosimetric quantities, which characterize the irradiation of the human. GSR Part 7 [1] states that in response to emergency, "consideration shall be given to actions to be taken to avoid or to minimize severe deterministic effects and to reduce the risk of stochastic effects. Deterministic effects shall be evaluated on the basis of relative biological effectiveness (RBE) weighted absorbed dose in tissue or organ. Stochastic effects in a tissue or organ

shall be evaluated on the basis of equivalent dose in tissue or organ. The detriment associated with the occurrence of stochastic effects in individuals in an exposed population shall be evaluated on the basis of the effective dose" (para. 4.28 of GSR Part 7 [1]). Dosimetric quantities of RBE-weighted absorbed dose in tissue or organ T, AD_T; equivalent dose in tissue or organ T, H_T; and effective dose E are defined in GSG-2 [4] and GSR Part 3 [5] and recommended for protection purposes.

The overall theory on the development of biological health effects is not yet developed. The protection quantities are essentially used for evaluation of consequences of human exposure to ionizing radiation in terms of developing deterministic and stochastic health effects separately as listed in **Table 1**. **Figure 1** presents relationship between protection quantities and basic physical quantities.

Radiation protection quantities could not be measured directly. In particular irradiation conditions, they could be calculated by taking into account estimates of basic protection quantities (results of their measurement or calculation), geometry of irradiation, characteristics of irradiated person, etc. For radiation protection purposes, these quantities are defined for reference persons representing different groups of the public and workers in [6] and standard irradiation geometries for external exposure in [7].

Definitions of these quantities can be found in the GSR Part 3 [5], 2007 Recommendations of the International Commission on Radiological Protection (ICRP) [8], and EPR Publication [9].

The determination of RBE-weighted dose AD_T in a tissue or organ T involves the use of tissue-specific and radiation-specific factors $RBE_{T,R}$ as a multiplier of absorbed dose for radiation R, to reflect the relative biological effectiveness of the radiation in inducing severe deterministic effect in organ T at high doses:

$$AD_T = \sum_R D_{T,R} \times RBE_{T,R}, \tag{1}$$

where $D_{T,R}$ is the average absorbed dose in the tissue or organ T for radiation R.

Factor $RBE_{T,R}$ depends on the quality of radiation and macroscopic distribution of energy of radiation imparted to matter of affected organ or tissue [9]. The role of such heterogeneity is significant in the case of internal exposure [9, 10]. The RBE-weighted absorbed dose in tissue or organ is defined for use together with

Dosimetric quantity	Symbol	Unit	Purpose
Protection quantities			
RBE-weighted absorbed dose in tissue or organ T	AD_T	Gy	For evaluating deterministic effects induced as a result of exposure of an organ or tissue T
Equivalent dose in tissue or organ T	H_T	Sv	For evaluating stochastic effects induced as a result of exposure of an organ or tissue T
Effective dose	E	Sv	For evaluating detriment related to the occurrence of stochastic effects in an exposed population
Operational quantities			
Personal dose equivalent	$H_P(d)$	Sv	For monitoring external exposure of an individual
Ambient dose equivalent	$H^*(d)$	Sv	For monitoring a radiation field of the strongly penetrating radiation
Directional dose equivalent	$H'(d,\Omega)$	Sv	For monitoring a radiation field of the weakly penetrating radiation

Table 1.
Dosimetric quantities.

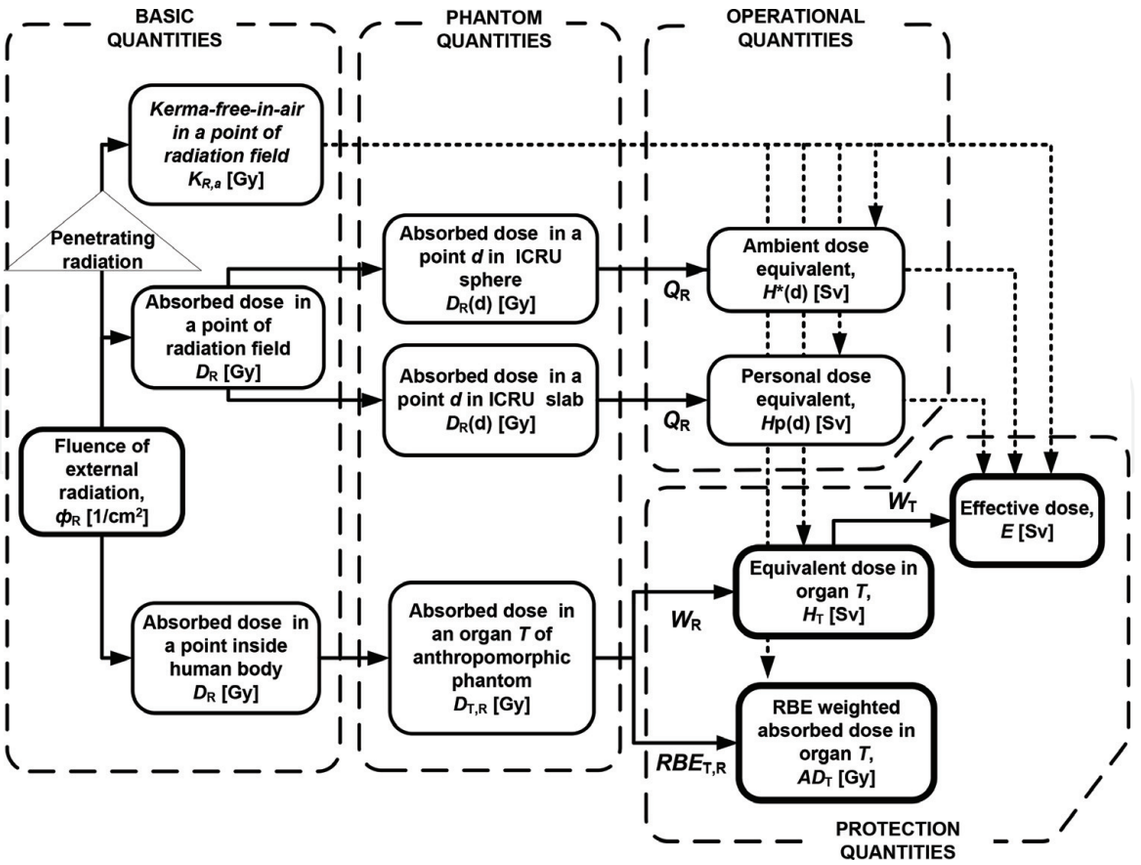

Figure 1.
Dosimetric quantities.

radiobiological model for assessing the risk of developing severe deterministic effects in critical organs and tissues at high doses [9–12].

The determination of equivalent dose H_T in a tissue or organ T involves the use of a radiation weighting factor w_R as a multiplier of absorbed dose for radiation R, to reflect the relative biological effectiveness of the radiation in inducing stochastic effects at low doses:

$$H_T = \sum_R D_{T,R} \times w_R. \tag{2}$$

The equivalent dose in tissue or organ is defined for use together with radiobiological model for assessing the risk of developing stochastic effects in organs and tissues at low doses of external [13] and internal [14] exposure.

The determination of effective dose E involves the use of a tissue weighting factor w_T as a multiplier of equivalent dose for tissue T, to account for the different sensitivities of different tissues or organs to the induction of stochastic effects and production of radiation detriment [8, 15]:

$$E = \sum_T H_T \times w_T = \sum_{T,R} D_{T,R} \times w_R \times w_T. \tag{3}$$

In accordance with the ICRP, the radiation detriment is the total harm to health experienced by an exposed group and its descendants as a result of the group's exposure to a radiation source. Detriment is a multidimensional concept. Its principal components are the stochastic quantities: probability of attributable fatal cancer, weighted probability of attributable nonfatal cancer, weighted probability of severe heritable effects, and length of life lost if the harm occurs [8]. The 2007

Recommendations of the ICRP state in para. (153) that the main and primary uses of effective dose in radiological protection for both occupational workers and the general public are:

- prospective dose assessment for planning and optimization of protection; and

- retrospective dose assessment for demonstrating compliance with dose limits, or for comparing with dose constraints or reference levels.

In this regard, organ or tissue equivalent doses, not effective doses, are required for assessing the probability of cancer induction in exposed individuals.

The recommended values of $RBE_{T,R}$, w_R and w_T are based on a review of published biological and epidemiological studies and are given in the definitions of protection quantities in GSR Part 3 [5].

2.3 Operational quantities

Since radiation protection quantities cannot be measured directly, the ICRU introduced operational quantities for practical use in radiation protection where exposure due to external sources is concerned. Definitions of these quantities can be found in [16–20] and in GSR Part 3 [5]. The operational quantities provide an estimate of effective or equivalent dose in tissue or organ in such a way that avoids overestimation in most radiation fields encountered in practice [7]. Radiation quality factor Q(L) is used in calculating the operational dose equivalent quantities used in monitoring [21]. The quality factor characterizes the biological effectiveness of the radiation type, based on the ionization density along the tracks of charged particles in tissue. Q is defined as a function of the unrestricted linear energy transfer, L_m (often denoted as L or linear energy transfer, LET), of charged particles in water. Detailed evaluation of the relationship between the physical, protection, and operational quantities was conducted by a joint task group of the ICRP and ICRU [7].

Strongly penetrating radiation and weakly penetrating radiation are considered in radiation dosimetry and are differentiated as follows. For most practical purposes, it may be assumed that strongly penetrating radiation includes photons of energy above about 12 keV, electrons of energy more than about 2 MeV, and neutrons. It may be also assumed that weakly penetrating radiation includes photons of energy below about 12 keV, electrons of energy less than about 2 MeV, and massive charged particles as protons and alpha particles [5].

The operational quantity for individual monitoring is the personal dose equivalent $H_P(d)$. Any statement of personal dose equivalent has to include a specification of the reference depth d. In order to simplify the notation, d is assumed to be expressed in millimeters.

For strongly penetrating radiation, the reference depth for controlling the radiation detriment in planned exposure situation is 10 mm. Personal dose equivalent $H_P(10)$ provides a conservative estimate of effective dose to adult for strongly penetrating radiation.

For weakly penetrating radiation, the reference depth for controlling stochastic effects due to irradiation of the basal membrane of the skin is 0.07 mm, and the deterministic effect in lens of the eye is 3 mm [15, 22, 23].

For monitoring of the lens of the eye, a depth of 3 mm is recommended by the International Commission on Radiation Units and Measurements [3], so the operational quantity to be used is $H_P(3)$. In practice, however, the use of $H_P(3)$ has not yet been implemented for routine individual monitoring. In specific cases, when

actual workplace radiation fields are known, monitoring of the eye through dosimeters calibrated for $H_P(0.07)$ could be acceptable. In [23], it is stated that $H_P(0.07)$ can be considered a good operational quantity for the lens of the eye for exposures to fields for which most of the dose is due to photons, including X radiation. In such cases, it has to be borne in mind that the uncertainty associated with the estimation of equivalent dose will be higher.

The operational quantities recommended for workplace monitoring are defined in a phantom known as the ICRU sphere [18]. This is a sphere of 30 cm diameter made of tissue equivalent material with a density of 1 g/cm^3 and an elemental composition (by mass) of 76.2% oxygen, 11.1% carbon, 10.1% hydrogen, and 2.6% nitrogen.

The two quantities recommended by the International Commission on Radiation Units and Measurements for workplace monitoring [3] are the ambient dose equivalent $H^*(d)$ and the directional dose equivalent $H'(d,\Omega)$.

The ambient dose equivalent $H^*(d)$ at a point in a radiation field is the dose equivalent that would be produced by the corresponding aligned and expanded field in the ICRU sphere, at a depth d on the radius opposing the direction of the aligned field.

The expanded field (see **Figure 2(b)**) is one in which the fluence and its angular and energy distribution are the same throughout the volume of interest as in the actual field at the point of reference (see **Figure 2(a)**). In the expanded and aligned field (see **Figure 2(c)**), the fluence and its energy distribution are the same as in the expanded field, but the fluence is unidirectional [12].

Any statement of ambient dose equivalent has to include a specification of the reference depth d. For strongly penetrating radiation, the recommended depth is 10 mm. When measuring $H^*(10)$, the radiation fields have to be uniformed over the sensitive volume of the instrument, and the instrument has to have an isotropic response.

The directional dose equivalent $H'(d,\Omega)$ at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded field in the ICRU sphere, at a depth d on a radius in a specified direction Ω. Any statement of directional dose equivalent has to include a specification of the reference depth d and the direction Ω of the radiation. For strongly penetrating radiation and weakly penetrating radiation, the recommended depths are 10 mm and 0.07 mm, respectively.

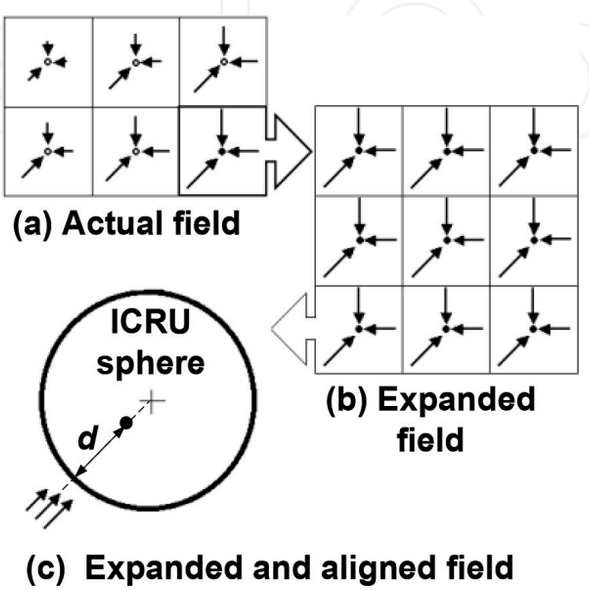

Figure 2.
Actual, expanded, and aligned radiation field.

If the field is unidirectional, the direction Ω is specified as the angle between the radius opposing the incident field and the specified radius. When the specified radius is parallel to the radiation field (i.e., when Ω = 0°) the quantity $H'(d, 0)$ is equal to $H^*(d)$. When measuring $H'(d, \Omega)$, the radiation fields have to be uniformed over the dimensions of the instrument, and the instrument has to have the appropriate directional response.

3. Generic dosimetric criteria

When determining the possible health hazard, three dosimetric quantities must be considered as shown in **Table 1**:

1. The RBE-weighted dose AD_T, which is used to evaluate the risk of severe deterministic effects

2. The equivalent dose H_T, which is used to evaluate the risk of stochastic effects

3. The effective dose E, which is used to evaluate the radiogenic detriment for purposes of radiation protection

Protection quantities are applicable in certain dose ranges. **Figure 3** provides an example of the applicable ranges for dosimetric quantities characterizing external penetrating radiation to evaluate risk of severe deterministic effects and stochastic effects, i.e., observable increase in the incidence of radiation-induced cancers. The ranges are not exactly defined because of competition of effects. For instance, both the risks of radiogenic cancers and severe deterministic effects have to be evaluated when whole-body absorbed dose is around 1 Gy. Below 100 mGy may not have any severe deterministic effects or an observable increase in the incidence of cancer, even in a very large exposed group. An increase in the cancer incidence rate due to radiation-induced cases is uncertain and will not be detectable [10, 12, 24].

The system of protective actions and other response actions in an emergency includes numerical values of generic criteria as well as of the corresponding operational criteria that form the basis for decision-making in an emergency.

Table 2 presents levels of RBE-weighted absorbed dose in critical organs and tissues which if exceeded will give rise to severe deterministic effects in 5% of those who are exposed.

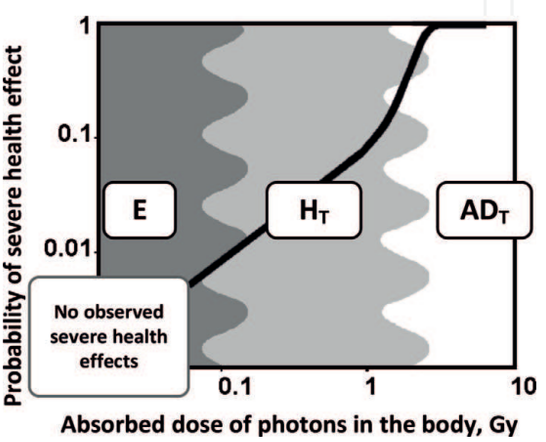

Figure 3.
Probability of severe health effects as a function of absorbed dose for external penetrating radiation.

Quantity	Level	Comments
Acute external exposure (for less than 10 h)		
$AD_{Red\ marrow}$	1 Gy	$AD_{Red\ marrow}$ represents the RBE-weighted absorbed dose in internal tissues or organs from exposure in a uniform field of strongly penetrating radiation [1, 4]
$AD_{Fetus/Embryo}$	0.1 Gy	No
$AD_{Soft\ tissue}$	25 Gy	Dose delivered to 100 cm^2 at a depth of 0.5 cm under the body surface in soft tissue [1, 4]
$AD_{Skn\ derma}$	10 Gy	Dose delivered to the 100 cm^2 dermis (skin structures at a depth of 40 mg/cm^2 (or 0.4 mm) below the body surface) [1, 4]
Acute intake of radioactive substance		
$AD_{Red\ marrow}(\Delta)$	0.2 Gy	For intake of radionuclides with $Z > 90$; Δ = 30 d [1, 4]
	2 Gy	For intake of radionuclides with $Z \leq 89$; Δ = 30 d [1, 4]
$AD_{Thyroid}(\Delta)$	2 Gy	For intake of thyroid-seeking radionuclides; Δ = 30 d [1, 4]
$AD_{Lung}(\Delta)$	30 Gy	For the purposes of these generic criteria, "lung" means the alveolar-interstitial region of the respiratory tract; Δ = 30 d [1, 4]
$AD_{Colon}(\Delta)$	20 Gy	For the purposes of these generic criteria, "colon" presents upper and lower large intestine: Δ = 30 d [1, 4]
$AD_{Fetus/Embryo}\left(\widetilde{\Delta}\right)$	0.1 Gy	For this particular case, $\widetilde{\Delta}$ means the period of in utero development [1, 4]

Table 2.
Criteria for assessing the high doses from external and internal exposure.

Quantity	Level	Comments
E	100 mSv in 1 year	No
$H_{Fetus/Embryo}$	100 mSv in 1 year	No
$H_{Thyroid}$	100 mSv in 1 year	For intake of thyroid-seeking radionuclides

Table 3.
Criteria for assessing the intermediate doses from external and internal exposure.

Table 3 provides the dosimetric criteria used to define the radiological health hazard level used for taking a decision on implementation of protective and other response actions in emergency exposure situation. These criteria are based on **Table 3** of [4], which provides the international generic criteria below which the risk of cancers and other health effects is too low to justify taking any protective or other response actions, such as a medical screening. The criteria were established for exposures at high-dose rates. For the lower-dose rates that will occur off the site following a release resulting from a severe emergency at a light-water reactor or its spent fuel pool, a comparable level of radiation-induced cancer risk would probably occur at a dose two or more times higher [8].

For restricting the exposure of emergency workers having assigned tasks in an emergency response, **Table 4** provides guidance values in terms of personal dose equivalent $H_P(10)$ from external exposure to strongly penetrating radiation. The values for $H_P(10)$ in **Table 4** assume that every effort has been made for protection against external exposure to weakly penetrating radiation and against exposure due to intakes or skin contamination (see para. 5.53 of GSR Part 7 [1]). The values of

Task	Quantity and guidance level		
	$H_P(10)$	E	AD_T
Lifesaving actions [(a)] Actions to prevent severe deterministic effects and actions to prevent the development of catastrophic conditions that could significantly affect people and the environment	500 mSv	500 mSv	50% from **Table 2**
Actions to avert a large collective dose	100 mSv	100 mSv	10% from **Table 2**

*Note: [(a)]This value may be exceeded—with due consideration of the generic criteria in **Table 2**—under circumstances in which the expected benefits to others clearly outweigh the emergency worker's own health risks and the emergency worker volunteers to take the action and understands and accepts these health risks [1].*

Table 4.
Criteria for assessing individual doses in emergency workers.

$H_P(10)$ in **Table 4** are used for planning and operational monitoring of the work's exposure.

To assure protection of the emergency workers, the total effective dose and the RBE-weighted absorbed dose in a tissue or organ via all exposure pathways need to be estimated as early as possible in a nuclear or radiological emergency.

4. Estimation of protection quantities

Protection quantities of AD_T, H_T and E characterize the exposure of an individual for purposes of implementation of protective actions and other actions to protect him or her in emergency exposure situation. The practical goal of radiation monitoring in emergency exposure situation is to provide the information required for estimation of protective quantities.

4.1 Estimation of protection quantities characterizing an external exposure

Properties of external radiation entering the body of individual R, basic physical characteristics of radiation field, such as particle fluence Φ_R, the kerma K_R, and geometry of irradiation G, provide a basis for estimation of the protection quantities from external exposure. For idealized irradiation geometries, the ICRP in [7] presents the relationship between basic physical quantities and protection quantities for monoenergetic photons, electrons and positrons, neutrons, protons, muons, pions, and helium ions in six considered irradiation geometries: AP, anteroposterior; PA, posteroanterior; LLAT, left lateral; RLAT, right lateral; ROT, rotational; and ISO, isotropic.

As presented in **Figure 4**, the major characteristic of a field of radiation is a fluence of radiation Φ_R. For monoenergetic radiation, an ambient dose equivalent and air kerma are proportional to fluence:

$$\Phi_R = K_{a,R}/k_{a,R} = H_R^*(d)/h_R^*(d). \tag{4}$$

For non-monoenergetic radiation

$$K_a = \sum_R \int \Phi_R(\varepsilon_R) \times k_a(\varepsilon_R) d\varepsilon_R, \tag{5}$$

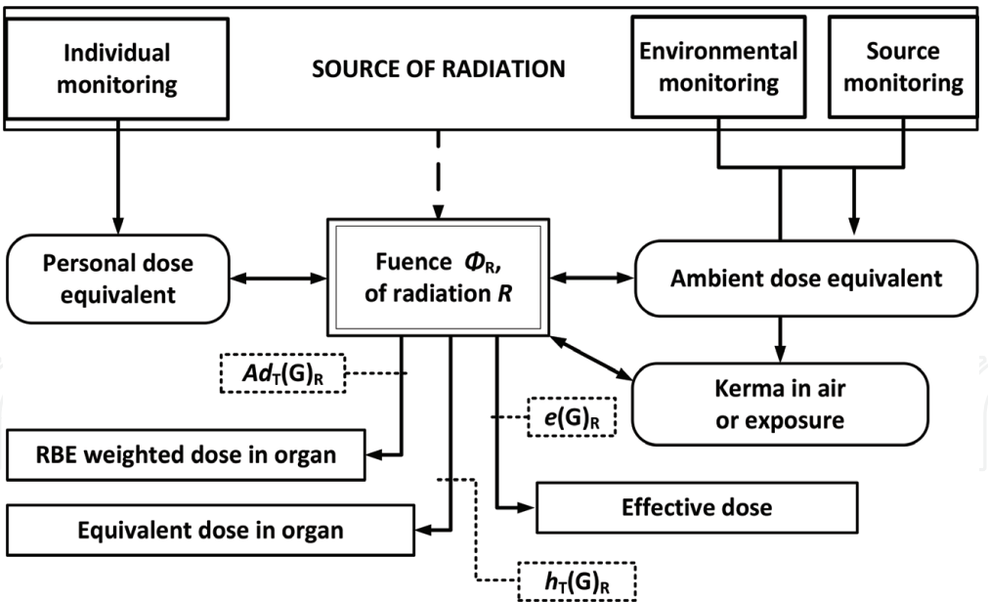

Figure 4.
General scheme for assessment of individual external dose from monitoring.

$$H^*(10) = \sum_R \int \Phi_R(\varepsilon_R) \times h^*(10, \varepsilon_R) d\varepsilon_R, \tag{6}$$

where $\Phi_R(\varepsilon_R)d\varepsilon_R$ is the fluence of radiation R with energy between ε_R and $\varepsilon_R + d\varepsilon_R$, $k_a(\varepsilon_R)$ is the dose coefficient of air kerma of radiation R with energy ε_R, and $h^*(10, \varepsilon_R)$ is the dose coefficient of ambient dose equivalent of radiation R with energy ε_R.

For irradiation geometry G, one has

$$H_P(10)_G = \sum_R \int \Phi_R(\varepsilon_R) \times h_P(10, \varepsilon_R)_G d\varepsilon_R, \tag{7}$$

$$AD_{T,G} = \sum_R \int \Phi_R(\varepsilon_R) \times Ad_T(10, \varepsilon_R)_G d\varepsilon_R, \tag{8}$$

$$H_{T,G} = \sum_R \int \Phi_R(\varepsilon_R) \times h_T(\varepsilon_R)_G d\varepsilon_R, \tag{9}$$

$$E_G = \sum_R \int \Phi_R(\varepsilon_R) \times e(\varepsilon_R)_G d\varepsilon_R, \tag{10}$$

where $k_a(\varepsilon_R)$ is the dose coefficient of air kerma of radiation R with energy ε_R, $h_P(10, \varepsilon_R)_G$ is the dose coefficient of personal dose equivalent of radiation R with energy ε_R and irradiation geometry G, $Ad_T(\varepsilon_R)_G$ is the dose coefficient of RBE-weighted dose in organ T of radiation R with energy ε_R and irradiation geometry G, $h_T(\varepsilon_R)_G$ is the dose coefficient of equivalent dose in organ T of radiation R with energy ε_R and irradiation geometry G, and $e(\varepsilon_R)_G$ is the dose coefficient of effective dose of radiation R with energy ε_R and irradiation geometry G.

For photon radiation, values of $Ad_T(\varepsilon_R)_G$ and $h_T(\varepsilon_R)_G$ are numerically equal when the same organ or tissue and irradiation geometry are considered.

The internet resource [25] provides tools for evaluation of absorbed dose in different organs from point or volumetric radioactive sources inside or outside the human body.

The publication [26] provides dose coefficients for exposure to bulk sources that are ground, water body and cloud containing radioactive material.

The publication [27] presents a compendium of neutron spectra, which could be used for estimation of protection quantities in accordance with Eqs. (7)–(10). The estimates show that for fission neutrons scattered from the concrete walls of the facility, soil, or from the air surrounding the facility (skyshine), the value of $Ad_T(\varepsilon_R)_G$ is numerically equal to 1/5 of $h_T(\varepsilon_R)_G$ when the same organ or tissue and irradiation geometry are considered. This linkage provide the possibility to use results of operational or routine monitoring of doses in workers for estimation of $AD_{Red\ marrow}$ in emergency exposure situation as required by GSR Part 7 and presented in **Table 4**.

As presented in [7], dose coefficients mentioned above are proportional to the photon energy in range of (0.1–6) MeV. Therefore, for the same irradiation geometry

$$\frac{H_{T,G} \times \varepsilon_j}{h_T(\varepsilon_j)_G} \cong \frac{H_P(10)_G \times \varepsilon_j}{h_P(10,\varepsilon_j)_G} \cong \frac{H^*(10) \times \varepsilon_j}{h^*(10,\varepsilon_j)} \cong \frac{K_a \times \varepsilon_j}{k_a(\varepsilon_j)} \ldots, \tag{11}$$

where ε_j is any photon energy from range of (0.1–6) MeV.

For mentioned range of photon' energy, the exposure is proportional to the kerma free-in-air (air kerma) or exposure. Thus, in the same point of field of photon radiation, exposure or kerma in air for an exposure of 100 R is 0.876 Gy [28]. This linkage provides the possibility to use old devices such as exposure meters (R-meters) in environmental monitoring for estimation of air kerma and protection quantities as given in Eq. (11).

The protection quantities AD_T, H_T and E received from exposure due to external sources can be also estimated from the operational quantities by using the following equations:

$$E \cong H_P(10). \tag{12}$$

$$H_{Skin} \cong H_P(0.07). \tag{13}$$

$$H_{Lense\ of\ eye} \cong H_P(0.07). \tag{14}$$

For strongly penetrating radiation, the critical organ for controlling the development of the severe deterministic effects in individual is the red marrow [1, 4]. The ICRU did not recommend depth for controlling the dose in the red marrow. For practical reasons, monitoring of the red marrow through dosimeters calibrated for $H_P(10)$ could be acceptable. The RBE-weighted absorbed dose in the red marrow received from exposure due to external sources can be estimated from the operational quantities by using the following equation:

$$AD_{Red\ Marrow} \cong H_P(10). \tag{15}$$

For weakly penetrating radiation and in emergency exposure situation, the reference depth for controlling severe deterministic effects due to irradiation of the derma of the skin is 0.4 mm and for controlling the severe deterministic effects due to irradiation of shallow soft tissue is 5 mm [1, 4, 9, 10]. The ICRU did not recommend operational quantity for controlling the dose in the skin derma or shallow soft tissue in emergency exposure situation. For practical reasons, monitoring of the skin derma and shallow soft tissue through dosimeters calibrated for $H_P(0.07)$ could be acceptable. The RBE-weighted absorbed dose in the skin derma or in shallow soft tissue received from exposure due to external sources can be conservatively estimated from the operational quantities by using the following equations:

$$AD_{Skin\ derma} \cong (0.07). \tag{16}$$

$$AD_{Soft\ tissue} \cong H_P(3). \tag{17}$$

Based on the foregoing, individual monitoring in planned, emergency and existing exposure situations through individual dosimeters calibrated for $H_P(10)$ and $H_P(0.07)$ could be acceptable.

4.2 Estimation of protection quantities characterizing an internal exposure

Internal doses cannot be measured directly; they can only be calculated from intake of radioactive substance through particular route such as respiratory system in the case of inhalation, gastrointestinal tract in the case of ingestion or wound, and undamaged skin in the case of contamination. The individual intake also could not be directly measured and can only be inferred from individual measurements of other quantities, such as measurements of activity in the body or in excretion samples or activity concentration in foodstuff or the environment. In circumstances where individual monitoring is inappropriate, inadequate, or not feasible, the occupational exposure of workers may be assessed on the basis of workplace monitoring and other relevant information such as location and durations of exposure. Individual measurements include measurements made by both direct and indirect methods. Methods for the measurement of activity content in the body, such as whole-body, lung, or thyroid counting, are examples of direct methods. Measurements of activity in collected biological samples or measurements made using personal air sampling are examples of indirect methods.

The conceptual framework for the assessment of doses from individual or environmental measurements is illustrated in **Figure 5**.

Chemical and physical properties of radioactive substance S containing radionuclide, the route of intake into the human body P, and value of intake $I_{S,P}$ composite a base for estimation of the protection quantities from internal exposure.

The measurable characteristics of internal exposure are quantities of body content or excretion rate of radionuclide M and concentration of radioactive substance in the environmental media $C(t)_S$.

According to the scheme in **Figure 5**, the value of the intake of radioactive substance S could be obtained by multiplying the integrated over time the measured concentration of radioactive substance in the environmental media $C(t)_S$ by the appropriate value of $v_P(g)$:

$$I(g)_{S,P} = v_P(g) \times \int_0^t C(x)_S dx, \tag{18}$$

where $v_P(g)$ is the consumption rate of an individual of age group g through route P and t is the period of consumption.

According to the scheme in **Figure 5**, the value of the intake of radioactive substance S is obtained by dividing the measured body content or excretion rate M by the appropriate value of $m(t,g)_{S,P}$:

$$I(g)_{S,P} = M/m(t,g)_{S,P}, \tag{19}$$

where $m(t,g)_{S,P}$ is the fraction of an intake that remains in the body (for direct methods) or that is being excreted from the body (for indirect methods) at time t after the intake. This fraction depends on the radionuclide, its chemical and physical form in substance S, the route of intake P, the age group of age g, and the time t [29, 30].

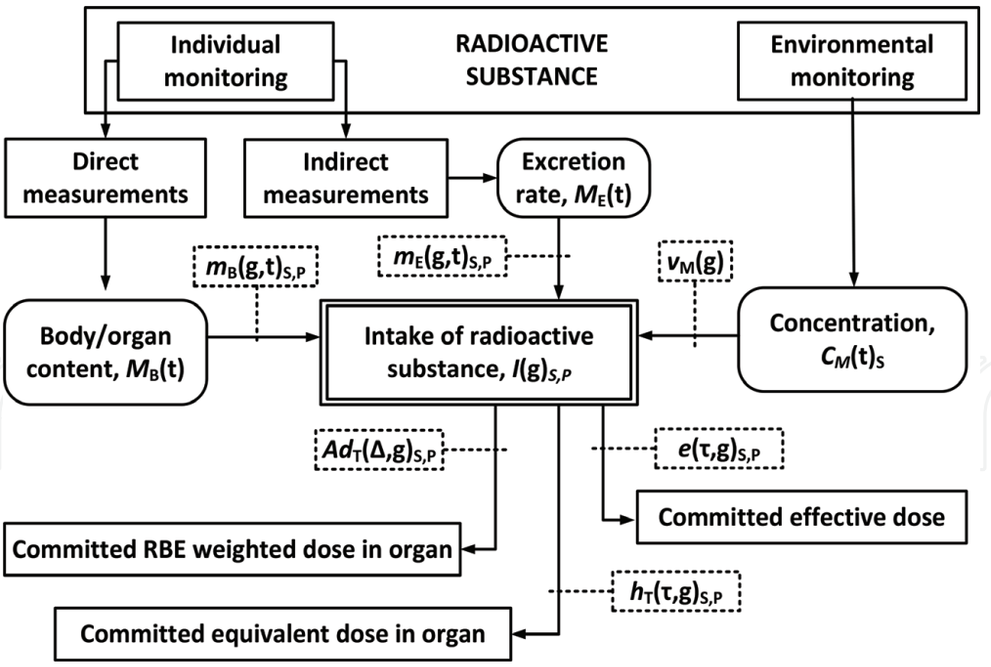

Figure 5.
General scheme for assessment of individual internal dose from monitoring.

Special attention has to be paid to the interpretation of bioassay measurements after the use of means for blocking the uptake of radionuclides or for enhancing their excretion, such as the administration of diuretics, laxatives, or blocking or chelating agents, as well as after the removal of contamination and/or surgical intervention at a wound site. These techniques influence and modify the biokinetic behavior of the incorporated radionuclides, thus invalidating the use of the standardized modeling approach for estimating intake and dose from the bioassay measurements.

In such cases, alternative approaches have to be employed, such as discarding data on excretion for excretion samples collected during the period in which excretion rates may be assumed to have been influenced by the treatment or modifying the standard models in order to take into account the effect of the treatment. Examples of analyses performed after the administration of the chelating agent Ca-DTPA (a calcium salt of diethylenetriaminepentaacetic acid) in cases of accidental intakes of actinides can be found in [31–33]. Bioassay measurements for dose assessment purposes are performed after a certain time period, posttreatment with Ca-DTPA, until the excretion of the radionuclide stabilizes in urine samples. Influence of iodine thyroid blocking on biokinetic of iodine in human body is discussed in [34]. Influence of administration of Prussian blue $\{Fe_4[Fe(CN)_6]_3\}$ on biokinetic of Cs is discussed in [35].

The specific protection quantities of committed RBE-weighted organ dose $AD_T(\Delta,g)$, committed equivalent organ dose $H_T(\tau,g)$, and committed effective dose $E(\tau,g)$ are used for evaluation of dose in an internal dosimetry:

$$AD_T(\Delta,g) \cong \sum_{S,P} \left\{ I(g)_{S,P} \times Ad_T(\Delta,g)_{S,P} \right\}, \tag{20}$$

$$H_T(\tau,g) = \sum_{S,P} \left\{ I(g)_{S,P} \times h_T(\tau,g)_{S,P} \right\}, \tag{21}$$

$$E(\tau,g) = \sum_{S,P} \left\{ I(g)_{S,P} \times e(\tau,g)_{S,P} \right\}, \tag{22}$$

where $Ad_T(\Delta,g)_{S,P}$ and $h_T(\tau,g)_{S,P}$ are respectively the committed RBE-weighted or equivalent dose in the organ or tissue T due to intake of 1 Bq of radionuclide substance S through pathway P, by the group of age g. These dose factors are the time integrals of the relevant dose rates:

$$Ad_T(\Delta,g)_{S,P} = \int_0^{\Delta} \dot{Ad}_T(t,g)_{S,P}\,dt, \tag{23}$$

$$h_T(\tau,g)_{S,P} = \int_0^{\tau} \dot{h}_T(\mathrm{t},g)_{S,P}\,dt, \tag{24}$$

$$e(\tau,g)_{S,P} = \sum_T \left\{ h_T(\tau,g)_{S,P} \times w_T \right\}, \tag{25}$$

where $\dot{Ad}_T(t,g)_{S,P}$ is the RBE-weighted dose rate in organ or tissue T due to intake of 1 Bq of radionuclide substance S through pathway P and $\dot{h}_T(\mathrm{t},g)_{S,P}$ is the equivalent dose rate in organ or tissue T due to intake of 1 Bq of radionuclide substance S through pathway P. For estimation of committed RBE-weighted dose, Δ is taken to be 30 days. The τ is the integration time elapsed after an intake of radioactive substances. When τ is not specified, it will be taken to be 50 years for adults and the time to age 70 years for intakes by children and persons younger than 20.

Various biokinetic models for calculating the values of $m(t,g)_{S,P}$, $Ad_T(\Delta,g)_{S,P}$ and $h_T(\tau,g)_{S,P}$ have been developed.

Values of $m(t,g)_{S,P}$ at selected times for a subset of radionuclides have been reported by the ICRP in graphical and tabular form [36, 37]. A compilation of $m(t,g)_{S,P}$ by workers in emergency and planned exposure situation is presented in various publications of the IAEA [29, 38]. The Internet resources containing the values of $m(t,g)_{S,P}$ different radioactive substances are presented in [39].

Special attention has to be paid to the patterns of inhalation intake in nuclear emergency. Chemical form of radionuclides in aerosol particles and their behavior in the human respiratory tract could be very specific [40, 41] and significantly different from these observed in planned exposure situation and presented in Table II.2C of GSR Part 3 [5].

A compilation of dose coefficients $Ad_T(\Delta,g)_{S,P}$ for intakes of radionuclides by adults is presented in publications of the IAEA [9, 29].

A compilation of dose coefficients $e(\tau,g)_{S,P}$ for ingestion and inhalation intakes of radionuclides by workers is presented in ICRP Publication 119 [42]. A compilation of $h_T(\tau,g)_{S,P}$ and $e(\tau,g)_{S,P}$ for workers and all age groups of members of the public can also be found in the ICRP database in [43]. Data for fetus or embryo due to inhalation or ingestion of radioactive substance by mother are presented in [44, 45] and for infant in [46, 47]. These dose coefficients are based on the calculation methods and parameters given in ICRP Publication 60 [15]. The current published values of $e(\tau,g)_{S,P}$ will be superseded in due course by new values [6, 36, 42] based on updated biokinetic models and on the methods of calculation and the parameters given in ICRP Publication 103 [8]. A compilation of $m(t,g)_{S,Wond}$, $h_T(\tau,g)_{S,Wound}$ and $e(\tau,g)_{S,Wound}$ for penetration of radioactive substance through wound is presented in [48, 49].

4.3 Estimation of protection quantities characterizing a total exposure

The protection quantities E and H_T relate to the sum of the effective doses or equivalent doses, respectively, received from exposure due to external sources within a given time period and the committed effective doses or committed equivalent doses, respectively, from exposure due to intakes of radionuclides occurring within the same time period.

The total equivalent dose in organ or tissue T received or committed during a given time period in individuals from age group g can be estimated from the operational quantities by using the following equation:

$$H_T(g) \cong H_P(10) + H_T(\tau, g), \tag{26}$$

where $H_P(10)$ is the personal dose equivalent from external exposure and $H_T(\tau, g)$ is assessed by Eq. (21).

The total effective dose E received or committed during a given time period in individuals from age group g can be estimated from the operational quantities by using the following equation:

$$E(g) \cong H_P(10) + E(\tau, g), \tag{27}$$

where $E(\tau, g)$ is assessed by Eq. (22).

In the calculation of the committed dose from specific radionuclides, allowance may need to be made for the characteristics of the material taken into the body through inhalation, ingestion, and contaminated wound.

5. Conclusions

To characterize the emergency exposure situation in event of nuclear or radiological emergency, doses to the members of the public, workers, emergency workers, as well as patients and helpers, if applicable, have to be derived from source monitoring, environmental monitoring, or individual monitoring, or from a combination of these. Result of dose assessment needs to be expressed in terms of protection quantities defined in the IAEA General Safety Requirements (GSR) Part 7 [1] and GSR Part 3 [5]. Dose assessment has to be based on the best available monitoring data and has to be promptly updated if any new information relevant for dose assessment becomes available.

The dose assessment has to be as realistic as possible, and in any case, doses for situations in which persons might be in danger of being harmed are not to be underestimated. Overestimation also has to be avoided because there are risks associated with protective actions.

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