Chapter 2
Basic Concepts in Radiation Dosimetry

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2.1 Introduction

Directly and indirectly ionizing radiations deposit their energy in a medium while passing through it. Radiation dosimetry is a procedure that deals with the methods for quantitative determination of that deposited energy. To be more specific, quantitative determination of energy absorbed in a given medium by directly or

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M. Maqbool (ed.), *An Introduction to Medical Physics*, Biological and Medical Physics, Biomedical Engineering, DOI 10.1007/978-3-319-61540-0_2

indirectly ionizing radiations is called radiation dosimetry. It plays a crucial role in radiation therapy, nuclear medicine and radiation protection. Due to its significance, accurate determination of the deposited energy (often termed a radiation absorbed dose) at the point of interest in the medium (i.e., human body or phantom) is needed. A number of quantities and units have been defined for describing the radiation beam, which ultimately leads to the determination of radiation absorbed dose to the medium by incident radiation. These quantities and units are explained in this chapter. Furthermore it covers the fundamental ideas and principles involved in radiation dosimetry.

2.2 Definitions of Dosimetric Quantities

2.2.1 Radiation and Energy Fluence

Generally, fluence is termed as the flux of radiation particles (e.g. photons, electrons, protons heavy ions, neutrons etc.) or energy integrated over a period of time. For radiation, it is defined as the total number of radiation particles that intersect a unit area in a specific time interval. It has units of number of the particles per m^2 (#/m^2 or simply, m^{-2}). Whereas, energy fluence which is measured in Joul/m^2 is defined as the energy delivered per unit area.

If dN is the number of the particles incident on a cross-sectional area dA (see, Fig. 2.1) then, the particle fluence (Φ), is given by [Nahum A (2007), Podgorsak (2005), Seuntjens et al. (2005)]:

$$\Phi = \frac{dN}{dA} \tag{2.1}$$

To make Φ independent of the incident angle of the radiation, it is assumed that the direction of each radiation particle is perpendicular to dA as shown in Fig. 2.1. The directional dependent Φ is often called planar particle fluence. It is defined as, the number of particles crossing a plane of unit area.

Now, if, dR is the radiant energy incident on a sphere of cross-sectional area dA, then mathematically, the energy fluence (Ψ) can be expressed as follows [Nahum A (2007), Podgorsak (2005), Seuntjens et al. (2005)]:

$$\psi = \frac{dR}{dA} \tag{2.2}$$

For a mono-energetic radiation beam, R is the product of N number of the particle and their energy E (i.e. $dR = dNE$). Equation (2.2) can now be written as follows [Seuntjens et al. (2005)]:

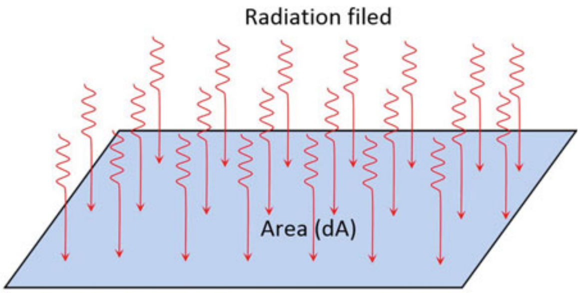

Fig. 2.1 Particle/energy incident on a unit area

$$\psi = \left(\frac{d(NE)}{dA}\right) = \left(\frac{dN}{dA}\right)E = \phi E \tag{2.3}$$

Equation (2.3) gives a relationship between Φ and Ψ for mono-energetic beams. However, in most cases, electromagnetic or particle radiation are not mono-energetic and the above mentioned concepts need to be modified accordingly. In these cases, the particle fluence and energy fluence is replaced by particle fluence spectrum $\Phi_{\mathrm{E}}(E)$, and energy fluence spectrum $\Psi_{\mathrm{E}}(E)$ differential in energy E, respectively and mathematically these are given as [Seuntjens et al. (2005)]:

$$\varphi_{\mathrm{E}}(E) = \frac{d\varphi(E)}{dE} \tag{2.4}$$

$$\psi_{\mathrm{E}}(E) = \frac{d\psi(E)}{dE} = \frac{d\varphi(E)}{dE}E \tag{2.5}$$

2.2.2 Radiation and Energy Fluence Rate

The particle and energy fluence rate are also useful quantities that describe mono-energetic photon and charged particle beams.

Mathematically, the particle fluence rate $\dot{\phi}$ is given as [Seuntjens et al. (2005)]:

$$\dot{\phi} = \frac{d\phi}{dt} \tag{2.6}$$

Here, $d\phi$ is the rate of change of the radiation particle fluence in time interval dt. It is measured in the units of $\mathrm{m}^{-2}\ \mathrm{s}^{-1}$.

Similarly, the energy fluence rate is given by [Podgorsak (2005), Seuntjens et al. (2005)]:

$$\dot{\psi} = \frac{d\psi}{dt} \tag{2.7}$$

Where, $d\psi$ is the rate of change of the energy fluence in time interval dt. $\dot{\psi}$ is measured in the units of W.m^{-2}.s^{-1} or J.m.s^{-1}.

2.2.3 KERMA

The indirectly ionizing radiation such as photons and neutrons encounter different interactions while passing through a medium. In case of photons, these interactions include, Rayleigh (coherent) scattering, Compton (incoherent) scattering, pair (including triplet and higher order) production and the photoelectric effect [Podgorsak EB (2005), Stabin MG (2007), Seuntjens JP (2005)]. For neutron interactions, neutrons-nuclei is the main mode of transfer of its energy to the medium. However, both for photons and neutrons, some of kinetic energy of the incident radiation is transferred to the primary ionizing particles produced inside the medium. Photoelectrons, Compton electrons, or positron–electron pairs are the resultant charged particles produced in case of photon interactions and the scattered nuclei, α-particles and heavy ions in the case of fast neutrons. The above mentioned interactions for photons are governed by their individual microscopic cross-section, and the mass attenuation coefficients [Podgorsak EB (2005), Stabin MG (2007)]. As there is no energy transfer in case of Rayleigh scattering, therefore, this will not be discussed further. The initial kinetic energy transferred to the primary ionizing particles from indirectly ionizing radiation is known as KERMA, K. It is measured in joules per kilogram or Gy (these units will be explained in the later section of this chapter). KERMA is an acronym for *K*inetic *E*nergy *R*eleased per unit *MA*ss. From the above discussion, K can be defined as "the measure of energy transfer from indirectly ionizing radiation (i.e., photons and neutrons) to ionizing radiation (i.e., electrons, protons, α-particles and heavy ions) inside a medium". What happens to these charged particles later on, has nothing to do with the KERMA [Cember and Johnson (2009), Seuntjens et al. (2005)].

Figure 2.2 gives a clear picture of KERMA. Here, indirectly ionizing radiation incident on a medium of volume '*V*'. Some of these radiations will transfer their kinetic energy to secondary charged particles inside volume 'V' (i.e., $E_{\mathrm{K},1}$ and $E_{\mathrm{K},2}$) and some may interact outside and will transfer their energy to the secondary charged particles (i.e., $E_{\mathrm{K},3}$). The collisional energy transferred to volume '*V*' is; $E_{\mathrm{tr}} = E_{\mathrm{K},1} + E_{\mathrm{K},2}$.

Now, for N particles, if, $d\bar{E}_{tr}$ is the mean energy transferred from indirectly ionizing radiation to charged particles per unit mass of volume '*V*', then mathematically K can be written as [Attix FH (2004), Nahum A 2007, Seuntjens et al. (2005), Stabin (2007)]:

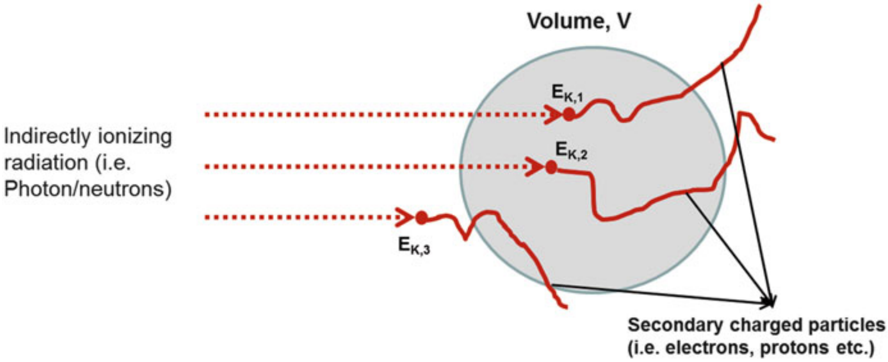

Fig. 2.2 Illustration of KERMA

$$K = \frac{d\bar{E}_{\mathrm{tr}}}{dm}. \tag{2.8}$$

The kinetic energy transferred to charged particles may be spent in two different ways:

(a) Local dissipation of the energy in ionization and excitation as a result of Coulomb-force interactions with atomic electrons in the medium. This is called collisional KERMA.
(b) Radiative loss of the energy due to Coulomb interaction of charged particles with the atomic nuclei. This is called radiative interaction. The bremsstrahlung x-rays produced are more penetrating and thus deposit their energy away from the point of interaction.

The KERMA can be expressed as [Attix FH (2004), Nahum A (2007)]:

$$K = K_{\mathrm{c}} + K_{\mathrm{r}} \tag{2.9}$$

where, K_{c} and K_{r} are the collisional and radiative KERMA respectively.

The average fraction of the energy transferred to the primary charges within the medium that is lost through radiative processes is represented by a factor called the radiative fraction, denoted by $\bar{g}$. K_{c} and K are related to each other as [Nahum A 2007]:

$$K_{\mathrm{c}} = K\left(1 - \bar{g}\right) \tag{2.10}$$

2.2.4 CEMA

CEMA is the acronym for *C*onverted *E*nergy per unit *MA*ss. The analogy of CEMA (C) is same as KERMA despite that it is defined for directly ionization radiation. It is a measure of the energy lost by directly ionizing radiation (i.e., electrons, protons, heavy ion beam etc.) except secondary charged particles to a medium, without the concern as to what happens after this transfer. In case of CEMA, the energy is transferred to the medium through various charged particle interactions. Mathematically 'C' can be written as [Seuntjens et al. (2005)]:

$$C = \frac{d\bar{E}_{\mathrm{tr}}}{dm}. \tag{2.11}$$

Here, $d\bar{E}_{\mathrm{tr}}$ is the energy lost by primary charged particles, as a result of collisions per unit mass dm of the medium. It is also measured in joules per kilogram or Gray (Gy) [Seuntjens et al. (2005)].

2.2.5 Absorbed Dose

The energy transferred to the primary charged particles per unit mass from incident radiation beam (either directly or indirectly ionizing radiation) may not be necessarily absorbed in the volume of interest. Some of this energy, as discussed earlier is absorbed elsewhere after escaping the volume [Attix FH (2004)].

Now, if, R_{in} and R_{out} is the radiant energy entering and leaving a volume (V) respectively and $\sum Q$ is the sum of changes of all mass-energy conversions of nuclei and of all particles involved in the interaction within V (as illustrated in Fig. 2.3), the mean energy ($\bar{\varepsilon}$) imparted to the medium of volume 'V' can be written as (Nahum A 2007):

$$\bar{\varepsilon}_{\mathrm{tr}} = R_{\mathrm{in}} - R_{\mathrm{out}} + \sum Q \tag{2.12}$$

From the above equation, absorbed dose is a measure of the amount of energy imparted $d\bar{\varepsilon}$ by secondary ionizing radiation to a volume V containing a finite mass dm [Attix FH (2004), Nahum A (2007), Seuntjens et al. (2005)]:

$$D = \frac{d\bar{\varepsilon}}{dm} \tag{2.13}$$

More simply, it can be defined as the energy absorbed per unit mass of any material. It is a non-stochastic quantity and is applicable to both directly and indirectly ionizing radiations.

The energy imparted from indirectly ionizing radiation to the medium is a two steps process. Firstly, the energy is transferred to the secondary charged particles (mainly electrons)[Cember and Johnson (2009), Hendee WR et al (2005)]. This

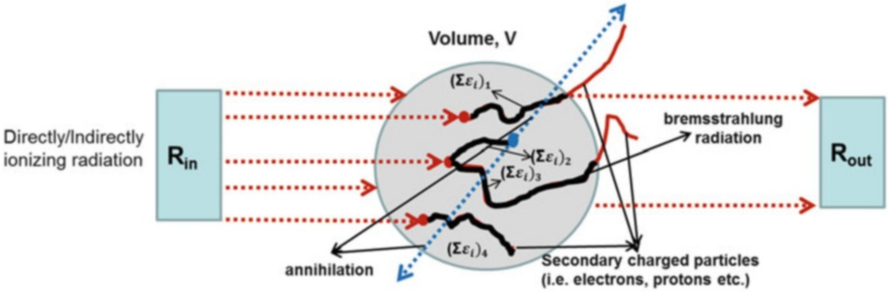

Fig. 2.3 Schematic diagram for energy imparted to volume '*V*'

process is described as KERMA. In the second step, some of the kinetic energy of the charged particles (i.e., KERMA) is transferred to the medium through various interactions (i.e., atomic excitations and ionizations etc.) within the medium (resulting in absorbed dose) and the remaining kinetic energy is lost in the form of radioactive losses (i.e., bremsstrahlung and annihilation in flight).

From the above discussion, it is concluded that both KERMA and dose have same physical dimensions (i.e. same units) but are different dosimetric quantities. The KERMA is a measure of all the energy transferred from the indirectly ionizing radiation (photon or neutron) to primary charged particles per unit mass, while absorbed dose is a measure of the energy absorbed per unit mass.

In an absorbing medium, KERMA decreases continuously with increasing depth. This is due to the continuous decrease in the flux of indirectly ionizing radiation. On the other hand, the absorbed dose is initially lower than KERMA level at the surface and below the surface (to some extent) in the medium. The increasing trend of absorbed dose continues until a maximum is reached. With increasing depth the ionization density increases due to the production of secondary charged particles. After reaching a maximum value, the dose starts decreasing with increasing depth. In fact, the maximum range of the primary charged particles determine the depth of maximum absorbed dose [Cember and Johnson (2009), Hendee WR et al (2005)]. A relation between KERMA and dose for photon/fast neutrons is shown in Fig. 2.4.

2.2.6 *Exposure*

The X- or γ radiation interaction with matter leads to the production of ion pairs. The simplest way to measure the quantitative effects of these radiations is to measure the number of ion pairs produced in air by using oppositely charged surfaces (typically charged metallic plates) [Podgorsak EB (2005), Stabin MG (2007)]. This approach is called exposure which was introduced early in the history of radioisotope research and in the design of early radiation monitoring devices.

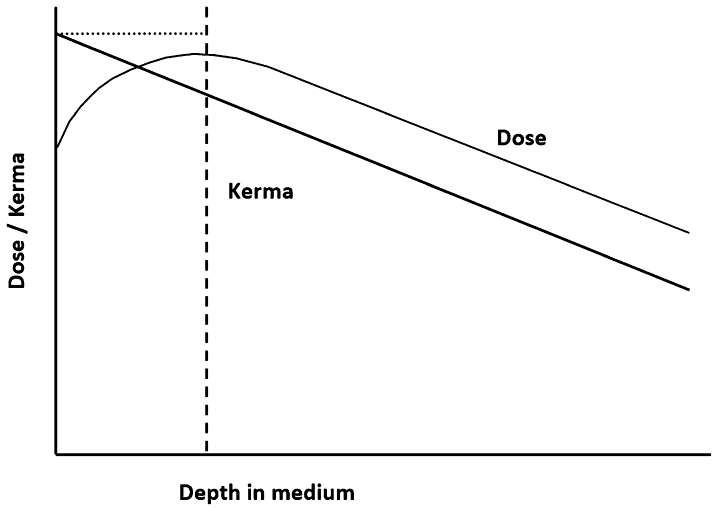

Fig. 2.4 KERMA and absorbed dose relationship for photon and neutrons radiations

Exposure is defined as the total charge *dQ* of either sign produced by X- or γ radiation in air of mass *dm*. The ions must not escape the air and must be collected. Mathematically it can be presented as follows [Attix FH (2004)]:

$$X = \frac{dQ}{dm} \tag{2.14}$$

According to the definition of exposure, it is measured in coulomb per kilogram (C/kg) in air which is the SI unit of exposure. However, there is no special name for the SI unit of exposure. For convenience here, its unit is being called an exposure unit (X-unit). Exposure is considered a convenient and useful quantity for describing *X*- or γ-rays, because of the following reasons:

(a) The energy fluence Φ_E is proportional to the exposure X. This is important in the determination of absorbed dose in other medium, if exposure is known for any given photon energy.

$$X \propto \left(\frac{\mu_{\mathrm{en}}}{\rho}\right)_{\mathrm{E,air}} \tag{2.15}$$

(b) The effective atomic number of air is approximately equal to that of soft biological tissue. Air filled ionization chambers are therefore used in most of the dosimetric measurements. This implies that quantitatively in soft tissue the quotient of collisional KERMA $\boldsymbol{K_c}$ and exposure X, is almost photon energy independent. In the energy range 4 keV to 10 MeV the following relation approximately holds:

$$\frac{(\mu_{\text{en}}/\rho)_{\text{E, tissue}}}{(\mu_{\text{en}}/\rho)_{\text{E, air}}} \approx 1.07 \pm 3\% \tag{2.16}$$

In early days (until late 1970s) exposure was the only quantity used for the calibration of ionization chambers. Currently majority of the chambers are calibrated in terms of absorbed dose to water. For photon energies greater than few mega electron volts, exposure measurement is impractical. Exposure is therefore limited to X- or γ-rays with energies lower than 3 MeV.

2.3 Units of Dosimetric Quantities

2.3.1 Gray

The physical dimension of KERMA, CEMA and absorbed dose is [E/m] (i.e. E = energy & m = mass of any material) is the same [Nahum A (2007), Podgorsak EB (2005)]. Hence, these can be measured in J/kg or erg/g. However, special unit defined for these quantities in the SI system is called the gray (Gy) that corresponds to the absorption of one joule per kilogram (i.e. 1Gy = 1 J/kg). Gray is also used for KERMA and CEMA [Cember and Johnson (2009)].

It is universally applicable in the dosimetry of all types of ionizing radiation (i.e. irradiation due to external fields of electromagnetic radiation, neutrons, or charged particles as well as due to the internally deposited radionuclides).

Example 2.3 During the cancer treatment of a patient, 6 Joule energy is deposited in a 1.5 kilogram tissue exposed to radiation. Find the absorbed dose delivered in Gy to the tissue? How much energy is needed to deliver the same absorbed dose to a tissue whose mass is 0.6 kilogram?

Solution The amount of energy deposited in the tissue = dE = 6 J

Mass of the exposed tissue = dm = 1.5 kg

Absorbed dose = D =?

Using Eq. (2.6)

$$D = dE/dm = 6/1.5$$
$$\boldsymbol{D = 4\ \text{Gy}}$$

In the second part of the question we have to calculate the deposited energy from the calculated D. In this case D = 4 Gy, dm = 0.6 kg. Using and rearranging the same equation,

$$dE = D.dm$$
$$\text{Integrating,} \quad E = D.m$$
$$E = 4 \times 0.6$$
$$\boldsymbol{E = 2.4\ \mathrm{J}}$$

2.3.2 RAD

The rad is an acronym for "radiation absorbed dose" and was introduced before the introduction of the SI units. One rad is defined as [Cember and Johnson (2009), Stabin (2007)]:

$$1\ \ \text{rad} = \frac{100\,\text{ergs}}{\text{g}} \tag{2.17}$$

Since, $1\ \ \text{J} = 10^7 \text{ergs} \,\&\, 1\ \ \text{kg} = 10^3\text{g}$, hence,

$$1\ \ \text{rad} = \left(\frac{100\ \text{ergs}}{\text{g}}\right) \times \frac{(1\text{J}/10^7\text{ergs})}{(1\text{kg}/10^3\text{g})} = 10^{2+3-7} \times \left(\frac{\text{J} \times \text{ergs} \times \text{g}}{\text{kg} \times \text{ergs} \times \text{g}}\right) = 0.01\,\frac{\text{J}}{\text{kg}}$$
$$= 0.01\ \ \text{Gy} = 1\ \ \text{cGy}$$

or,

$$1\ \ \text{Gy} = 100\ \ \text{rad} \tag{2.18}$$

2.3.3 Exposure Unit

As mentioned earlier SI unit of exposure is measured in X unit. By definition exposure X is [Cember and Johnson (2009)]:

$$1\text{X} = \left(1\tfrac{\text{C}}{\text{kg}}\right)_{\text{air}} \times \left(\frac{1\ \ \text{ion}}{1.6 \times 10^{-19}\text{C}}\right) \times \left(\frac{34\ \ \text{eV}}{\text{ion}}\right) \times \left(\frac{1.6 \times 10^{-19}\text{J}}{\text{eV}}\right) \times \left(1\frac{\text{Gy}}{\text{J/kg}}\right)$$
$$= 34\ \ \text{Gy}_{\text{air}} \tag{2.19}$$

2.3.4 Roentgen

The conventional unit for exposure is roentgen R, which is equal to 2.58×10^{-4} C/kg in the SI system.

1R can be expressed as [Cember and Johnson (2009)]:

$$1\ \mathrm{R} = \left(1\,\frac{\mathrm{sC}}{\mathrm{cm}^3}\right)_{\mathrm{air}} \times \left(\frac{1\ \mathrm{cm}^3\mathrm{air}}{1.29 \times 10^{-3}\ \mathrm{g/cm}^3\mathrm{air}}\right) \times \left(\frac{1\mathrm{ion}}{4.8 \times 10^{-10}\mathrm{sC}}\right) \times \left(\frac{34\ \mathrm{eV}}{\mathrm{ion}}\right) \times \left(\frac{1.6 \times 10^{-12}\mathrm{erg}}{\mathrm{eV}}\right) \times \left(\frac{1\ \mathrm{rad}}{100\ \mathrm{erg/g}}\right) = 0.877\ \mathrm{rad}_{\mathrm{air}} \tag{2.20}$$

where, sC is charge in statcoulomb ($1\mathrm{sC} = 3 \times 10^9$ C), and 1 erg $= 1 \times 10^{-7}$ Joul.

The relationship between the X and R may be calculated as follows:

$$\frac{34\left(\frac{\mathrm{J/kg}}{\mathrm{C/kg}}\right) \times \left(\frac{10^7\mathrm{erg}}{\mathrm{J}}\right) \times \left(\frac{1\mathrm{kg}}{1000\mathrm{g}}\right)}{\left(\frac{87.7\mathrm{erg/g}}{\mathrm{R}}\right)} = 3877\left(\frac{\mathrm{R}}{\mathrm{C/kg}}\right) \tag{2.21}$$

or

$$\begin{aligned} 1\mathrm{X} &= 3877\ \mathrm{R} \\ 1\mathrm{R} &= (1/3877)\mathrm{X} = 2.58 \times 10^{-4}\mathrm{X} \end{aligned} \tag{2.22}$$

Example 2.1 In an experiment, 2 kg of dry air is exposed to x-rays. As a result of ionization in dry air, 5.16×10^{-3} C charge is produced. Determine the Exposure in the units of Roentgen.

Solution Quantity of charge produced as a result of air ionization $= 5.16 \times 10^{-3}$ C.

Quantity of dry air exposed to x-rays = 2 kg

$$\begin{aligned} \text{Exposure} &= 5.16 \times 10^{-3}\mathrm{C}/2\ \mathrm{kg} \\ &= 2.58 \times 10^{-3}\mathrm{C/kg} \\ &= 25.8 \times 10^{-4}\mathrm{C/kg} \\ &= 10 \times 2.58 \times 10^{-4}\mathrm{C/kg} \\ &= \mathbf{10\ Roentgen} \\ \text{Thus,}\quad \textbf{Exposure} &= \mathbf{10\ R} \end{aligned}$$

Example 2.2 Calculate the quantity of charge produced as a result of 4 R Exposure in dry air of mass 0.008 mg. If this charge is the result of electrons production then find the number of electrons produced.

Solution Exposure = 4 R = $4 \times 2.58 \times 10^{-4}$ C/kg = 1.032×10^{-3} C/kg

Mass of the air ionized = 0.008 mg = 8×10^{-9} kg, (1 mg = 10^{-6} kg).

$$\begin{aligned}\text{Charge produced} = q &= \text{Exposure} \times \text{mass}\\ &= 1.032 \times 10^{-3}(\text{C/kg}) \times 8 \times 10^{-9}\text{kg}\\ \textbf{Charge produced} = \boldsymbol{q} &= \mathbf{8.256{\times}10^{-12}C}\end{aligned}$$

Magnitude of the charge of a single electron $= e = 1.6 \times 10^{-19}$ C
Number of electrons produced $= n = ?$

$$\begin{aligned}n = q/e &= 8.256 \times 10^{-12}\text{C}/1.6 \times 10^{-19}\text{C}\\ &= 5.16 \times 10^{7}\end{aligned}$$

Number of electrons produced $= \boldsymbol{n} = \mathbf{5.16} \times \mathbf{10^{7}}$

2.4 Relationship between Dosimetric Quantities

In this section, correlations will be established between quantities that describe radiation fields (photons fluence and energy fluence etc.) and other dosimetric quantities (KERMA, radiation dose and exposure etc.).

2.4.1 Fluence—KERMA Relationship

Consider a thin layer of a medium of thickness dl and surface area dA (i.e. volume $V = dl \times dA$). Suppose N is the number of photons each having energy E, entering into this volume element perpendicularly as shown in Fig. 2.5.

The interaction coefficient for this medium will give the amount of energy transferred to the medium from incident photons[Ahmad N (1999), Martin JE (2013)]. The mass energy-transfer coefficient (μ_{tr}/ρ) is given as [Nahum A (2007)]:

$$\frac{\mu_{\text{tr}}}{\rho} = \frac{1}{\rho dl}\frac{dR_{\text{tr}}}{R} \tag{2.23}$$

The fraction of incident radiant energy $\frac{dR_{\text{tr}}}{R}$ can be written as $\frac{dE_{\text{tr}}}{N \times E}$, and by rearranging Eq. (2.23), we have [Nahum A (2007)]:

$$dE_{\text{tr}} = \mu_{\text{tr}} dlNE \tag{2.24}$$

Dividing both sides of Eq. (2.24) by the mass of the layer 'dm' and rearranging [Nahum A (2007)]:

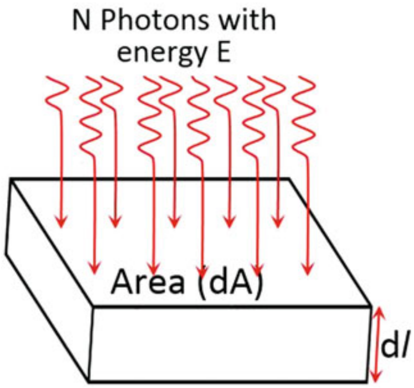

Fig. 2.5 Illustration of N radiation tracks of energy E crossing thin layer of a medium having dl, dA, dm and ρ as thickness, area, mass and density respectively

$$\frac{dE_{\mathrm{tr}}}{dm} = \mu_{\mathrm{tr}} E \left[\frac{Ndl}{dm} \right] \tag{2.25}$$

Now, by replacing dm with ρdV, the following relation can be developed [Nahum A (2007)]:

$$\frac{dE_{\mathrm{tr}}}{dm} = \frac{\mu_{\mathrm{tr}}}{\rho} E \left[\frac{Ndl}{dV} \right] \tag{2.26}$$

where, the left hand side of Eq. (2.26) is KERMA in a medium 'w' and the quantity in the square brackets is the fluence. The above equation will take the following form [Nahum A (2007)]:

$$K_{\mathrm{w}} = \left(\frac{\mu_{\mathrm{tr}}}{\rho} \right) E\phi \tag{2.27}$$

or in terms of energy fluence ψ [Nahum A (2007)]:

$$K_{\mathrm{w}} = \left(\frac{\mu_{\mathrm{tr}}}{\rho} \right)_{\mathrm{w}} \psi \tag{2.28}$$

It is to be noted that the perpendicular incidence of radiation mentioned in Fig. 2.5 was assumed for simplicity. However, Eqs. (2.25) and (2.26) are valid for arbitrary angles of radiation incidence as well.

The above calculations were done for mono-energetic radiation. In majority of practical cases, however, one deals with spectrum of energies. In such situations, Eq. (2.27) can be re-written as follows [Nahum A (2007)]:

$$K_{\mathrm{w}} = \int_{0}^{E_{\max}} E\phi_{\mathrm{E}} \left(\frac{\mu_{\mathrm{tr}}}{\rho}\right)_{\mathrm{w}} dE \tag{2.29}$$

where, the energy dependence of has been shown explicitly.

By definition, using Eq. (2.28), collisional KERMA K_{c} can be expressed in terms of mass energy absorption coefficient (μ_{en}/ρ). It is assumed that, part of the initial kinetic energy of charged particles that is converted into bremsstrahlung photons is excluded from the energy absorbed. The two coefficients (μ_{en} & μ_{tr}) can be related by the following equation [Nahum A (2007)]:

$$\mu_{\mathrm{en}} = \mu_{\mathrm{tr}}(1 - g) \tag{2.30}$$

This is similar to what relates K_c and K in Eq. (2.9).

For mono-energetic beam,

$$(K_{\mathrm{c}})_{\mathrm{w}} = \left(\frac{\mu_{\mathrm{en}}}{\rho}\right)_{\mathrm{w}} E\phi \tag{2.31}$$

whereas for a spectrum of incident photon energies:

$$(K_{\mathrm{c}})_{\mathrm{w}} = \int_{0}^{E_{\max}} E\phi_{\mathrm{E}} \left(\frac{\mu_{\mathrm{en}}}{\rho}\right)_{\mathrm{w}} dE \tag{2.32}$$

2.4.2 KERMA—Absorbed Dose Relationship

Mathematical relationship between KERMA and absorbed dose for photons/neutrons can be derived from fluence-KERMA relationship. As discussed earlier, KERMA is the energy transfer to primary charged particles and the absorbed dose in a medium accounts for the mean value of energy absorbed in an elementary volume. Furthermore, the primary charged particles can leave the elementary volume after getting energy from incident radiation and taking a fraction of the initial transferred kinetic energy out of the volume. The situation is illustrated in Fig. 2.6, where E_{tr}^{n} is the net energy transferred to primary charged particles. Part of the initial kinetic energy that is converted into bremsstrahlung photons escape the volume of interest. Suppose ε is the imparted energy to the volume, E_{in}^{n} and E_{out}^{n} is the net kinetic energy enter and leaves the volume respectively. Then, Eq. (2.12) can be rewrite as [Nahum A (2007)]:

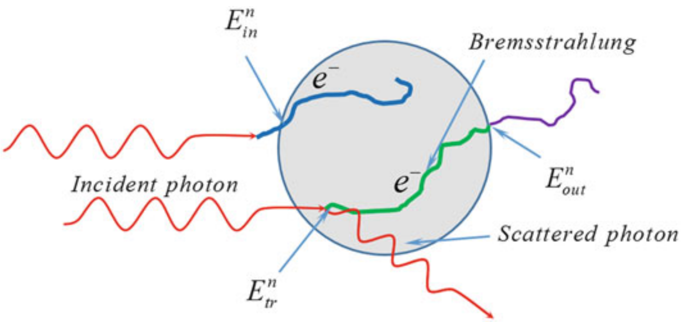

Fig. 2.6 Schematic Illustration for energy imparted (E^n_{tr}) by secondary electrons created by photons, and how the loss of charged particles is compensated

$$\varepsilon = E^n_{\mathrm{tr}} - E^n_{\mathrm{out}} + E^n_{\mathrm{in}} \tag{2.33}$$

Now, if the primary charged particle track that leaves the volume is compensated by an identical track (i.e., $E^n_{\mathrm{out}} = E^n_{\mathrm{in}}$, known as charged particle equilibrium) then Eq. (2.33) can be written as [Nahum A (2007)]:

$$\varepsilon = E^n_{\mathrm{tr}} \tag{2.34}$$

In this case, absorbed dose will be equal to collision KERMA (K_c)

If Eq. (2.34) is divided by the mass of the layer/volume element, the following relation can easily be obtained [Nahum A (2007)]:

$$(D)_{\mathrm{w}} \stackrel{\mathrm{CPE}}{=} (K_{\mathrm{c}})_{\mathrm{w}} \tag{2.35}$$

where CPE is the charged particle equilibrium (will be explained in a later section).

It should be noted that the absorbed dose is equal to the collision KERMA under the special condition of charged particle equilibrium.

For mono-energetic, indirectly ionizing radiation using Eq. (2.35) the following relation can be obtained from Eq. (2.31)[Nahum A (2007)]:

$$D \stackrel{\mathrm{CPE}}{=} \left(\frac{\mu_{\mathrm{en}}}{\rho}\right) E\varphi \tag{2.36}$$

Similarly, in case of spectrum of incident photon energies [Nahum A (2007)]:

$$D \stackrel{\mathrm{CPE}}{=} \int_0^{E_{\max}} E\phi_{\mathrm{E}} \left(\frac{\mu_{\mathrm{en(E)}}}{\rho}\right) dE \tag{2.37}$$

These are very important relationships, often used in radiation dosimetry.

2.4.3 Exposure–Dose Relationship

With the help of ionization chambers, dose is measured in the air cavity of the chamber. Air is however, not always the medium of interest for dose determination. Dose measured in air can be converted to dose in other medium. The energy absorption (dose) is approximately proportional to the electronic density of the medium in the energy region where Compton scattering is dominant. The tissue dose may not be necessarily equal to the air dose. For human tissue the electron density is 3.28×10^{23} electrons/gram, whereas for air it is 3.01×10^{23} electrons/gram. The absorbed dose in tissue, from an exposure of 1 C/kg air is therefore:

$$\begin{aligned} D_{\text{tissue}}(\text{from } 1\,\text{C/kg}) &= D_{\text{air}} \times \frac{\text{tissue electron density}}{\text{air electron density}} \\ D_{\text{tissue}}(\text{from } 1\,\text{C/kg}) &= 34(\text{Gy}) \times \frac{3.28 \times 10^{23}}{3.01 \times 10^{23}} = 37\,\text{Gy} \end{aligned} \tag{2.38}$$

In Eq. (2.38), the ratio of electron densities can be approximated by the ratios of mass absorption coefficients, i.e.,

$$D_{\text{tissue}}(\text{from } 1\,\text{C/kg}) = 34 \times \left(\frac{\mu_{\text{med}}/\rho}{\mu_{\text{air}}/\rho}\right)\text{Gy} \tag{2.39}$$

2.4.4 Roentgen Equivalent Man (rem)

Roentgen equivalent man or rem is one of the old units used in medical and radiation physics. REM has the same relationship with Sievert as RAD has with Gray.

$$\mathbf{1\,Sv = 100\,rem} \tag{2.11}$$

For example a dose of 1500 rem is equivalent to 15 Sv.

The smaller unit, usually used, is millirem, where 1 mrem $= 10^{-3}$ rem.

How much is a millirem? The following information gives an idea of a millirem dose.

- The annual background radiation for a typical American is 370 mrems.
- The average from watching color TV is two mrem each year.
- The nuclear industry contributes to less than 1 mrem/year to an individual's background radiation.

2.5 Relative Biological Effectiveness

The effects of various radiations on a body tissues compared to a reference effect gives the idea of effectiveness of different radiations in providing the same effect. This comparison of the effectiveness is generally known by a term called relative biological effectiveness RBE. The biological effects of radiation are not only directly proportional to the energy deposited per unit mass or per unit volume, but also on the way in which this energy is distributed along the path length of the radiation. Thus the relative biological effectiveness is defined as the ratio of the doses required by two radiations to cause the same level of effect. Thus, the RBE depends upon the dose and the biological end point. Mathematically it is defined as,

$$\mathrm{RBE} = \frac{\text{Dose of reference } \textit{radiation} \text{ required to produce a particular effect}}{\text{Dose of radiation required to produce the same effect}} \tag{2.8}$$

The reference radiation is generally taken as 250 keV x-rays. The RBE for the reference radiation is considered as 1. Suppose that it takes 200 mGy of x-rays but only 20 mGy of neutrons to produce the same biological effect, the RBE would be 200/20 = 10 using x-rays as the reference radiation.

Example 2.4 During an experiment on a certain tissue it is observed that 25% of the tissue cells are damaged by 12 Gy of 250 keV energy x-rays. The same damage is produced to the same tissue by 4 Gy of protons, 1 Gy of neutrons and 0.6 Gy of α-rays in separate trials. Calculate the RBE for protons, neutrons and α-rays used in that experiment.

Solution Using Eq. (2.8) for these radiations *we get,*

$$\begin{aligned}
&\text{RBE for protons} = 12/4 = 3 \\
&\text{RBE for neutrons} = 12/1 = 12 \\
&\text{RBE for } \alpha\text{-rays} = 12/0.6 = 20
\end{aligned}$$

The same biological effect can be provided to the tissues by 20 times smaller dose of α-rays as compared to the dose of x-rays providing the same effect.

2.6 Radiation Weighting Factor, Equivalent Dose and Sievert

The exact dose delivery to body tissues is important when radiations interact with a patient in diagnostics or treatment procedures. However, it must also be paid attention that different kinds of radiations have different effects on tissues even if the quantity of dose is the same. For example, α-rays are more dangerous to the tissues due to their high ionization ability as compared to γ-rays. Thus, in order to take into account the differences in the biological effects of different radiations,

Table 2.1 Radiation weighting factors of various kinds of radiations

Type of radiation	Energy range	Radiation weighting factor, W_R
X-rays and γ-rays	All energies	1
Electrons, positrons and muons	All energies	1
Neutrons	<10 keV	5
Neutrons	10 keV–100 keV	10
Neutrons	>100 keV–2 MeV	20
Neutrons	>2 MeV–20 MeV	10
Neutrons	>20 MeV	5
Protons (other than recoil protons)	>2 MeV	2–5
α-rays, fission fragments and heavy nuclei	All energies	20

ICRU 60 (1998)

International Committee for Radiological Units (ICRU) and International Commission on Radiological Protection (ICRP) introduced a new term called Q*uality Factor*. Quality factor, denoted by '*Q*', is a dimensionless factor or a number that represents the effectiveness of a particular radiation that interacts with human body. Later on its name quality factor was changed by a new term *Radiation Weighting factor* 'W_R'[Grupen C (2010), Khan FM (2005)].

As a result of introducing radiation weighting factor, ICRU defined a new quantity called Equivalent Dose. *Equivalent Dose* is obtained by multiplying radiation weighting factor to the absorbed dose. Equivalent Dose 'H' is mathematically given below.

$$\mathbf{H} = \mathbf{D}.\mathbf{W_R} \tag{2.8}$$

The radiation weighting factors of different radiations are given in Table 2.1.

Equivalent Dose is measured in the units of Sievert 'S_v'. Mathematically, Sievert is the same unit as Gray but conceptually Gy represents the delivery of dose to treat or diagnose an effect. On the other hand, Sievert relates the absorbed dose in human tissue to the effective biological damage of the radiation. Sievert is more concerned about the damaging effect of different radiations to normal tissues.

Example 2.3 A radiation worker received 2 Gy of absorbed dose during a leak in the radiation safety set up. Find the equivalent dose H, if the absorbed is received through (i) γ-rays of energy 5 MeV (ii) neutrons of energy 2.5 MeV (iii) α-particles of energy 1.2 MeV.

Solution Absorbed Dose $= D = 2$ Gy. Using Eq. (2.8), $\boldsymbol{H = D.W_R}$

(i)	5 MeV γ-rays :	$W_R = 1,$	$H = D.W_R = 2 \times 1 = 2$ Sv
(ii)	2.5 MeV neutrons :	$W_R = 10$	$H = D.W_R = 2 \times 10 = 20$ Sv
(iii)	1.2 MeV α-particles :	$W_R = 20$	$H = D.W_R = 2 \times 20 = 40$ Sv

Example 2.4 An equivalent dose of 25 Sv is recommended in a certain application of radiations. How much absorbed dose should be given to achieve the goal if it has to be supplied through (i) positrons
of energy 5 MeV (ii) neutrons of energy 6 keV (iii) protons of energy 2 MeV ($W_R = 2$) (iv) α-rays of energy 3 MeV.

Solution Equivalent dose $= H = 25$ Sv. Using Eq. (2.8) $\boldsymbol{H = D.W_R}$ or $\boldsymbol{D = H/W_R}$

(i)	5 MeV positrons :	$W_R = 1,$	$D = H/W_R = 25/1 = 25$ Gy.
(ii)	6 keV neutrons :	$W_R = 5,$	$D = H/W_R = 25/5 = 5$ Gy.
(iii)	2 MeV protons :	$W_R = 2,$	$D = H/W_R = 25/2 = 12.5$ Gy.
(iv)	3 MeV α-rays :	$W_R = 20,$	$D = H/W_R = 25/20 = 1.25$ Gy

2.7 Linear Energy Transfer

Linear Energy Transfer, represented by LET, is another concept taken into account when radiation dose is provided to patients or radiation exposure of workers occurs in radiation area. When passing through a medium, ionizing radiation may interact with it during its passage and, as a result, deposit energy along its path (called a *track*). The average energy deposited per unit length of track is called *linear energy transfer (LET)*. In simple words It describes the energy deposition ability of a charged radiation and is defined as the amount of energy deposited per unit length of a tissue or a material by radiation. The energy average is calculated by dividing the track into equal energy intervals and averaging the lengths of the tracks that contain that specific energy amount. LET is generally expressed in the unit of keV/μm. Mathematically, the LET is described below.

$$\mathbf{LET} = -\boldsymbol{dE/dX} \tag{2.9}$$

where the negative sign shows more loss in energy when bigger length is traversed by the radiation. The SI unit of LET, obtained from Eq. (2.9), is J/m. However, Joule is a big unit of energy. Similarly, for a small body section or a material, meter is also a big unit measuring length. Therefore, keV/μm is used as a common unit for LET.

Because the amount of ionization produced in an irradiated object corresponds to the amount of energy it absorbs and because both chemical and biologic effects in

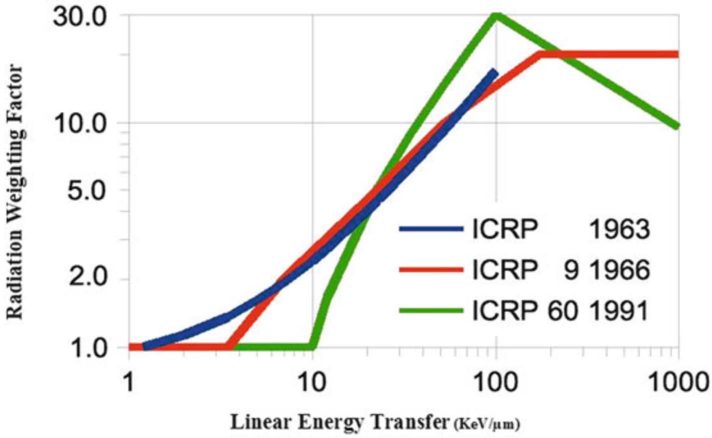

Fig. 2.7 Curves showing variation in radiation weighting factor with changing LET

Table 2.2 Linear energy transfer and radiation weighting factor

LET (keV/μm)	W_R
<3.5	1
3.5–7.0	1–2
7.0–23	2–5
23–53	5–10
53–175	10–20
X-rays and γ-rays of any energy	1
Electron and positron of any energy	1

tissue coincide with the degree of ionization experienced by the tissue, LET is an important factor in assessing potential tissue and organ damage from exposure to ionizing radiation. LET can also be related to the radiation weighting factor of various radiations used in a particular event. Higher is the LET of a particular kind of radiation in certain tissue or material, more energy is deposited by that radiation and hence the quality factor or radiation weighting factor of that radiation is high. LET generally depends upon the nature of radiation as well as the tissue or material exposed.

Figure 2.7 gives a graph between LET and radiation weighting factor given and modified by ICRP. Table 2.2 also gives a comparison and a way of changes in W_T with changes in LET. Though modification was brought by ICRP with time but the overall trend of the curve is still the same [Ahmad N (1999), Martin JE (2013)].

The figure shows that the radiation weighting factor increases to a maximum for LET around 100 keV/μm followed by a decrease with increasing LET. The reason being more energy deposition by a radiation causes more damage to the tissues and hence W_T increases with increasing LET. At LET around 100 keV/μm, the energy of radiation provides maximum damage to the tissues. With LET $>$ 100 a decrease occurs instead of further increase. At this stage the amount of energy deposited in tissues is more than the body cells available to be damaged. Thus, the energy deposition rate is higher than the damage given to the tissues, causing a waste of huge energy as compared to the ratio in which tissues are damaged. Therefore, we can say that a portion of energy is wasted because the same effect can be given to the tissues with much less energy deposition.

2.8 Tissue Weighting Factor and Effective Dose

We are now familiar with the importance of the type of radiation when body tissues are exposed to radiations. Another important factor that must not be ignored when radiation is used to deliver dose to human body in diagnoses and treatment applications is the tissues response to radiation. Even if the same equivalent dose is provided to various parts of the body, the response of different tissues is different. Some tissues are more sensitive to radiation while some others are relatively less sensitive[Grupen C (2010), Khan FM (2005)]. Thus, even under the same equivalent dose, the damage provided to different tissues is different. In order to take into account for the response of various tissues into the interacting radiation a new factor called *Tissue Weighting Factor* 'W_T' is introduced by ICRU and a new quantity called *Effective Dose* 'E' is obtained when W_T is also taken into account during radiation exposure. The effective dose is obtained by multiplying equivalent dose with the tissue weighting factor of the tissue exposed to radiation. Mathematically,

$$\boldsymbol{E} = \boldsymbol{H}.\boldsymbol{W}_{\mathbf{T}} \tag{2.9}$$

Since from Eq. (2.8),

$$\boldsymbol{H} = \boldsymbol{D}.\boldsymbol{W}_{\mathbf{R}}$$

Thus Eq. (2.9) becomes,

$$\boldsymbol{E} = \boldsymbol{D}.\boldsymbol{W}_{\mathbf{R}}.\boldsymbol{W}_{\mathbf{T}} \tag{2.10}$$

Equations (2.9) and (2.10) gives mathematical expressions for effective dose in terms of equivalent dose and absorbed dose, taking into account both radiation weighting factor and tissue weighting factor. Effective dose is always used as a measure of risk.

Tissue weighting factors for various tissues are given in Table 2.3.

2.9 Classification of People Exposed to Radiation

For the purpose of radiation safety and protection, ICRP has classified people exposed to radiation in three categories. Those three categories are listed below.

Table 2.3 Tissue weighting factors W_T of various tissues (ICRP.org)

Organ	Tissue weighting factor W_T, (ICRP 2007)
Breast	0.12
Colon	0.12
Bone Marrow (Red)	0.12
Lung	0.12
Stomach	0.12
Gonads	0.08
Bladder	0.05
Liver	0.05
Thyroid gland	0.05
Esophagus	0.05
Skin	0.01
Bone surface	0.01
Brain	0.01
Salivary glands	0.1
Remainder	0.05
Whole body	1.0

(a) *Radiation Workers*.
This category consists of those people who work directly in radiation area. For example a person working as a radiation safety officer or a person working in a nuclear reactor is classified as radiation worker.
(b) *General Public*.
This category includes common people in public. This category has different exposure limits than the radiation workers.
(c) *Medical Exposure*.
This category consists of patients exposed to radiation for diagnostic or therapeutic purposes.

The dose limits for people in different categories are different. A person from general public cannot receive the same maximum dose or exposure as a radiation worker gets and a patient cannot get the same maximum dose a person from the general public is allowed to obtain.

2.10 Annual Limit on Intake (ALI)

In order to restrict human body against excessive exposure to radiation, ICRP defines the term Annual Limit on Intake. ALI imposes a maximum limit on each individual organ as well as the whole body in each year. However, this maximum limit is different for different categories.

The maximum annual dose limit for all three categories is given below.

Radiation Workers For radiation workers ICRP recommends the following annual effective dose limit.

(i) 500 mSv (50 rem) to all tissues except the lens of an eye.
(ii) 150 mSv (15 rem) to the lens of an eye.
(iii) 50 mSv (5 rem) to the whole body per year, with not more than 100 mSv over five years.

It must be noted that the whole body limit and any individual organ limit must be satisfied at the same time. For example if the lungs of a person is exposed to 30 rem annually and at the same the whole body receives 8 rem then it is not allowed according to the rules.

Mathematically, the above mentioned limits can be summarized as follows.

$$\text{For a single organ} \quad \boldsymbol{W_T H \leq 500\ \mathrm{mSv}} \tag{2.12}$$

$$\text{For the whole body} \quad \sum_{T} \boldsymbol{W_T H \leq 50\ \mathrm{mSv\ per\ year}} \tag{2.13}$$

General Public The ICRP recommendation of the annual limit on intake for a member of general public is 5 mSv or 500 rem.

Medical Exposure No specific dose limit is recommended by the ICRP for medical exposure. The commission, however, did recommend that only necessary exposure should be made.

Example 2.5 Using the tissue weighting factors in Table 2.3, calculate the implied limits for each of the following organs, assuming that each organ is irradiated completely in isolation: the bone surface, the lungs and the bladder.

Solution

Use equation	$\boldsymbol{W_T H \leq 50\ \mathrm{mSv}}$
	$\boldsymbol{H = 50/W_T \mathrm{mSv}}$
For bone surface	$\boldsymbol{H = 50/0.01 = 5000\ \mathrm{mSv}\ (500\ \mathrm{rem})}$
For lungs	$\boldsymbol{H = 50/0.12 = 416.66\ \mathrm{mSv}\ (41.66\ \mathrm{rem})}$
For bladder	$\boldsymbol{H = 50/0.05 = 1000\ \mathrm{mSv}\ (100\ \mathrm{rem})}$

Since the maximum allowed dose to each individual tissue is 500 mSv per year therefore, the bone surface and the bladder can still not have the above calculated dose. Each of these tissue has a maximum of 500 mSv restrictions which must be followed to avoid radiation hazards and damages.

Example 2.6 A student, working in a radiation laboratory, is exposed to radiation and received the given equivalent doses to exposed organs in 1 year. Thyroid = 120 mSv, Stomach = 80 mSv, Bone Marrow = 160 mSv, Bladder = 100 mSv, and Lungs = 140 mSv.

Is he safe according to the ICRP regulations?

Solution Use the equation $\sum W_T H \leq 0.05\ \mathrm{Sv}^{\mathrm{T}}$.

Expanding the sum over all tissues

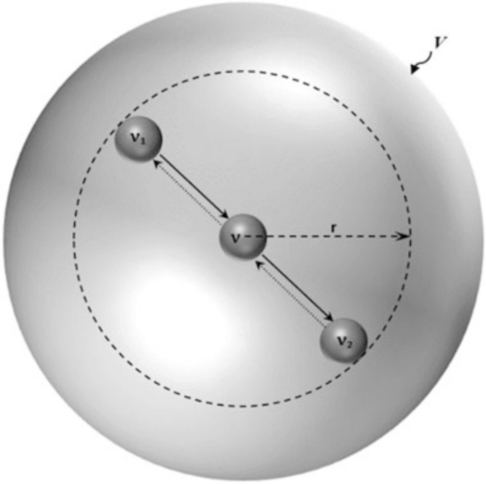

Fig. 2.8 Charged particle equilibrium in an elemental volume ν inside a hypothetical bigger volume V

$$(W_{\mathrm{T}}H)_{\text{thyroid}} + (W_{\mathrm{T}}H)_{\text{stomach}} + (W_{\mathrm{T}}H)_{\text{bone marrow}} + (W_{\mathrm{T}}H)_{\text{bladder}} + (W_{\mathrm{T}}H)_{\text{lungs}} \leq 50\ \text{mSv}$$

$$(0.05 \times 120)_{\mathrm{T}} + (0.12 \times 80)_{\mathrm{S}} + (0.12 \times 160)_{\mathrm{BM}} + (0.05 \times 100)_{\mathrm{B}} + (0.12 \times 140)_{\mathrm{L}}$$

$$= (0.05 \times 120)_{\mathrm{T}} + (0.12 \times 80\text{mSv})_{\mathrm{S}} + (0.12 \times 160)_{\mathrm{BM}} + (0.05 \times 100)_{\mathrm{B}} + (0.12 \times 140)_{\mathrm{L}}$$

$$= 6.0 + 9.6 + 19.2 + 5 + 16.8$$

$$= 56.6\text{mSv}$$

Since 56.6 mSv $>$ 50 mSv thus, the dose exceeds the annual allowable dose limit. Therefore, the student is not safe according to the ICRP regulations.

2.11 Charged Particle Equilibrium

Charged particle equilibrium (CPE) for the volume of interest v exists if each of the outgoing charged particle is replaced by an identical charged particle of the same energy. CPE will generally exist in a homogeneous medium provided that the separation of the outer boundaries of the medium and the volume of interest is greater than the range of the charged particles. Generally, CPE will not exist around the interfaces of heterogeneous media. Absorbed dose measurement at a given point is however independent of CPE. Charged particle equilibrium can be achieved whether the radiation source is inside or outside of the medium. Consider a hypothetical sphere of extended volume V as shown in Fig. 2.8. A radioactive source is uniformly distributed in V, emitting radiations isotropically. Suppose a smaller volume ν inside the bigger V is the volume of interest. In the surrounding, two other elemental volumes (ν_1 and ν_2) are symmetrically defined. Charge particles (solid lines) moving from ν_1 to v are identical to those traveling from ν to ν_2.

On the other side, lesser but identical particles (dotted lines) flow from ν_2 to ν and ν to ν_1. The same is applicable to all symmetrically defined elemental volumes inside the sphere of radius r ($r >$ range of the charged particles). Based on this analogy the net charge leaving and entering volume ν is zero, i.e., CPE exits in this volume.

2.12 Stopping Power

For indirectly ionizing radiations, linear and mass attenuation coefficients are used to determine beam attenuation in a medium of interest. For directly ionizing radiations, however, linear and mass stopping powers are the quantities of relevance. The linear stopping power is defined as the rate of energy loss (*dE)* per unit track length (*dx*) in the medium. The mass stopping power is obtained by normalizing the linear stopping power to the mass density (ρ) of the medium, as in Eq. (2.40).

$$ {}_{\rho}S_{\mathrm{c}} = \left(\frac{dE}{Ndx}\right)_{\mathrm{c}} \tag{2.40} $$

The subscript c and N indicates the collisional stopping power and no. of incident particle respectively. Like KERMA, the total stopping power for directly ionizing radiation can also be split into two components i.e. collisional stopping power (S_{c}), and the energy lost in the form of bremsstrahlung (i.e. radiative component; S_{r}). However, the later part may escape the elemental volume and for dosimetric purposes we are less interested in the radiative component of the stopping power as S_{c} has the main contribution in the locally deposited energy in the elemental volume. Mass stopping power is typically measured in MeV·cm^2/g.

2.12.1 Stopping Power and CEMA

Equation (2.40) can be rewrite as:

$$ dE = {}_{\mathrm{\rho}}S^{\rho}\mathrm{c}Ndx \tag{2.41} $$

Now dividing both sides by the mass of the layer dm. by using $dm = \rho dV$ and rearrangements of Eq. (2.41) will lead us to the result:

$$ \frac{dE}{dm} = \frac{{}_{\mathrm{\rho}}S^{\rho}\mathrm{c}}{\rho}\left(\frac{Ndx}{dV}\right) \tag{2.42} $$

If we recall our memory, the quantity in the brackets is the fluence ϕ, therefore:

$$\frac{dE}{dm} = \frac{{}_{\rho}S^{\rho}\mathrm{c}}{\rho}\phi \tag{2.43}$$

As mentioned earlier, CEMA is used instead of KERMA for directly ionizing radiation.

The secondary charged particles produce due to interactions of directly ionizing radiation in the medium are often called delta-rays (δ-rays). The δ-rays equilibrium is the situation in which any charged particle kinetic energy leaving the volume of interest is replaced by an exactly equal amount entering the volume and being deposited in it or imparted to it.

Like KERMA, CEMA is also not necessarily equal to absorbed dose, as some of the secondary charged particles (i.e. δ-rays) may leave the volume of interest. The analogy of absorbed dose for directly ionizing radiation is the same as for indirectly ionizing radiation. The difference is in only the type of interaction with the medium and may be the generated secondary particles. Now, to calculate absorbed dose for directly ionizing radiation, it must be assumed that there is δ-ray equilibrium in order to be able to equate CEMA with absorbed dose. The dose for mono-energetic directly ionizing radiation is given as:

$$D \overset{\delta-\mathrm{eqm}}{=} \left(\frac{S_{\mathrm{c}}}{\rho}\right)\phi \tag{2.44}$$

Similarly, in the case of a spectrum of incident directly ionizing radiation:

$$D \overset{\delta-\mathrm{eqm}}{=} \int_{0}^{E_{\max}} \phi_{\mathrm{E}}\left(\frac{S_{\mathrm{c}}(E)}{\rho}\right)dE \tag{2.45}$$

2.12.2 *Fluence and Dose Relationship for Electrons*

The absorbed dose D in a medium 'w' can be related to the electron fluence (Φ) by Eq. (2.46), provided the following two conditions met:

(a) Radiative photons escape the volume of interest and
(b) Secondary electrons are locally absorbed or there exists a charged particle equilibrium (CPE) of secondary electrons.

$$D_{\mathrm{w}} = \Phi\left[\left(\frac{dE}{\rho dx}\right)_{\mathrm{c,w}}\right]_{\mathrm{E}} \tag{2.46}$$

As expected, absorbed dose is proportional to charge particle fluence and collisional mass stopping power in the medium.

2.13 Cavity Theory

Radiation detectors of various types are used for absorbed dose measurements in a medium. The composition of the dose sensitive part of dosimeter (cavity in case of ionization chamber) is different from the medium in which it is inserted. The signal from the detector can be related to the absorbed in the medium using certain calibration and correction factors. Cavity theory in reality is the determination of a dose conversion factor '*f*' for a given detector medium '*g*' to the surrounding medium '*w*' containing the cavity (Eq. (2.47))[Nahum A (2007)]

$$f = \frac{D_{\mathrm{w}}}{D_{\mathrm{g}}} \tag{2.47}$$

where D_{g} and D_{w} are doses measured in the cavity and the medium of interest respectively.

2.13.1 Bragg–Gray Cavity Theory

The Bragg–Gray cavity theory was developed for the first time to determine the '*f*' factor (Eq. (2.48)) in an arbitrary radiation quality *Q*.

A detector cavity that do not disturb the electron fluence, when it is inserted into the medium, is known as Bragg–Gray (B-G) cavity [Turner JE (2007)]. Consider Fig. 2.9 where a uniform medium '*w*' contains a small cavity filled with a different medium '*g*'. The cavity dimension is such that the fluence of the charged particles (Φ) remains unperturbed with its presence in the medium '*w*'.

For absorbed dose in cavity medium '*g*' (cavity) Eq. (2.46) can be written as [Attix FH (2004)],

$$D_{\mathrm{g}} = \Phi\left[\left(\frac{dE}{\rho dx}\right)_{\mathrm{c,g}}\right]_{\mathrm{E}} \tag{2.48}$$

It is reasonable to assume that for the Brag-Gray cavity the fluence Φ remains the same with and without the introduction of the detector cavity in the medium. From Eqs. (2.46) and (2.48) the ratio of absorbed doses in '*g*' and '*w*' is [Attix FH (2004)]:

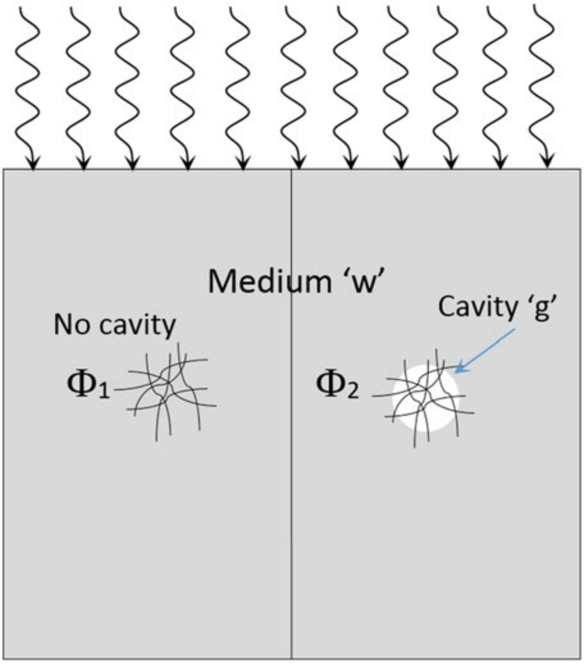

Fig. 2.9 Charge particles fluence is not affected by the insertion of detector cavity (B-G cavity) in the medium compared to the fluence exits on the left side with no cavity in place (i.e., $\Phi_1 = \Phi_2$)

$$\frac{D_\mathrm{w}}{D_\mathrm{g}} = \frac{\left(dE/\rho dx\right)_w}{\left(dE/\rho dx\right)_\mathrm{g}} \tag{2.49}$$

The subscript 'c' is not used in the above equation, since it is understood that local absorbed dose depends on collisional stopping power '${}_\rho S_\mathrm{c}$' only. The term on the right hand side of Eq. (2.49) is simply the ratio of mass collision stopping powers ${}_\mathrm{\rho}S^{\rho}\mathrm{g}^{\mathrm{w}}$. Hence it transforms into the following form [Attix FH (2004)].

$$\frac{D_\mathrm{w}}{D_\mathrm{g}} = {}_\mathrm{\rho}S^{\rho}\mathrm{g}^{\mathrm{w}} \tag{2.50}$$

The detector must fulfill a necessary condition if it is to be considered as Bragg–Gray cavity, that is "the introduction of the detector cavity in the medium must not perturb the charged particles fluence and energy distribution" [Nahum A (2007)].

A corollary that can be deduced from the first condition is:

The absorbed dose in the cavity is supposed to be deposited entirely by the charged particles crossing it.

Bragg–Gray cavity conditions are reasonably fulfilled for the majority of air-filled ionization chambers used in radiotherapy (megavoltage photon and electron beams) dosimetry [Turner JE (2007)].

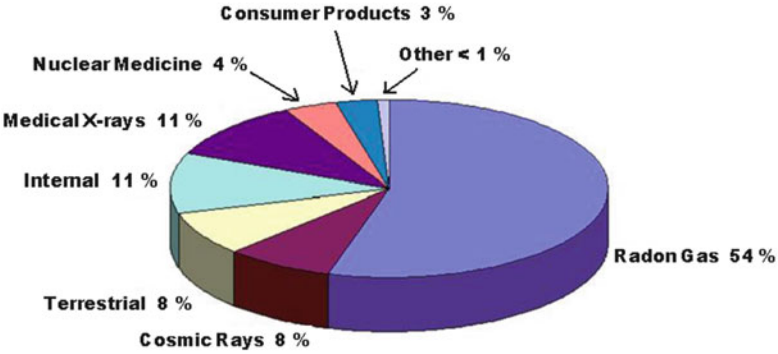

Fig. 2.10 Public exposure of radiation in the USA [Turner JE (2007)]

2.14 Biological Effects of Radiation

Radiations have been used widely in cancer treatment and other health issues. The successful outcomes and benefits of radiations have broadened the horizon of its use in a number of ways. Both directly and indirectly ionizing radiation have great impact on health care technology [Caon M, Bibbo G, Pattison J (1997)]. However, radiations can also be very dangerous to health if exposed in an unwanted and uncontrolled way. Human body can be affected in different ways when exposed to radiations. Despite of the NRC regulations to protect public from excessive expose to radiation still people expose to radiation coming out from various sources. Figure 2.10 shows public exposure to radiation in the USA. The figure shows that the highest contributor radon gas [Orabi M (2017)] and other natural sources of radiation are the major source or cause of radiation an average American is exposed to every year.

Human-made sources of radiation like medical x-rays, radiation during nuclear medicine diagnosis and treatment and other consumer products like watches, mobile phones and televisions are provide considerable amount of radiations to an average American each year. Though, the exposure from human made sources is less than the exposure from natural sources, but still big enough to produce damaging effects in the tissues.

Example 2.7 Assume that an average American person receives 40 rem of dose in 1 year. Find the dose he receives from radon gas, cosmic rays and consumer products?

Solution Total dose an average American receives per annum = 40 rem.

Figure 2.10 shows that 54% of this dose is received by exposure to radon gas. Therefore, the dose he receives from radon gas = 40 × 54/100 = 21.6 rem.

8% of the annual dose is given by cosmic rays. Therefore, the dose he receives from cosmic rays = 40 × 8/100 = 3.2 rem.

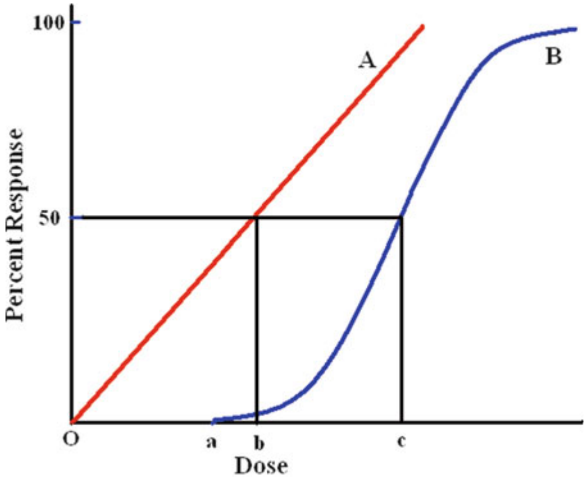

Fig. 2.11 Deterministic and stochastic effects, and tissues response to given dose

The dose he gets when exposed to radiation emitted by consumer products is 3%. Therefore, the dose he receives from consumer products = 40 × 3/100 = 1.2 rem.

All radiations have various effects on body tissues. Based on the quantity of dose and the body or exposed tissues response the biological effects of radiations can be classified into two major categories; Deterministic Effects and Stochastic Effects. Response of tissues to a given dose and the occurrence and chances of both kinds of effects are given in Fig. 2.11. Both effects are discussed below.

(A) *Stochastic Effects*. Stochastic effects are statistical effects for which there is no minimum dose required. Curve A in Fig. 2.11 shows a response curve representing stochastic effects. It is a probabilistic effect or a statistical phenomenon. It is possible that an effect can start in the body tissues when exposed to a very low dose. On the other hand a person may not get affected even exposed to a higher dose, though those chances are very low. The curve shows that the tissue response is linearly proportional to the quantity of radiation dose tissues are exposed to. That means, despite of its probabilistic nature, the chances of stochastic effects in the exposed tissues increase with increasing dose. Stochastic effects are also referred as late effects because these effects do not appear immediately after tissues are being exposed to radiation. Examples of such late effects or stochastic effects are cancer and hereditary effects in a part or whole body exposed to radiation. The legend scientist Marie Curie died of Aplastic Anemia, brought on by exposure to radiation. Overall, we can summarize the characteristics of stochastic effects in the following points.

- They have no threshold dose.
- They increase in likelihood as dose increases.
- Their severity is not dose related.
- There is no dose above which stochastic effects are certain to occur.

Table 2.4 Threshold doses for some deterministic effects in case of radiation exposure for many years

Dose in gray	Effect on tissues (when exposed for many years)
0.1 Gy	Detectible opacities
0.2 Gy	Sterility for woman
0.4 Gy	Visual impairment
0.4 Gy	Temporary sterility for man
0.4 Gy	Depression of haematopoiesis
1.0 Gy	Chronic radiation syndrome
2.0 Gy	Permanent sterility for man
3.0 Gy	Erythema

(B) *Deterministic effects*. Deterministic effects or non-stochastic effects are those kinds of effects that need a minimum dose or a threshold dose to start occurring. Depending upon the nature of the effect, the threshold dose could be little or more. Curve B in Fig. 2.11 shows a response curve representing deterministic effects. No effect can be observed if tissues or body is exposed to a dose smaller than the threshold dose. Example of such kind of effects is burning of the skin when exposed to radiation. During tanning process or sun bath, body is exposed to high energy ultraviolet radiations. If exposed to a high dose skin burning occurs which gets severe with more exposure. It is found that 200–300 rad (2–3 rem) to the skin can result in the reddening of the skin (erythema), similar to a mild sunburn that may result in hair loss due to damage to hair follicles. Similarly 125–200 rad (1.25–2 rem) to the ovaries can result in prolonged or permanent suppression of menstruation in about 50 percent (50%) of women, 600 rad (6 rem) to the ovaries or testicles can result in permanent sterilization and 50 rad to the thyroid gland can result in benign (non-cancerous) tumors [Cember and Johnson (2009)].

(C) The response of the tissues is low in the beginning when tissues are exposed to a dose higher than threshold dose however, increases very rapidly when exposed to high dose as shown in the figure. When the dose increases to a very high level affecting most of the tissues exposed the response decreases again. The reason being at such a high dose most of the tissues or cells are already affected and very few are left behind that will still affect. Some examples of deterministic effects are given in Table 2.4. Overall, we can summarize the characteristics of deterministic effects in the following points [Cember and Johnson (2009)].

- A threshold dose is required below which no effect is seen.
- They always occur once the threshold dose is reached.
- Worsening of the effect occurs as dose increases over the threshold.
- Different effects, tissues and people have different threshold doses for deterministic effects.
- These effects could be early or could appear late.

Table 2.5 Deterministic effects in the body with dose and time to onset

Exposure (rem)	Effect or outcome	Time to onset (without treatment)
5–10	Changes in blood counts	
50	Nausea	Hours
55	Fatigue	
70	Vomiting	
75	Hair loss	2–3 weeks
90	Diarrhea	
100	Hemorrhage	
400	Possible death	Within 2 months
1000	Destruction of intestinal lining	
	Internal bleeding	
	And death	1–2 weeks
2000	Damage to central nervous system	
	Loss of consciousness;	Minutes
	And death	Hours to days

Table 2.4 gives some of the effects on various tissues and organs when exposed to radiation for many years. In Table 2.5 the dose given to tissues or organs and its effect is shown along with the time in which the effect starts appearing.

Problems

1. A student, working on an experiment, exposed 12 kg of dry air to x-rays. As a result of ionization in the air, 3.612×10^{-3} C charge is produced. Determine the Exposure in the units of Roentgen.
2. A person received an average whole-body x-ray dose of 0.6 mGy and 0.8 mGy from 10-MeV neutrons. What is the whole-body equivalent dose in mSv?
3. A patient is recommended with 12 Gy absorbed dose to treat a tumor in his bladder. If this dose is delivered through (i) electrons (ii) neutrons of energy 5 MeV, and (iii) α-rays of energy 4 MeV, then find the effective dose received by the bladder of the patient in each case.
4. A female radiation worker receives the following equivalent doses to various organs when worked in a radiation area for 1 year. Bone Surface = 200 mSv, Breast = 150 mSv, Lungs = 40 mSv, and Liver =?

   Keeping the maximum allowed annual limit in mind, find the maximum dose received by lungs to keep her in the ALI limits? Do you think she is safe according to ICRP rules?
5. Calculate the allowable equivalent dose to the thyroid of a worker for a year in which he is exposed to non-uniform irradiation involving the whole body, the lungs and the thyroids. During the year he receives equivalent doses of 25 mSv to the whole body and 150 mSv to the lungs.

6. A person working in a grocery store in the US received an average dose of 80 rem in 1 year. Find the average contribution from radon gas and from medical x-rays.

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