6 Radiation Dosimetry

UNITS

During the early days of radiological experience, there was no precise unit of radiation dose that was suitable either for radiation protection or for radiation therapy. For purposes of radiation protection, a common "dosimeter" used was a piece of dental film with a paper clip attached. A daily exposure great enough just to produce a detectable shadow, called a "paper-clip" unit, was considered a maximum permissible dose. For greater doses and for therapy purposes, the dose unit was frequently the "skin erythema unit." Because of the great energy dependence of these dose units as well as other inherent shortcomings, neither of these two units could be biologically meaningful or useful either in the quantitative study of the biological effects of radiation or for radiation safety purposes. Furthermore, since the fraction of the energy in a radiation field that is absorbed by the body is energy dependent, it is necessary to distinguish between radiation *exposure* and radiation *absorbed dose*.

Absorbed Dose

Gray

Radiation damage depends on the absorption of energy from the radiation and is approximately proportional to the mean concentration of absorbed energy in irradiated tissue. For this reason, the basic unit of radiation dose is expressed in terms of absorbed energy per unit mass of tissue, that is,

$$\text{radiation absorbed dose} = \frac{\mathrm{d}E}{\mathrm{d}m}. \tag{6.1}$$

The unit for radiation absorbed dose in the SI system is called the *gray* (Gy) and is defined as follows:

One *gray* is an absorbed radiation dose of one joule per kilogram.

$$1\,\text{Gy} = 1\frac{\text{J}}{\text{kg}}. \tag{6.2}$$

The gray is universally applicable to all types of ionizing radiation dosimetry—irradiation due to external fields of gamma rays, neutrons, or charged particles as well as that due to internally deposited radionuclides.

rad (Radiation Absorbed Dose)

Before the introduction of the SI units, radiation dose was measured by a unit called the *rad*.

One *rad* is defined as an absorbed radiation dose of 100 $\frac{\text{ergs}}{\text{g}}$.

$$1\text{ rad} = 100\frac{\text{ergs}}{\text{g}}. \tag{6.3a}$$

Since 1 J = 10^7 ergs, and since 1 kg = 1000 g,

$$1\text{ Gy} = 100\text{ rads}. \tag{6.3b}$$

$$1\text{ rad} = 0.01\text{ Gy} = 1\text{ centigray (cGy)}. \tag{6.3c}$$

Although the gray is the SI and newer unit and has replaced the rad in literature and most countries, the rad and its derivatives nevertheless continue to be useful units and are used in the official radiation safety regulations in the United States.

It is important to understand that radiation absorbed dose concept, the gray and the rad, is a macroscopic construct and is not intended for microdosimetry on the cellular or subcellular levels. Radiation absorbed dose has been found to be correlated with biomedical effects on the tissue, organ, and organism levels and thus is appropriate for radiation safety measurements and for medical diagnostic and therapeutic uses of radiation. The radiation absorbed dose concept implies that the absorbed energy is uniformly distributed throughout the entire mass of the tissue of interest. On the cellular and subcellular levels that are of interest to molecular biologists, the biological effects are proportional to the number and types of intramolecular bonds that are broken rather than to the concentration of absorbed energy within the cell. On the tissue level, the number of such intramolecular breaks in the tissue is proportional to the radiation absorbed dose. The distinction between microdosimetry and radiation absorbed dose may be illustrated with the following thought experiment.

EXAMPLE 6-1

Consider a single cell, with dimensions 10 μm × 10 μm × 10 μm and mass = 10^{-12} kg, in a tissue of weight 0.1 g. A low LET particle, 1 keV/μm, transfers 10 keV to the tissue as it passes through. Calculate the radiation absorbed dose to the tissue.

Solution

The radiation absorbed dose to the tissue is calculated as

$$\text{radiation absorbed dose}\,(\text{tissue}) = \frac{10\ \text{keV}\cdot 1.6\times 10^{-16}\,\dfrac{\text{J}}{\text{keV}}\cdot 1\dfrac{\text{Gy}}{\text{J/kg}}}{1\times 10^{-4}\ \text{kg}} = 1.6\times 10^{-11}\,\text{Gy},$$

which is an infinitesimally small dose. However, if the absorbed dose were to be (erroneously) applied to the single cell, the calculated dose would be

$$\text{radiation absorbed dose}\,(\text{cell}) = \frac{10\ \text{keV}\cdot 1.6\times 10^{-16}\,\dfrac{\text{J}}{\text{keV}}\cdot 1\dfrac{\text{Gy}}{\text{J/kg}}}{1\times 10^{-12}\ \text{kg}} = 1.6\times 10^{-3}\,\text{Gy}.$$

EXTERNAL EXPOSURE

X- and Gamma Radiation

Exposure Unit

For external radiation of any given energy flux, the absorbed dose to any region within an organism depends on the type and energy of the radiation, the depth within the organism at the point at which the absorbed dose is required, and elementary constitution of the absorbing medium at this point. For example, bone, consisting of higher atomic-numbered elements (Ca and P) than soft tissue (C, O, H, and N), absorbs more energy from an X-ray beam per unit mass of absorber than soft tissue. For this reason, the X-ray fields to which an organism may be exposed are frequently specified in *exposure units*. The exposure unit is a *radiometric* unit rather than a dosimetric unit. That is, it is a measure of the photon fluence and is related to the amount of energy transferred from the X-ray field to a unit mass of *air*. If the amount of exposure, measured in exposure units, is known, then knowing the energy of the X-rays and the composition of the irradiated medium, we can calculate the absorbed dose to any part of the irradiated medium.

> One exposure unit is defined as that quantity of X- or gamma radiation that produces, in air, ions carrying one coulomb of charge (of either sign) per kilogram of air. It does not have a special name, and is being called an "X unit" in this textbook for convenience.

$$1\ \text{X unit} = 1\ \text{C/kg air}. \tag{6.4}$$

The exposure unit is based on ionization of air because of the relative ease with which radiation-induced ionization can be measured. At quantum energies less than several kilo

electron volts and more than several mega electron volts, it becomes difficult to fulfill the requirements for measuring the exposure unit. Accordingly, the use of the exposure unit is limited to X- or gamma rays whose quantum energies do not exceed 3 MeV. For higher energy photons, exposure is expressed in units of watt-seconds per square meter and exposure rate is expressed in units of watts per square meter. The operational definition of the exposure unit may be converted into the more fundamental units of energy absorption per unit mass of air by using the fact that the charge on a single ion is 1.6×10^{-19} C and that the average energy dissipated in the production of a single ion pair in air is 34 eV. Therefore,

$$1\text{ X unit} = 1\frac{\text{C}}{\text{kg}}\text{air} \cdot \frac{1\text{ ion}}{1.6\times10^{-19}\,\text{C}} \cdot 34\frac{\text{eV}}{\text{ion}} \cdot 1.6\times10^{-19}\,\frac{\text{J}}{\text{eV}} \cdot 1\frac{\text{Gy}}{\text{J/kg}}$$
$$= 34\text{ Gy}\,(\text{in air}). \tag{6.5}$$

It should be noted that the exposure unit is an integrated measure of exposure and is independent of the time over which the exposure occurs. The strength of an X-ray or gamma-ray field is usually expressed as an exposure rate, such as coulombs per kilogram per hour. The total exposure, of course, is the product of exposure rate and time.

Roentgen

Formerly, before the SI system was introduced, the unit of X-ray exposure was called the *roentgen* and was symbolized by R. The roentgen was defined as that quantity of X- or gamma radiation that produces ions carrying one statcoulomb (sC) (1 sC = 3.33564×10^{-10} C) of charge of either sign per cubic centimeter of dry air at 0°C and 760 mm Hg:

$$1\text{ R} = 1\,\frac{\text{sC}}{\text{cm}^3\ \text{dry air}}. \tag{6.6}$$

Since 1 ion carries a charge of 4.8×10^{-10} sC and the mass of 1 cm^3 of standard air is 0.001293 g, we can calculate the dose to the air from an exposure of 1 R:

$$1\text{ R} = 1\frac{\text{sC}}{\text{cm}^3\text{air}} \cdot \frac{1\text{ cm}^3\text{ air}}{1.293\times10^{-3}\,\text{g}} \cdot \frac{1\text{ ion}}{4.8\times10^{-10}\,\text{sC}} \cdot 34\frac{\text{eV}}{\text{ion}} \cdot 1.6\times10^{-12}\,\frac{\text{erg}}{\text{eV}} \cdot \frac{1\text{ rad}}{100\,\frac{\text{erg}}{\text{g}}}$$

1 R = 0.877 rad (to air).

When exposure is measured in roentgens, X-ray or gamma-ray field strength is measured in units such as roentgens per minute or milliroentgens per hour. A milliroentgen, which is symbolized by "mR," is equal to 0.001 R.

The relationship between the coulomb per kilogram exposure unit and the roentgen may be calculated as follows:

$$1\frac{\text{C}}{\text{kg}} = 1\frac{\text{C}}{\text{kg}} \cdot 1.293\,\frac{\text{kg}}{\text{m}^3} \cdot 1\times10^{-6}\,\frac{\text{m}^3}{\text{cm}^3} \cdot \frac{1\text{ sC}}{3.33564\times10^{-10}\,\text{C}} \cdot \frac{1\text{ R}}{1\,\frac{\text{sC}}{\text{cm}^3\ \text{air}}} = 3876\text{ R}$$

or

$$1\ \text{X unit} = 3876\ \text{R}. \tag{6.7a}$$

$$1\text{R} = \frac{1}{3876} \cdot \text{X unit} = 2.58 \times 10^{-4}\ \text{X unit}. \tag{6.7b}$$

EXAMPLE 6-2

Health physics measurements of X-ray and gamma-ray fields are usually made in units of milliroentgen per hour. If a health physicist finds a gamma-ray field of 1 mR/h, what is the corresponding exposure expressed in SI units?

Solution

According to Eq. (6.7b),

$$1\frac{\text{mR}}{\text{h}} \cdot 1 \times 10^{-3}\ \frac{\text{R}}{\text{mR}} = 2.58 \times 10^{-7}\ \frac{\text{C}}{\text{kg}}$$

$$1\ \frac{\text{mR}}{\text{h}} \cdot 1 \times 10^{-3}\ \frac{\text{R}}{\text{mR}} \cdot \frac{1\dfrac{\text{C}}{\text{kg}}}{3876\ \text{R}} = 0.258\ \frac{\mu\text{C}}{\text{h}}.$$

Exposure Measurement: The Free Air Chamber

The operational definition of the exposure unit can be satisfied by the instrument shown in Figure 6-1. The X-ray beam enters through the portal and interacts with the cylindrical column of air defined by the entry port diaphragm. All the ions resulting from interactions between the X-rays and the volume of air (A–B–C–D), which are determined by the intersection of the X-ray beam with the electric lines of force from the edges of the collector plate C, are collected by the plates, causing current to flow in the external circuit. Most of these collected ions are those produced as the primary ionizing particles that lose their energy by ionizing interactions as they pass through the air. (The primary ionizing particles are the Compton electrons and the photoelectrons resulting from the interaction of the X-rays (photons) with the air.) The guard ring G and the guard wires W help to keep these electric field lines straight and perpendicular to the plates. The electric field intensity between the plates is on the order of 100 V/cm—high enough to collect the ions before they recombine but not great enough to accelerate the secondary electrons produced by the primary ionizing particles to ionizing energy. The guard wires are connected to a voltage-dividing network to ensure a uniform potential drop across the plates. The number of ions collected because of X-ray interactions in the collecting volume is calculated from the current flow and then the exposure rate, in roentgens per unit time, is computed. For the exposure unit to be measured in this way, all the energy of the primary electrons must be dissipated in the

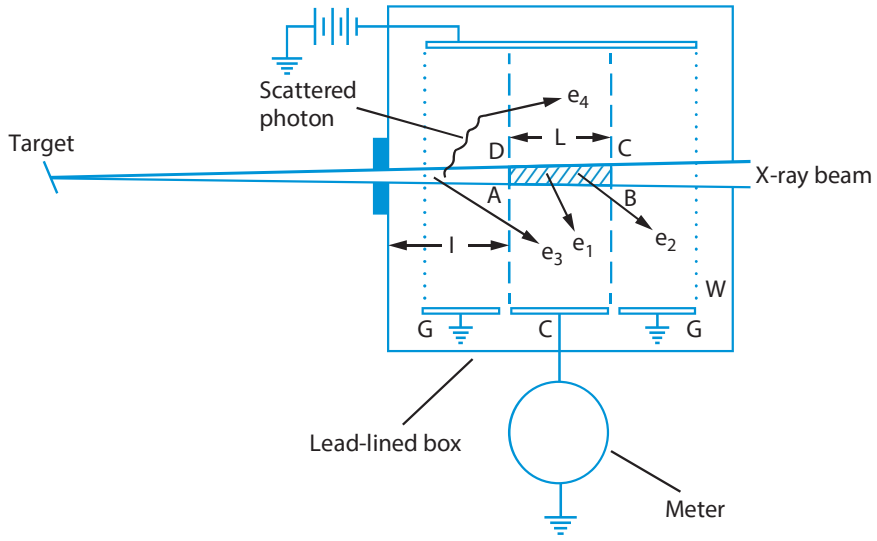

Figure 6-1. Schematic diagram of a parallel-plate ionization chamber. (From *Design of Free Air Ionization Chamber*. Washington, DC: National Bureau of Standards; 1957. NBS Handbook 64.)

air within the meter. This condition can be satisfied by making the air chamber larger than the maximum range of the primary electrons. (For 300-keV X-rays, the spacing between the collector plates is about 30 cm and the overall box is a cube of about 50-cm edge.) The fact that many of the ions produced as a consequence of X-ray interactions within the sensitive volume are not collected is of no significance if as many electrons from interactions elsewhere in the X-ray beam enter the sensitive volume as leave it. This condition is known as *electronic equilibrium,* or more generally, *charged particle equilibrium*. Charged particle equilibrium is defined for small volumes at a given location where *for every charged particle leaving the defined volume, another particle of the same type and kinetic energy enters the volume.*

When electronic equilibrium is attained, an electron of equal energy enters into the sensitive volume for every electron that leaves. A sufficient thickness of air, dimension l in Figure 6-1, must be allowed between the beam entrance port and the sensitive volume in order to attain electronic equilibrium. For highly filtered 250-kV X-rays, 9 cm of air is required; for 500-kV X-rays, the air thickness required for electronic equilibrium in the sensitive volume increases to 40 cm.

Under conditions of electronic equilibrium and assuming negligible attenuation of the X-ray beam by the air in length l, most of the ions collected from the sensitive volume result from primary photon interactions in air which occur at the beam entrance port; the measured exposure, consequently, is at that point and not in the sensitive volume. Free air chambers are in use that measure the quantity of X-rays whose quantum energies reach as high as 500 keV. Higher energy radiation necessitates free air chambers of much greater size. The technical problems arising from the use of such large chambers make it impractical to use the free air ionization chamber as a primary measuring device for quantum energies in excess of 500 keV. These problems include recombination of ions in the large chamber before they can be collected and secondary ionization due to acceleration of the initial ions by the great potential difference required for large chambers.

The use of the free air ionization chamber to measure X-ray exposure rate in coulombs per kilogram per second may be illustrated by the following example.

EXAMPLE 6-3

The opening of the diaphragm in the entrance port of a free air ionization chamber is 1 cm in diameter, and the length AB of the sensitive volume is 5 cm. A 200-kV X-ray beam projected into the chamber produces a steady current in the external circuit of 0.01 μA. The temperature at the time of the measurement was 20°C (293 K) and the pressure was 750 mm Hg. What is the exposure rate in this beam of X-rays?

Solution

A current of 0.01 μA corresponds to a flow of electrical charge of 10^{-8} C/s. The sensitive volume in this case, $\pi \cdot (0.5\ \text{cm})^2 \cdot 5\ \text{cm} = 3.927\ \text{cm}^3$. When the pressure and temperature are corrected to standard conditions, we have

$$\dot{X} = \frac{10^{-8}\ \frac{\text{C}}{\text{s}}}{3.927\ \text{cm}^3 \cdot 1.29 \times 10^{-6}\ \frac{\text{kg}}{\text{cm}^3}} \cdot \frac{293\ \text{K}}{273\ \text{K}} \cdot \frac{760\ \text{mm Hg}}{750\ \text{mm Hg}}$$

$$= 2.14 \times 10^{-3}\ \frac{\text{C/kg}}{\text{s}}.$$

In the traditional system of units, this exposure rate corresponds to

$$2.14 \times 10^{-3}\ \frac{\text{C/kg}}{\text{s}} \cdot \frac{\text{R}}{2.58 \times 10^{-4}\ \text{C/kg}} = 8.3\ \frac{\text{R}}{\text{s}}.$$

Exposure Measurement: The Air Wall Chamber

The free air ionization chamber described above is practical only as a primary laboratory standard. For field use, a more portable instrument is required. Such an instrument could be made by compressing the air around the measuring cavity. If this were done, then the conditions for defining the exposure unit would continue to be met. In practice, of course, it would be quite difficult to construct an instrument whose walls were made of compressed air. However, it is possible to make an instrument with walls of "air-equivalent" material—that is, a wall material whose X-ray absorption properties are very similar to those of air. Such a chamber can be built in the form of an electrical capacitor; its principle of operation can be explained with the aid of Figure 6-2.

The instrument consists of an outer cylindrical wall, about 4.75-mm thick, made of electrically conducting plastic. Coaxial with the outer wall, but separated from it by a very high-quality insulator, is a center wire. This center wire, or central anode, is positively charged with respect to the wall. When the chamber is exposed to X-radiation or to gamma

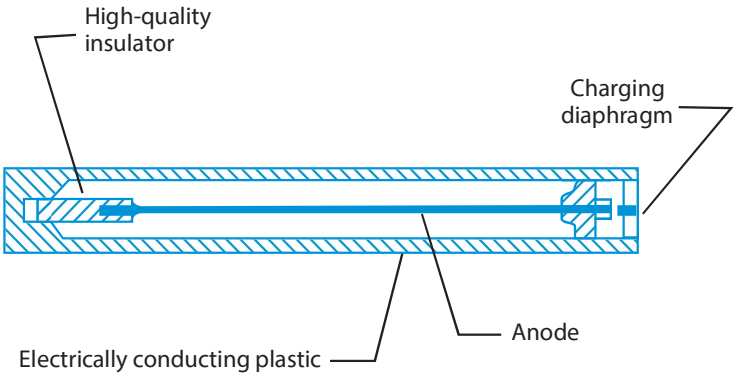

Figure 6-2. Non-self-reading condenser-type ionization chamber.

radiation, the ionization, which is produced in the measuring cavity as a result of interactions between photons and the wall, discharges the condenser, thereby decreasing the potential of the anode. This decrease in the anode voltage is directly proportional to the ionization produced in the cavity, which in turn is directly proportional to the radiation exposure. For example, consider the following instance:

EXAMPLE 6-4

Given

chamber volume = 2 cm^3,
chamber is filled with air at STP,
capacitance = 5 pF,
voltage across chamber before exposure to radiation = 180 V,
voltage across chamber after exposure to radiation = 160 V, and
exposure time = 0.5 h.

Calculate the radiation exposure and the exposure rate.

Solution

The exposure is calculated as follows:

$$C \cdot \Delta V = \Delta Q \tag{6.8}$$

where

C = capacitance, farads,
V = potential, volts, and
Q = charge, coulombs

$$5 \times 10^{-12}\,\text{F} \cdot (180 - 160)\,\text{V} = 1 \times 10^{-10}\,\text{C}.$$

Since one exposure unit is equal to 1 C/kg, using a density of air of 1.29×10^{-6} kg·cm^{-3} the exposure measured by this chamber is

$$\frac{1\times10^{-10}\,\text{C}}{2\,\text{cm}^3\cdot1.29\times10^{-6}\,\frac{\text{kg}}{\text{cm}^3}}=3.87\times10^{-5}\,\frac{\text{C}}{\text{kg}},$$

which corresponds to

$$3.87\times10^{-5}\,\frac{\text{C}}{\text{kg}}\cdot3876\,\frac{\text{R}}{\text{C/kg}}=0.150\ \text{R}=150\ \text{mR},$$

and the exposure rate was

$$\frac{3.87\times10^{-5}\,\text{C/kg}}{0.5\,\text{h}}=7.7\times10^{-5}\,\frac{\text{C/kg}}{\text{h}}=77\,\frac{\mu\text{C/kg}}{\text{h}}$$

or $300\,\frac{\text{mR}}{\text{h}}$.

A chamber built according to this principle is called an "air wall" chamber. When such a chamber is used, care must be taken that the walls are of the proper thickness for the energy of the radiation being measured. If the walls are too thin, an insufficient number of photons will interact to produce primary electrons; if they are too thick, the primary radiation will be absorbed to a significant degree by the wall and an attenuated primary-electron fluence will result.

The determination of the optimum thickness may be illustrated by an experiment in which the ionization produced in the cavity of an ionization chamber is measured as the wall thickness is increased from a very thin wall until it becomes relatively thick. When this is done and the cavity ionization is plotted against the wall thickness, the curve shown in Figure 6-3 results.

Since the cavity ionization is caused mainly by primary electrons resulting from photon (gamma-ray) interactions with the wall, increasing the wall thickness allows more photons to interact, thereby producing more primary electrons, which ionize the gas in the chamber as they traverse the cavity. However, when the wall thickness reaches a point where a primary electron produced at the outer surface of the wall is not sufficiently energetic to pass through the wall into the cavity, the ionization in the cavity begins to decrease. The wall thickness at which this just begins is the *equilibrium wall thickness*.

As the wall material departs from air equivalence, the response of the ionization chamber becomes energy dependent. By proper choice of wall material and thickness, the maximum in the curve of Figure 6-4 can be made quite broad, and the ionization chamber, as a consequence, can be made relatively energy independent over a wide range of quantum energies. In practice, this approximately flat response spans the energy range from about 200 keV to about 2 MeV. In this range of energies, the Compton effect is the predominant mechanism

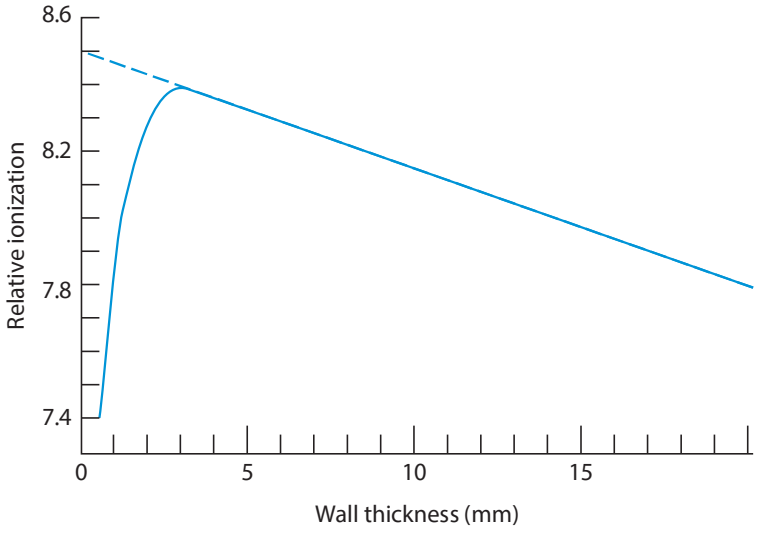

Figure 6-3. Ion pairs per unit volume as a function of wall thickness. The ionization chamber in this case was made of pure carbon and was a 20-mm-long cylinder with an inside diameter of 20 mm. (Reproduced with permission from Mayneord WV, and Roberts JE. An attempt at precision measurements of gamma rays. *Br J Radiol*. 1937;10(113):365–386. Permission conveyed through Copyright Clearance Center, Inc.)

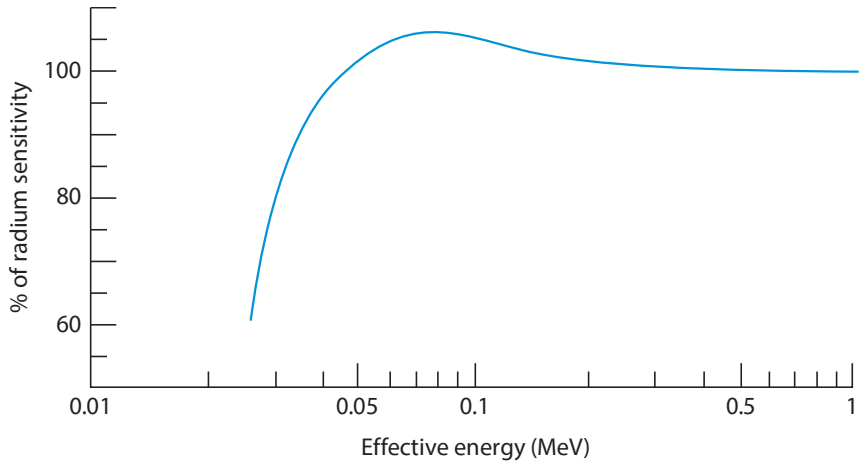

Figure 6-4. Energy dependence characteristics of the pocket dosimeter shown in Figure 9-20.

of energy transfer. For lower energies, the probability of a Compton interaction increases approximately in direct proportion to the wavelength, while the probability of a photoelectric interaction is approximately proportional to the cube of the photon's wavelength. As photon energy decreases, the total number of primary electrons produced increases and therefore the sensitivity of the chamber also increases; the increased sensitivity, however, reaches a peak as the quantum energy decreases and then, because of the severe attenuation of the incident radiation by the chamber wall, the sensitivity rapidly decreases. These effects are shown in Figure 6-4, a curve showing the energy correction factor for a pocket dosimeter.

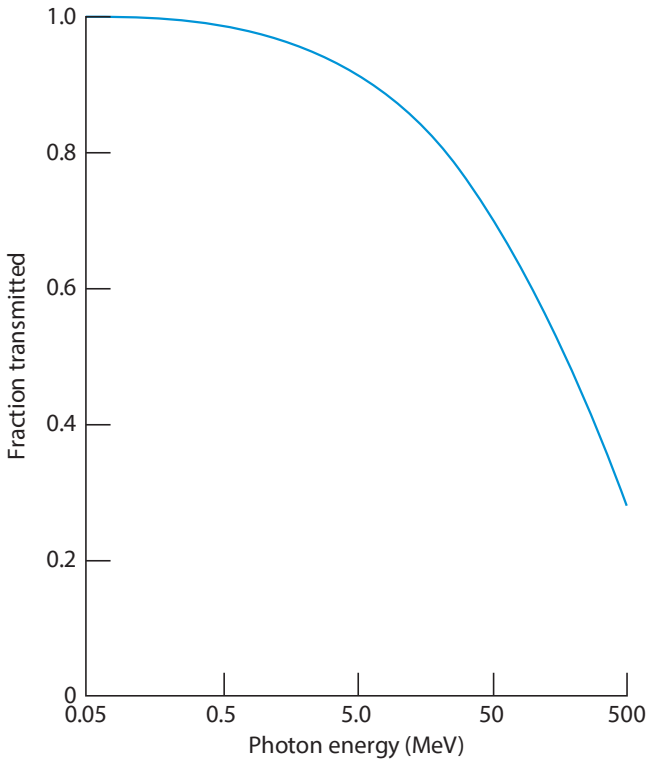

Figure 6-5. Fractional number of photons transmitted through an air wall of thickness equal to the maximum range of the primary electrons. (From *Protection Against Betatron–Synchrotron Radiations up to 100 Million Electron Volts*. Washington, DC: US Government Printing Office; 1955. NBS Handbook 55.)

For quantum energies greater than 3 MeV, neither coulombs per kilogram nor roentgen is used as the unit of measurement of exposure. This is due to the fact that the high energy, and consequently the long range of the primary electrons produced in the detector wall, makes it impossible to build an instrument that meets the criteria for measuring the exposure. Because of the long range of the primary electrons, very thick walls are necessary. However, when the walls are sufficiently thick, on the basis of the range of the primary electrons, they attenuate the photons (gamma radiation) to a significant degree, as shown in Figure 6-5. Under these conditions, it is not possible to attain electronic equilibrium since the radiation intensity within the wall is not constant and the primary electrons, consequently, are not produced uniformly throughout the entire volume of wall from which they may reach the cavity.

Exposure–Dose Relationship

The air wall chamber, as the name implies, measures the energy absorption in air. In most instances, we are interested in the energy absorbed in tissue. Since energy absorption is approximately proportional to the electronic density of the absorber in the energy region where exposure units are valid, it can be shown that the tissue dose is not necessarily equal

to the air dose for any given radiation field. For example, if we consider muscle tissue to have a specific gravity of 1 and to have an elementary composition of 5.98×10^{22} hydrogen atoms per gram, 2.75×10^{22} oxygen atoms per gram, 1.72×10^{21} nitrogen atoms per gram, and 6.02×10^{21} carbon atoms per gram, then the electronic density is 3.28×10^{23} electrons per gram. For air, whose density is 1.29×10^{-3} g/cm^3, the electronic density is 3.01×10^{23} electrons per gram. The energy absorption, in joules per kilogram of tissue, corresponding to an exposure of 1 C/kg air is, therefore,

$$\frac{3.28}{3.01} \cdot 34 \frac{\text{J}}{\text{kg}} \text{air} = 37 \frac{\text{J}}{\text{kg}} \text{tissue}.$$

This value agrees very well with calorimetric measurements of energy absorption by soft tissue from an exposure of 1 C/kg air. By analogy, **an exposure of 1 R, which corresponds to 87.7 ergs/g of air, leads to an absorption of 96 ergs/g muscle tissue**. This tissue dose from a 1-R exposure is very close to the tissue dose of 100 ergs/g, which corresponds to 1 rad. For this reason, an exposure of 1 R is frequently considered approximately equivalent to an absorbed dose of 1 rad, and the unit "roentgen" is loosely (but incorrectly) used to mean "rad." Because of this simple approximate one-to-one relationship of the roentgen to the rad, the roentgen continues to be used. To be up to date vis-à-vis measurement units, an exposure of 1 R is often called a dose of 1 centigray (cGy).

The exposure unit bears a simple quantitative relationship to the dosimetric unit (the gray or the rad) that permits the calculation of absorbed dose in any medium whose exposure (in coulombs per kilogram or statcoulombs per cubic centimeter air) is known. This relationship may be illustrated by the following example.

EXAMPLE 6-5

Consider a gamma-ray beam of quantum energy 0.3 MeV. If the photon flux is 1000 quanta/cm^2/s and the air temperature is 20°C, what is the exposure rate at a point in this beam and what is the absorbed dose rate for soft tissue at this point?

Solution

From Figure 5-20, the linear energy absorption coefficient for air, μ_{air}, at STP, for 300-keV photons is found to be 3.7×10^{-5} cm^{-1}. The exposure rate, $\dot{X}$, in C/kg/s in a photon flux ϕ is given by

$$\dot{X} = \frac{\phi \dfrac{\text{photons}}{\text{cm}^2 \cdot \text{s}} \cdot E \dfrac{\text{MeV}}{\text{photon}} \cdot 1.6 \times 10^{-13} \dfrac{\text{J}}{\text{MeV}} \cdot \mu_{\text{air}} \text{cm}^{-1}}{\rho_{\text{air}} \dfrac{\text{kg}}{\text{cm}^3} \cdot 34 \dfrac{\text{J/kg}}{\text{C/kg}}}, \tag{6.9a}$$

where

μ_{air} = linear energy absorption coefficient for air for the photon energy and
ρ_{air} = density of air.

The radiation absorbed dose rate from this exposure is given by

$$\dot{D} = \frac{\phi \dfrac{\text{photons}}{\text{cm}^2 \cdot \text{s}} \cdot E \dfrac{\text{MeV}}{\text{photon}} \cdot 1.6 \times 10^{-13} \dfrac{\text{J}}{\text{MeV}} \cdot \mu_m \text{cm}^{-1}}{\rho_m \dfrac{\text{kg}}{\text{cm}^3} \cdot 1 \dfrac{\text{J/kg}}{\text{Gy}}}, \tag{6.9b}$$

where

μ_m = linear energy absorption coefficient of the medium and
ρ_m = density of the medium.

Substituting the appropriate numerical values into Eq. (6.9a), we have

$$\dot{X} = \frac{10^3 \dfrac{\text{photons}}{\text{cm}^2 \cdot \text{s}} \cdot 0.3 \text{ MeV} \cdot 1.6 \times 10^{-13} \dfrac{\text{J}}{\text{MeV}} \cdot 3.7 \times 10^{-5} \text{cm}^{-1}}{\left(1.29 \times 10^{-6} \dfrac{\text{kg}}{\text{cm}^3} \cdot \dfrac{273 \text{ K}}{293 \text{ K}}\right) \cdot 34 \dfrac{\text{J/kg}}{\text{C/kg}}}$$

$$\dot{X} = 4.3 \times 10^{-11} \frac{\text{C/kg}}{\text{s}}.$$

Since health physics measurements are usually given in units of per hour, this exposure rate corresponds to

$$4.3 \times 10^{-11} \frac{\text{C/kg}}{\text{s}} \cdot 3.6 \times 10^3 \frac{\text{s}}{\text{h}} = 1.6 \times 10^{-7} \frac{\text{C/kg}}{\text{h}}$$
$$= 0.16 \frac{\mu\text{C/kg}}{\text{h}}.$$

Since $0.258 \dfrac{\mu\text{C}}{\text{kg}} = 1\text{mR}$, the exposure rate expressed in traditional units is

$$\dot{R} = 0.16 \frac{\mu\text{C/kg}}{\text{h}} \cdot \frac{1 \text{ mR}}{0.258 \dfrac{\mu\text{C}}{\text{kg}}} = 0.6 \frac{\text{mR}}{\text{h}}.$$

The absorbed dose rate, in grays per second, is given by Eq. (6.9b) as

$$\dot{D} = \frac{\phi \dfrac{\text{photons}}{\text{cm}^2 \cdot \text{s}} \cdot E \dfrac{\text{MeV}}{\text{photon}} \cdot 1.6 \times 10^{-13} \dfrac{\text{J}}{\text{MeV}} \cdot \mu_m \text{ cm}^{-1}}{\rho_m \dfrac{\text{kg}}{\text{cm}^3} \cdot 1 \dfrac{\text{J/kg}}{\text{Gy}}}.$$

The ratio of absorbed dose rate to the exposure dose rate is given by the ratio of Eq. (6.9b) to Eq. (6.9a):

$$\frac{\dot{D}}{\dot{X}} = \frac{\left(\dfrac{\phi \cdot E \cdot 1.6\times10^{-13} \cdot \mu_{\mathrm{m}}}{\rho_{\mathrm{m}}}\right)}{\left[\dfrac{\left(\phi \cdot E \cdot 1.6\times10^{-13} \cdot \mu_{\mathrm{air}}\right)}{\rho_{\mathrm{air}} \cdot 34\dfrac{\mathrm{J/kg}}{\mathrm{C/kg}}}\right]}.$$

The absorbed dose rate, in Gy per unit time, resulting from an exposure of $\dot{X}$ C/kg per unit time, therefore is

$$\dot{D}, \frac{\mathrm{Gy}}{\mathrm{time}} = 34\frac{\mathrm{Gy}}{\dfrac{\mathrm{C}}{\mathrm{kg}}} \cdot \frac{\left(\dfrac{\mu_{\mathrm{m}}\,\mathrm{cm}^{-1}}{\rho_{\mathrm{m}}\dfrac{\mathrm{g}}{\mathrm{cm}^3}}\right)}{\left(\dfrac{\mu_{\mathrm{air}}\,\mathrm{cm}^{-1}}{\rho_{\mathrm{air}}\dfrac{\mathrm{g}}{\mathrm{cm}^3}}\right)} \cdot \dot{X}\frac{\mathrm{C/kg}}{\mathrm{time}}. \tag{6.10}$$

Since the mass absorption coefficient is given by Eq. (5.28) as

$$\mu_{\mathrm{mass}} = \frac{\mu_{\mathrm{linear}}}{\rho},$$

Eq. (6.10) may be written as

$$\dot{D}, \mathrm{Gy} = 34\frac{\mathrm{Gy}}{\mathrm{C/kg}} \cdot \left(\frac{\mu_{\mathrm{m}}}{\mu_{\mathrm{air}}}\right)_{\mathrm{mass}} \cdot \dot{X}\frac{\mathrm{C/kg}}{\mathrm{time}}. \tag{6.11}$$

Equation (6.11) is also applicable to the calculation of the absorbed dose if the exposure dose is used instead of the exposure dose rate.

To obtain a dose in rads to any medium when the exposure is given in roentgens, we use the analogous expression

$$\mathrm{rads} = \frac{87.7}{100} \cdot \left(\frac{\mu_{\mathrm{m}}}{\mu_{\mathrm{air}}}\right)_{\mathrm{mass}} \cdot \mathrm{roentgens}. \tag{6.12}$$

EXAMPLE 6-6

What is the radiation absorbed dose corresponding to an exposure dose of 25.8 μC/kg (100 mR) from 300-keV photons?

Solution

When the values for the energy absorption coefficient for muscle tissue for 0.3-MeV photons, $\mu_m = 0.03164\ \text{cm}^2/\text{g}$, and $\mu_{air} = 0.02872\ \text{cm}^2/\text{g}$, from Table 5-4, are substituted into Eq. (6.11), we have

$$\text{Dose} = 34\frac{\text{Gy}}{\text{C/kg}} \cdot \frac{0.03164\ \frac{\text{cm}^2}{\text{g}}}{0.02872\ \frac{\text{cm}^2}{\text{g}}} \cdot 25.8\times10^{-6}\ \frac{\text{C}}{\text{kg}} = 9.7\times10^{-4}\,\text{Gy}$$

$$= 0.97\ \text{mGy} = 97\ \text{mrads}.$$

Equations (6.11) and (6.12) show that the radiation dose absorbed from any given exposure is determined by the ratio of the mass absorption coefficient of the medium to that of air. In the case of tissue, the ratio of dose to exposure remains approximately constant over the quantum energy range of about 0.1 to 10 MeV because the chief means of interaction between the tissue and the radiation is Compton scattering, and the cross section for Compton scattering depends mainly on electronic density of the absorbing medium. In the case of lower energies, photoelectric absorption becomes important, and the cross section for this mode of interaction increases with atomic number of the absorber. As a consequence of this dependence on atomic number, bone, which contains approximately 10% by weight of calcium, absorbs much more energy than soft tissue from a given air dose of low-energy X-rays. This point is illustrated in Figure 6-6, which shows the ratio of energy absorption of tissues (muscle, bone, and adipose) to air as a function of photon energy.

Absorbed Dose Measurement: Bragg–Gray Principle

If a cavity ionization chamber is built with a wall material whose radiation absorption properties are similar to those of tissue, then, by taking advantage of the *Bragg–Gray principle*, an instrument can be built to measure tissue dose directly. According to the Bragg–Gray principle, the amount of ionization produced in a small gas-filled cavity surrounded by a solid absorbing medium is proportional to the energy absorbed by the solid. Implicit in the practical application of this principle is that the gas cavity be small enough relative to the mass of the solid absorber to leave the angular and velocity distributions of the primary electrons unchanged. This requirement is fulfilled if the primary electrons lose only a very small fraction of their energy in traversing the gas-filled cavity. If the cavity is surrounded

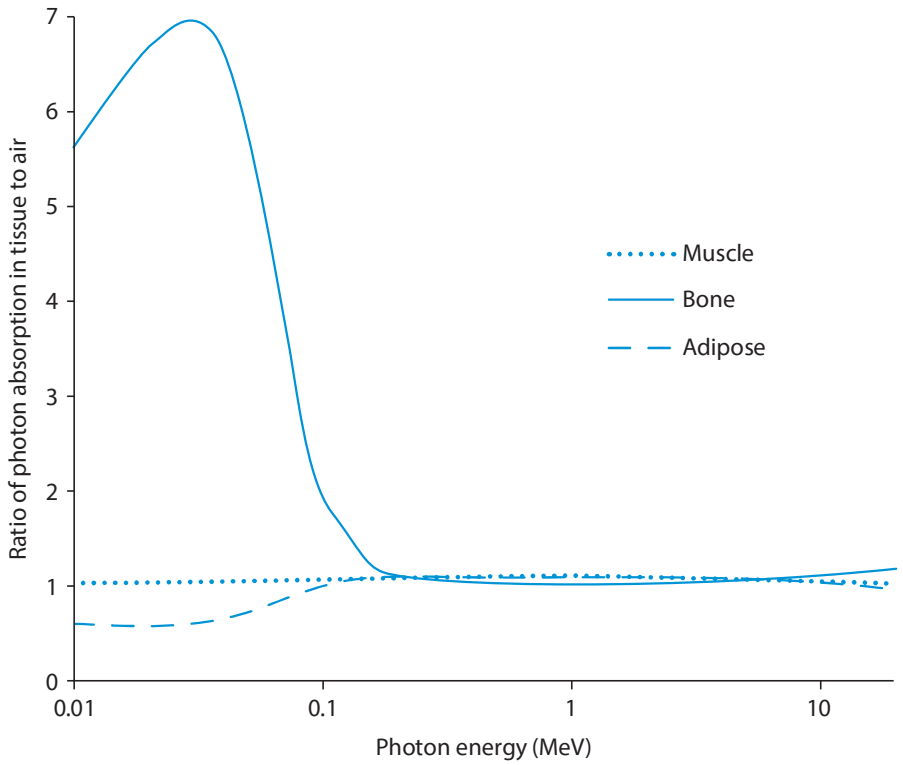

Figure 6-6. Energy absorption per X unit (coulomb per kilogram) exposure for several tissues. (Based on data from *Tables of X-Ray Mass Attenuation Coefficients and Mass Energy-Absorption Coefficients from 1 keV to 20 MeV for Elements Z = 1 to 92 and 48 Additional Substances of Dosimetric Interest*, J. H. Hubbell and S. M. Seltzer, Radiation Physics Division, PML, NIST, 1996. Accessed June, 2016.)

by a solid medium of proper thickness to establish electronic equilibrium, then the energy absorbed per unit mass of wall, dE_m/dM_m, is related to the energy absorbed per unit mass of gas in the cavity, dE_g/dM_g, by

$$\frac{dE_m}{dM_m} = \frac{S_m}{S_g} \cdot \frac{dE_g}{dM_g}, \tag{6.13}$$

where

S_m = mass stopping power of the wall material and
S_g = mass stopping power of the gas.

Since the ionization per unit mass of gas is a direct measure of dE_g/dM_g, Eq. (6.13) can be rewritten as

$$\frac{dE_m}{dM_m} = \rho_m \cdot \omega \cdot J, \tag{6.14}$$

where

$$\rho_m = \frac{S_m}{S_g},$$

ω = the mean energy dissipated in the production of an ion pair in the gas and
J = the number of ion pairs per unit mass of gas.

Using the appropriate equations for stopping power given in Chapter 5, we can compute ρ_m for electrons of any given energy. For those cases where the gas in the cavity is the same substance as the chamber wall, such as methane and paraffin, ρ_m is equal to unity. Appendix G shows the stopping power of several substances for monoenergetic electrons. For gamma radiation, however, the problem of evaluating ρ_m is more difficult. The relative fraction of the gamma rays that will interact by each of the competing mechanisms, as well as the spectral distribution of the primary electrons (Compton, photoelectric, and pair-produced electrons), must be considered, and a mean value for relative stopping power must be determined. For air, ω, the mean energy loss for the production of an ion pair in air has a value of 34 eV. To determine the radiation absorbed dose, it is necessary only to measure the ionization J per unit mass of gas.

EXAMPLE 6-7

Calculate the absorbed dose rate from the following data on a tissue-equivalent chamber with walls of equilibrium thickness embedded within a phantom and exposed to ^{60}Co gamma rays for 10 minutes. The volume of the air cavity in the chamber is 1 cm^3, the capacitance is 5 pF, and the gamma-ray exposure results in a decrease of 72 V across the chamber.

Solution

The charge collected by the chamber is

$$\begin{aligned} Q &= C \cdot \Delta V \\ &= 5 \times 10^{-12}\ \text{F} \cdot 72\ \text{V} \\ &= 3.6 \times 10^{-10}\ \text{C}. \end{aligned}$$

The number of electrons collected, which corresponds to the number of ion pairs formed in the air cavity, is

$$\frac{3.6 \times 10^{-10}\ \text{C}}{1.6 \times 10^{-19}\ \dfrac{\text{C}}{\text{electron}}} = 2.25 \times 10^{9}\ \text{electrons}.$$

Since 34 eV are expended per ion pair formed in air and since the stopping power of tissue relative to air for Co-60 is (from Appendix G)

$$\rho_{\text{tissue}} = \frac{S_{\text{tissue}}}{S_{\text{air}}} = \frac{1.83\ \dfrac{\text{cm}^2}{\text{g}}}{1.67\ \dfrac{\text{cm}^2}{\text{g}}} = 1.1,$$

we have from the Bragg–Gray relationship of Eq. (6.14):

$$\frac{\mathrm{d}E_{\mathrm{m}}}{\mathrm{d}M_{\mathrm{m}}} = \rho_{\mathrm{m}} \cdot \omega \cdot J$$

$$= \frac{1.1 \cdot 34 \ \frac{\mathrm{eV}}{\mathrm{ip}} \cdot 2.25 \times 10^{9} \frac{\mathrm{ip}}{\mathrm{cm}^3} \cdot 1.6 \times 10^{-19} \frac{\mathrm{J}}{\mathrm{eV}}}{1.29 \times 10^{-6} \frac{\mathrm{kg}}{\mathrm{cm}^3} \cdot 1 \frac{\mathrm{J/kg}}{\mathrm{Gy}}}$$

$$= 0.0104 \ \mathrm{Gy} = 10.4 \ \mathrm{mGy} \ (1.04 \ \mathrm{rad}).$$

The exposure time was 10 minutes and the dose rate therefore is 1.04 mGy/min (104 mrad/min).

Kerma

In the case of indirectly ionizing radiation, such as X-rays, gamma rays, and fast neutrons, we are sometimes interested in the initial kinetic energy of the primary ionizing particles (the photoelectrons, Compton electrons, or positron–negatron pairs in the case of photon radiation and the scattered nuclei in the case of fast neutrons) that result from the interaction of the incident radiation with a unit mass of interacting medium. This quantity of transferred energy is called the *kerma, K,* and is measured in SI units in joules per kilogram, or grays. In the traditional system of units, it is measured in ergs per gram or in rads.

The term *kerma* is defined as "…the mean sum of the initial kinetic energies of all the charged particles liberated in a mass d*m* of a material by the uncharged particles incident on d*m*."[1] Although kerma and dose are both measured in the same units, they are different quantities. The kerma is a measure of all the energy transferred from the uncharged particle (photon or neutron) to primary ionizing particles per unit mass, whereas absorbed dose is a measure of the energy absorbed per unit mass.

Not all the energy transferred to the primary ionizing particles in a given volume of material may be absorbed in that volume. Some of this energy may leave that volume and be absorbed elsewhere. This could result from bremsstrahlung or annihilation radiation which is generated by the primary ionizing particles, but which leaves the volume element without further interactions within that volume. It may also be the result of failure to attain electronic equilibrium within the volume element under consideration. In a large medium, where electronic equilibrium exists and where we have insignificant energy loss by bremsstrahlung, kerma is equal to absorbed dose. The difference between kerma and dose is illustrated by Example 6.8.

[1]ICRU 85, Fundamental Quantities and Units for Ionizing Radiation, Journal of the ICRU Volume 11 No 1, Bethesda, MD 2011.

EXAMPLE 6-8

A 10-MeV photon penetrates into a 100-g mass and undergoes a single interaction, a pair-production interaction, that leads to a positron and an electron of 4.5 MeV each. Both charged particles dissipate all their kinetic energy within the mass through ionization and bremsstrahlung production. Three bremsstrahlung photons of 1.6, 1.4, and 2 MeV each that are produced escape from the mass before they interact. The positron, after expending all its kinetic energy, interacts with an ambient electron within the mass and they mutually annihilate one another to produce two photons of 0.51 MeV each, and both these photons escape before they can interact within the mass. Calculate

(a) the kerma and
(b) the absorbed dose.

Solution

(a) Kerma is defined as the *sum of the initial kinetic energies per unit mass of all charged particles produced by the radiation*. In this case, a positron–negatron pair of 4.5 MeV each (2×4.5 MeV) represents all the initial kinetic energy. The kerma, K, in this case is

$$K = \frac{\text{kinetic energy released}}{\text{mass}} = \frac{2 \cdot (4.5\ \text{MeV}) \cdot 1.6 \times 10^{-13}\ \frac{\text{J}}{\text{MeV}}}{0.1\ \text{kg} \cdot 1\ \frac{\text{J/kg}}{\text{Gy}}}$$
$$= 1.44 \times 10^{-11}\ \text{Gy}.$$

(b) Dose is defined as the *energy absorbed per unit mass*. Here we have the 9 MeV of initial kinetic energy, of which $(1.6 + 1.4 + 2)$ MeV was converted into bremsstrahlung and into two photons of 0.51-MeV annihilation radiation. All these photons escaped from the 100-g mass. The absorbed dose, therefore, is

$$D = \frac{\text{absorbed energy}}{\text{mass}}$$
$$= \frac{\left[10\ \text{MeV} - \left(1.6 + 1.4 + 2 + (2 \times 0.51)\right)\text{MeV}\right] \cdot 1.6 \times 10^{-13}\ \frac{\text{J}}{\text{MeV}}}{0.1\ \text{kg} \cdot 1\ \frac{\text{J/kg}}{\text{Gy}}}$$
$$= 6.4 \times 10^{-12}\ \text{Gy}.$$

The National Council on Radiation Protection and Measurements (NCRP) specifies X-ray machine output and X-ray levels in units of *air kerma*. Since electronic equilibrium

generally is attained in an X-ray field in air, air kerma is, for practical purposes, a measure of exposure. The numerical relationship between air kerma and exposure in C/kg and in R is demonstrated as follows:

$$1\,\text{C/kg} = 34\ \text{Gy}\ (\text{in air})$$

$$1\ \text{Gy}\ (\text{air kerma}) = \frac{1}{34}\frac{\text{C}}{\text{kg}} = 0.02941\frac{\text{C}}{\text{kg}}$$

$$1\ \text{mGy}\ (\text{air kerma}) = 0.02941\times10^{-3}\frac{\text{C}}{\text{kg}} = 2.94\times10^{-5}\frac{\text{C}}{\text{kg}} = 29.4\frac{\mu\text{C}}{\text{kg}}.$$

In traditional units, exposure measured in R that corresponds to 1-Gy air kerma is

$$1\ \text{Gy}\ (\text{air kerma}) = 0.02941\frac{\text{C}}{\text{kg}}\cdot 3876\frac{\text{R}}{\text{C/kg}} = 114\ \text{R}$$

$$1\ \text{mGy}\ (\text{air kerma}) = 114\ \text{mR}.$$

Thus, to convert a measurement in the traditional units of roentgens, we divide the roentgen measurement by 114 to obtain the equivalent exposure in air kerma. Because of the differences between the energy absorption of air and soft tissue, an exposure of 1 mGy air kerma leads to a soft tissue absorbed dose of approximately 1 mGy, or 100 mrads.

EXAMPLE 6-9

The NCRP recommends 0.1-mGy air kerma in 1 week as the shielding design criterion for limiting occupational exposure to medical X-rays. What is the corresponding weekly exposure limit in units of

(a) mR?

(b) $\frac{\mu\text{C}}{\text{kg}}$?

Solution

(a) $0.1\text{-mGy air kerma}\cdot 114\frac{\text{mR}}{\text{mGy}} = 11.4\ \text{mR in 1 week}.$

(b) $0.1\text{-mGy air kerma}\cdot 29.4\frac{\mu\text{C/kg}}{\text{mGy}}\text{air kerma} = 2.94\frac{\mu\text{C}}{\text{kg}}\text{in 1 week}.$

Kerma decreases continuously with increasing depth in an absorbing medium because of the continuous decrease in the flux of the indirectly ionizing radiation. The absorbed dose, however, is initially less at the surface of an absorbing medium than below the surface.

It increases as electronic equilibrium is approached and the ionization density increases due to the increasing number of secondary ions produced by the primary ionizing particles (the positron–electron pairs, Compton electrons, and photoelectrons in the case of photon beams and scattered nuclei in the case of fast neutrons). This increase in absorbed dose continues until a maximum is reached, after which the absorbed dose decreases with continuing increase in depth. The maximum absorbed dose occurs at a depth approximately equal to the maximum range of the primary ionizing particles. The relation between kerma and dose for photon radiation and for fast neutrons is shown in Figure 6-7.

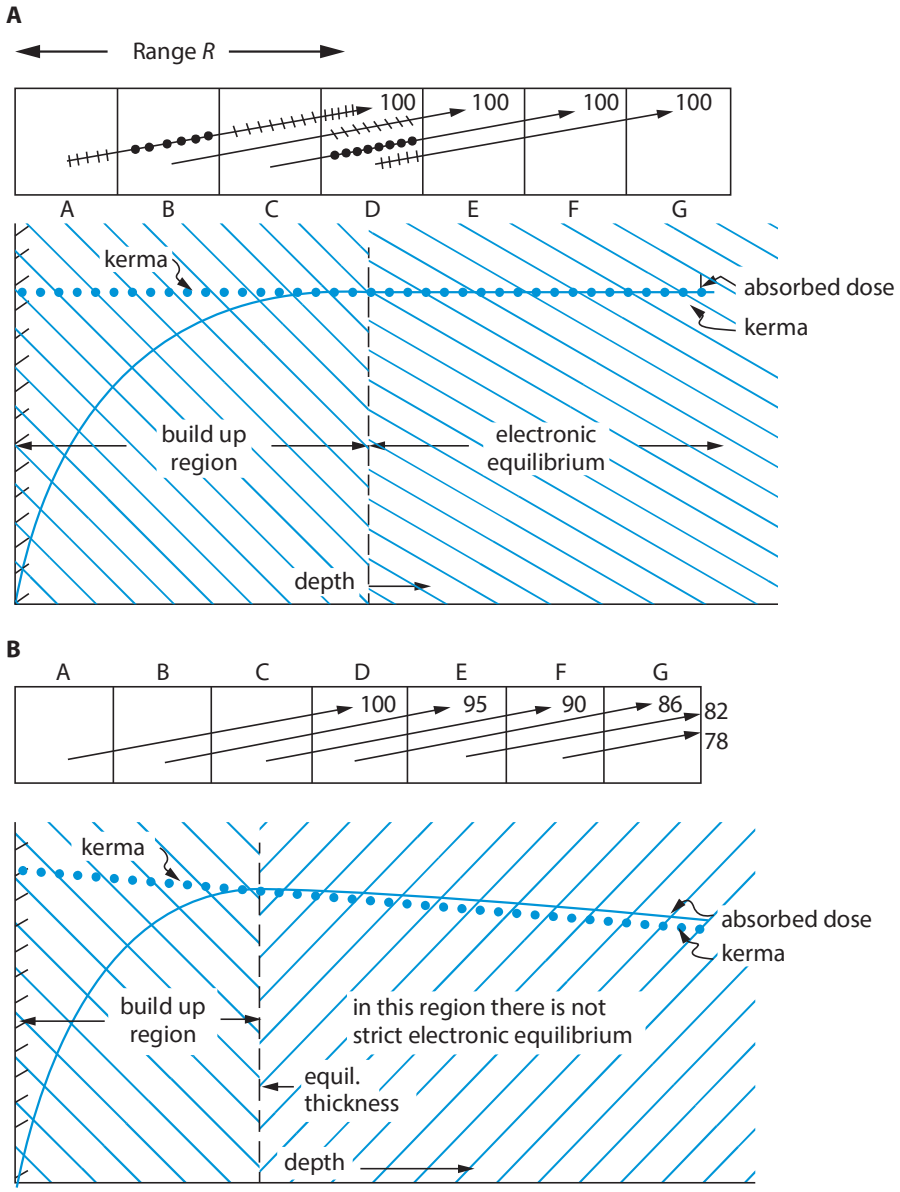

Figure 6-7. **(A)** Graph shows how absorbed dose increases with depth and where electronic equilibrium is achieved when no attenuation of the primary beam occurs; **(B)** attenuation of the beam is included in this figure, and one can see there is no electronic equilibrium past the equilibrium thickness. Units of kerma and dose are the same. (Reproduced with permission from Johns HE, Cunningham JR. *The Physics of Radiology*, 4th ed. Springfield, IL: Charles C. Thomas Publisher, Ltd.; 1983.)

For reasons discussed in Chapter 7, for health physics purposes, the air kerma is called a *sievert* (Sv) in SI units and a *rem* in traditional units (1 Sv = 100 rems).

Source Strength: Specific Gamma-Ray Emission

The radiation intensity from any given gamma-ray source is used as a measure of the strength of the source. The gamma-radiation exposure rate from a point source of unit activity at unit distance is called the *specific gamma-ray constant* and is given in units of sieverts per hour at 1 m from a 1-MBq point source (or, in the traditional system, roentgen per hour at 1 m from a 1-Ci point source). The source strength may be calculated if the decay scheme of the isotope is known. In the case of ^{131}I, for example, whose gamma rays are shown in Figure 4-6 and whose corresponding true absorption coefficients are found in Figure 5-20, we have the following:

MeV/PHOTON	PHOTONS PER TRANSFORMATION	μ_{en}/ρ, cm^2/g	μ (ENERGY ABSORPTION), m^{-1}
0.723	0.018	0.029	0.0036
0.643	0.002	0.03	0.0037
0.637	0.072	0.03	0.0037
0.503	0.004	0.03	0.0037
0.364	0.817	0.029	0.0036
0.326	0.003	0.029	0.0036
0.284	0.061	0.028	0.0034
0.177	0.003	0.026	0.0032
0.080	0.026	0.024	0.0029

Values can also be found in Appendix F.

The gamma-radiation exposure level, in Sv/h (Gy/h air kerma), is calculated by considering the energy absorbed per unit mass of air at the specified distance from the 1-MBq point source due to the photon flux at that distance, as shown in Eq. (6.15). For a distance of 1 m from the point source, we have

$$\dot{X} = \frac{f\,\frac{\text{photons}}{\text{t}} \cdot E\,\frac{\text{MeV}}{\text{photon}} \cdot 1.6\times10^{-13}\,\frac{\text{J}}{\text{MeV}} \cdot 1\times10^{6}\,\frac{\text{tps}}{\text{MBq}} \cdot 3.6\times10^{3}\,\frac{\text{s}}{\text{h}} \cdot \text{m}^{-1}}{4\pi\,(1\ \text{m})^2 \cdot \rho\,\frac{\text{kg}}{\text{m}^3} \cdot 1\,\frac{\text{J/kg}}{\text{Sv(air kerma)}}}, \quad (6.15)$$

where

$\dot{X}$ = exposure rate, Sv/h (air kerma),
f = fraction of transformations that result in a photon of energy E,
E = photon energy, MeV,
μ = linear energy absorption coefficient, m^{-1},
ρ = density of air, kg/m^3, d = distance from the source, m,

and transformations per second is abbreviated as tps.

This calculation is made for each different quantum energy and the results are summed to obtain the source strength. For 1 MBq, a 0.080-MeV gamma ray, at a distance of 1 m, we have

$$\dot{X} = \frac{2.6\times10^{-2}\cdot\left(8\times10^{-2}\right)\cdot1.6\times10^{-13}\cdot\left(1\times10^{6}\right)\cdot3.6\times10^{3}\cdot\left(2.9\times10^{-3}\right)}{4\pi\left(1\right)^{2}\cdot1.29\cdot\left(1\right)}$$

$$= 2.1\times10^{-10}\,\frac{\text{Sv}}{\text{h}}\,(\text{air kerma}).$$

Because this is at 1 m for 1 MBq, we can also express this as

$$\dot{X} = 2.1\times10^{-10}\,\frac{\text{Sv}\cdot\text{m}^2}{\text{MBq}\cdot\text{h}}.$$

The exposure rate for each of the other quanta emitted by ^{131}I is calculated in a similar manner, except that the corresponding frequency and absorption coefficient is used for each of the quanta of different energy. Low-yield photons were omitted to simplify the calculation. The results of this calculation are tabulated below:

PHOTON ENERGY, MeV	PHOTONS PER TRANSFORMATION	μ	$\frac{\text{Sv}\cdot\text{m}^2}{\text{MBq}\cdot\text{h}}$
0.723	0.018	0.0036	1.7×10^{-9}
0.643	0.002	0.0037	1.7×10^{-10}
0.637	0.072	0.0037	6.0×10^{-9}
0.503	0.004	0.0037	2.6×10^{-10}
0.364	0.817	0.0036	7.3×10^{-9}
0.326	0.003	0.0036	1.2×10^{-10}
0.284	0.061	0.0034	2.1×10^{-9}
0.177	0.003	0.0032	6.0×10^{-11}
0.080	0.026	0.0029	2.1×10^{-10}
			$\sum = 4.9\times10^{-8}\,\frac{\text{Sv}\cdot\text{m}^2}{\text{MBq}\cdot\text{h}}$

Note that the inclusion of all low-yield photons in the decay scheme would increase the value to

$$5.26\times10^{-8}\,\frac{\text{Sv}\cdot\text{m}^2}{\text{MBq}\cdot\text{h}}.$$

Equation (6.15) contains several constants: 1×10^{6} tps/MBq, 3.6×10^{3} s/h, 1.6×10^{-13} J/MeV, $4\pi(1\text{ m})^2$, and 1.29 kg/m^3. If all these constants are combined, the source strength Γ, in Sv air kerma per MBq per hour at 1 m, is given by

$$\Gamma = 3.55\times10^{-5}\sum_i f_i\cdot E_i\cdot\mu_i\,\frac{\text{Sv}\cdot\text{m}^2}{\text{MBq}\cdot\text{h}}, \qquad (6.16a)$$

where

f_i = fraction of the transformations that yield a photon of the ith energy,
E_i = energy of the ith photon, MeV, and
μ_i = linear energy absorption coefficient in air of the ith photon.

For many practical purposes, Eq. (6.16a) may be simplified. For quantum energies from about 60 keV to about 2 MeV, Figure 5-20 shows that the linear energy absorption coefficient varies little with energy; over this range, μ is about 3.5×10^{-3} m^{-1}. With this value, Eq. (6.16) may be approximated as

$$\Gamma = 1.24 \times 10^{-7} \sum f_i \cdot E_i \frac{\text{Sv} \cdot \text{m}^2}{\text{MBq} \cdot \text{h}}. \tag{6.16b}$$

If we divide Eq. (6.16b) by 34 Sv/C/kg, we obtain the specific gamma-ray emission in exposure units:

$$\Gamma = \frac{1.24 \times 10^{-7}}{34 \dfrac{\text{Sv air kerma}}{\text{C/kg}}} \sum_i f_i \cdot E_i \frac{(\text{Sv air kerma}) \cdot \text{m}^2}{\text{MBq} \cdot \text{h}}$$

$$= 3.66 \times 10^{-9} \sum_i f_i \cdot E_i \frac{\text{C/kg} \cdot \text{m}^2}{\text{MBq} \cdot \text{h}}. \tag{6.17}$$

When exposure is measured in roentgens and activity in curies, the specific gamma-ray emission is closely approximated by

$$\Gamma = 0.5 \sum_i f_i \cdot E_i \frac{\text{R} \cdot \text{m}^2}{\text{Ci} \cdot \text{h}} \tag{6.18}$$

Table 6-1 lists the specific gamma-ray emission of some isotopes that are frequently encountered by health physicists.

Beta Radiation

Examination of a beta-particle absorption curve, such as the one in Figure 5-2, shows it to be approximately linear when plotted on semilog paper. This means that the decrease in beta intensity with increasing depth into an absorbing medium, for depths less than the beta range, can be approximated by

$$\varphi = \varphi_0 e^{-\mu_\beta t}, \tag{6.19}$$

where

φ = intensity at depth t,
φ_0 = initial intensity, and
μ_β = beta absorption coefficient.

TABLE 6-1 Specific Gamma-Ray Emission Constant of Some Radioisotopes

ISOTOPE	Γ		*f* factor (cGy/R)
	$\frac{R \cdot m^2}{Ci \cdot h}$	$\frac{(C/kg) \cdot m^2}{MBq \cdot s}$	
Antimony-122	0.257	4.97×10^{-13}	0.964
Cesium-137	0.343	6.64×10^{-13}	0.962
Chromium-51	0.0178	3.44×10^{-14}	0.876
Cobalt-60	1.29	2.50×10^{-12}	0.965
Fluorine-18[a]	0.53	1.17×10^{-12}	0.876
Gold-198	0.23	4.46×10^{-13}	0.965
Iodine-125	0.175	3.38×10^{-13}	0.921
Iodine-131	0.22	4.26×10^{-13}	0.963
Indium-192	0.46	8.91×10^{-13}	0.964
Iron-59	0.718	1.39×10^{-12}	0.965
Mercury-203	0.13	2.52×10^{-13}	0.963
Potassium-42	0.137	2.65×10^{-13}	0.965
Radium-226	0.00394	7.63×10^{-15}	0.962
Sodium-22	1.18	2.29×10^{-12}	0.965
Sodium-24	1.82	3.53×10^{-12}	0.964
Zinc-65	0.307	5.94×10^{-13}	0.965

f factors are based on spectrally averaged tissue-to-air stopping power ratios assuming ICRU-44 soft tissue.
[a]For use in PET facilities, from AAPM Task Group 108: PET and PET/CT Shielding Requirements, Madsen MT et al., *Med Phys*. 33(1):4–5; January 2006.

Source: Data from Exposure Rate Constants and Lead Shielding Values for over 1,100 Radionuclides; David S. Smith and Michael G. Stabin, *Health Phys*. 102(3):271–291; 2012.

If the maximum beta energy E_m is given in MeV, then the beta absorption coefficients for air and for tissue are

$$\mu_\beta(\text{air}) = 16\left(E_m - 0.036\right)^{-1.4} \frac{\text{cm}^2}{\text{g}} \tag{6.20}$$

and

$$\mu_\beta(\text{tissue}) = 18.6\left(E_m - 0.036\right)^{-1.37} \frac{\text{cm}^2}{\text{g}}. \tag{6.21}$$

Beta Skin Contamination

When we refer to "skin dose," or to "shallow dose," we mean the dose to the viable, actively growing basal cells in the basement membrane of the skin. These cells are covered by a tissue layer of nonliving cells whose nominal thickness is 0.007 g/cm^2.

If the skin is contaminated with a radionuclide, we can calculate the dose rate to the contaminated tissue by assuming that 50% of the radiation goes down into the skin and 50% goes up and leaves the skin. If the skin is contaminated at a level of 1 Bq/cm^2 and the mean beta energy is $\bar{E}$ MeV, then the energy fluence rate, φ_β, to the basal cells at a depth of 0.007 g/cm^2 in the skin is

$$\varphi_\beta, \frac{\text{J/cm}^2}{\text{h}} = 1\ \frac{\text{Bq}}{\text{cm}^2}\cdot 1\ \frac{\text{tps}}{\text{Bq}}\cdot 0.5\cdot \bar{E}\,\frac{\text{MeV}}{\text{t}}\cdot 1.6\times 10^{-13}\ \frac{\text{J}}{\text{MeV}}\cdot 3.6\times 10^{3}\,\frac{\text{s}}{\text{h}} \cdot e^{-\left(\mu_{\beta,t}\,\frac{\text{cm}^2}{\text{g}}\cdot 0.007\,\frac{\text{g}}{\text{cm}^2}\right)}. \tag{6.22}$$

The dose rate to the basal cells, $\dot{D}_\beta$, is

$$\dot{D}_\beta = \frac{\varphi_b\,\frac{\text{J}}{\text{cm}^2\cdot\text{h}}\cdot \mu_{\beta,t}\,\frac{\text{cm}^2}{\text{g}}}{10^{-6}\,\frac{\text{J/g}}{\text{mGy}}}\ \frac{\text{mGy/h}}{\text{Bq/cm}^2}. \tag{6.23}$$

After substituting Eq. (6.22) into Eq. (6.23) and simplifying, we have

$$\dot{D}_\beta = 2.9\times 10^{-4}\,\bar{E}\cdot\mu_{\beta,t}\cdot e^{-(\mu_{\beta,t}\cdot 0.007)}\,\frac{\text{mGy/h}}{\text{Bq/cm}^2}. \tag{6.24}$$

EXAMPLE 6-10

A worker accidentally spills 3700 Bq (10 μCi) of a ^{32}P solution over an area of 10 cm^2 on her skin. What is the dose rate to the contaminated skin?

For ^{32}P:

$E_{max} = 1.71$ MeV and
$\bar{E} = 0.695$ MeV.

Solution

For ^{32}P, the absorption coefficient for tissue is calculated from Eq. (6.21):

$$\mu_{\beta,t}\ (\text{tissue}) = 18.6\left(E_m - 0.036\right)^{-1.37} = 18.6(1.71-0.036)^{-1.37} = 9.18\,\frac{\text{cm}^2}{\text{g}}.$$

Substituting the values for $\bar{E}$ and for $\mu_{\beta,t}$ into Eq. (6.24), which gives the dose rate from 1 Bq/cm^2, yields the dose conversion factor (DCF) for skin contamination with ^{32}P:

$$\text{DCF}\ (^{32}\text{P,skin}) = 2.9\times10^{-4}\,\bar{E}\cdot\mu_{\beta,t}\cdot e^{-(\mu_{\beta,t}\cdot 0.007)}$$

$$= 2.9\times10^{-4}\cdot 0.695\ \text{MeV}\cdot 9.18\frac{\text{cm}^2}{\text{g}}e^{-(9.18\cdot 0.007)}$$

$$\text{DCF}\ (^{32}\text{P,skin}) = 1.74\times10^{-3}\frac{\text{mGy/h}}{\text{Bq/cm}^2}.$$

The dose rate to the contaminated skin, therefore, is

$$\dot{D},\frac{\text{mGy}}{\text{h}} = \text{DCF}\,\frac{\text{mGy/h}}{\text{Bq/cm}^2}\cdot C_a\,\frac{\text{Bq}}{\text{cm}^2}. \tag{6.25}$$

$$\dot{D} = 1.74\times10^{-3}\,\frac{\text{mGy/h}}{\text{Bq/cm}^2}\cdot\frac{3.7\times10^3\,\text{Bq}}{10\ \text{cm}^2} = 0.64\,\frac{\text{mGy}}{\text{h}}\left(64\,\frac{\text{mrads}}{\text{h}}\right).$$

Beta Dose from Surface Contamination

If we have a plane beta-emitting surface, such as a contaminated area, then the surface dose rate may be easily calculated. If the surface concentration is C_a Bq/cm^2, then we may assume that 50% of the betas go up from the surface and 50% go down into the surface. Furthermore, some of the downward directed betas are backscattered. Under these conditions, the energy fluence rate at the contaminated surface, $\varphi_0(E)$ is

$$\varphi_0\ \frac{\text{J/cm}^2}{\text{h}} = C_a\,\frac{\text{Bq}}{\text{cm}^2}\cdot 1\,\frac{\text{tps}}{\text{Bq}}\cdot 0.5\cdot f_b\cdot\bar{E}\,\frac{\text{MeV}}{\text{t}}\cdot 1.6\times10^{-13}\,\frac{\text{J}}{\text{MeV}}\cdot 3.6\times10^3\,\frac{\text{s}}{\text{h}}. \tag{6.26}$$

Assuming that 25% of the beta energy is backscattered, then $f_b = 1.25$ and Eq. (6.26) simplifies to

$$\varphi_0 = 3.6\times10^{-10}\cdot C_a\cdot\bar{E}\,\frac{\text{J/cm}^2}{\text{h}}. \tag{6.27}$$

At a height d above the center of the contaminated surface, the beta energy flux will be reduced by the thickness of air d and will be closely approximated by

$$\varphi_d = \varphi_0\cdot e^{-(\mu_{\beta,a}\cdot d)}\,\frac{\text{J/cm}^2}{\text{h}} \qquad \text{for } d < \text{range of the betas} \tag{6.28}$$

$$= 3.6\times10^{-10}\cdot C_a\cdot\bar{E}\cdot e^{-(\mu_{\beta,a}\cdot d)}\,\frac{\text{J/cm}^2}{\text{h}}. \tag{6.29}$$

The energy fluence rate to the basal cells will be further reduced by the shielding effect of 0.007 g/cm^2 of nonviable surface skin. Therefore, the energy flux φ_b, to the basal cells is

$$\varphi_\beta = \varphi_d \cdot e^{-\left(\mu_{\beta,t}\ \frac{\text{cm}^2}{\text{g}} \cdot 0.007\ \frac{\text{g}}{\text{cm}^2}\right)} \frac{\text{J/cm}^2}{\text{h}}. \tag{6.30}$$

Substituting Eq. (6.29) for φ_d into Eq. (6.30), we have

$$\varphi_\beta = 3.6\times10^{-10} \cdot C_a \cdot \bar{E} \cdot e^{-(\mu_{\beta,a} \cdot d)} \cdot e^{-(\mu_{\beta,t} \cdot 0.007)} \frac{\text{J/cm}^2}{\text{h}}. \tag{6.31}$$

The dose rate to the basal cells in mGy/h, at a height $d\frac{\text{g}}{\text{cm}^2}$ above a contaminated area, is

$$\dot{D}_\beta, \frac{\text{mGy}}{\text{h}} = \frac{\varphi_b \frac{\text{J/cm}^2}{\text{h}} \cdot \mu_{\beta,t} \frac{\text{cm}^2}{\text{g}}}{10^{-6} \frac{\text{J/g}}{\text{mGy}}}, \tag{6.32}$$

which yields, after we substitute Eq. (6.31) for φ_β in Eq. (6.32),

$$\dot{D}_\beta = \frac{3.6\times10^{-10} \cdot C_a \cdot \bar{E} \cdot \mu_{\beta,t} \cdot e^{-(\mu_{\beta,a} \cdot d)} \cdot e^{-(\mu_{\beta,t} \cdot 0.007)}}{10^{-6} \frac{\text{J/g}}{\text{mGy}}}. \tag{6.33a}$$

After dividing by $10^{-6} \frac{\text{J/g}}{\text{mGy}}$, we have

$$\dot{D}_\beta = 3.6\times10^{-4} \cdot C_a \cdot \bar{E} \cdot \mu_{\beta,t} \cdot e^{-(\mu_{\beta,a} \cdot d)} \cdot e^{-(\mu_{\beta,t} \cdot 0.007)} \frac{\text{mGy}}{\text{h}}. \tag{6.33b}$$

EXAMPLE 6-11

A solution of ^{32}P is spilled and it contaminates a large surface to an areal concentration of $37\frac{\text{Bq}}{\text{cm}^2}$. What is the estimated beta dose rate to the skin at a height of 1 m above the contaminated area? Temperature in the laboratory is 27°C. (Neglect shielding by clothing.)

For ^{32}P:

$$\text{Range} = (0.542 \cdot 1.71) - 0.133 = 0.8\frac{\text{g}}{\text{cm}^2},$$
$$E_m = 1.71\ \text{MeV},$$
$$\bar{E} = 0.695\ \text{MeV}$$

$$\text{Air density at } 27^\circ\text{C} = 1.29\times10^{-3}\cdot\left(\frac{273}{300}\right) = 1.2\times10^{-3}\,\frac{\text{g}}{\text{cm}^3},\ \text{and}$$

$$d = 100\ \text{cm}\cdot 1.2\times10^{-3}\,\frac{\text{g}}{\text{cm}^3} = 0.12\ \frac{\text{g}}{\text{cm}^3}.$$

The beta absorption coefficient in tissue was calculated in Example 6.10 and was found to be 9.18 cm^2/g. For air, the beta absorption coefficient is calculated by substituting 1.71 for the value of E_m in Eq. (6.20):

$$\mu_{\beta,\text{air}} = 16\cdot(1.71-0.036)^{-1.4}\,\frac{\text{cm}^2}{\text{g}} = 7.78\frac{\text{cm}^2}{\text{g}}.$$

The dose rate to the skin at 1 m above the contaminated area is calculated with Eq. (6.33b):

$$\begin{aligned}\dot{D}_\beta &= 3.6\times10^{-4}\cdot C_\text{a}\cdot\mu_{\beta,t}\cdot\bar{E}\cdot e^{-\mu_{\beta,\text{a}}\cdot d}\cdot e^{-\mu_{\beta,t}\cdot 0.007}\,\frac{\text{mGy}}{\text{h}}\\ &= 3.6\times10^{-4}\cdot 37\cdot 0.695\cdot 9.18\cdot e^{-7.78\cdot 0.12}\cdot e^{-9.18\cdot 0.007}\\ &= 3.1\times10^{-2}\,\frac{\text{mGy}}{\text{h}}\left(3.1\frac{\text{mrads}}{\text{h}}\right).\end{aligned}$$

In the traditional system of units, if the contamination concentration is given as $\mu\text{Ci}/\text{cm}^2$ and dose rate is measured in $\frac{\text{mrads}}{\text{h}}$ at a height $d\frac{\text{g}}{\text{cm}^2}$ above a large contaminated area, Eq. (6.33b) becomes

$$\dot{D}_\beta = 1.3\times10^3\cdot C_\text{a}\cdot\bar{E}\cdot\mu_{\beta,t}\cdot e^{-\mu_{\beta,\text{a}}\cdot d}\cdot e^{-\mu_{\beta,t}\cdot 0.007}\,\frac{\text{mrads}}{\text{h}}. \tag{6.34}$$

Beta Submersion Dose

Inside an infinite cloud of a radionuclide,

rate of energy emission = rate of energy absorption.

In an infinite cloud containing C Bq/m^3 of a beta emitter whose mean beta energy is $\bar{E}$ MeV, the dose rate is

$$\dot{D}_\infty(\text{air}),\frac{\text{mGy}}{\text{h}} = \frac{C\frac{\text{Bq}}{\text{m}^3}\cdot 1\frac{\text{tps}}{\text{Bq}}\cdot\bar{E}\frac{\text{MeV}}{\text{t}}\cdot 1.6\times10^{-13}\,\frac{\text{J}}{\text{MeV}}\cdot 3.6\times10^3\,\frac{\text{s}}{\text{h}}}{1.29\frac{\text{kg}}{\text{m}^3}\cdot 1\frac{\text{J}/\text{kg}}{\text{Gy}}\cdot\frac{1\text{Gy}}{10^3\,\text{mGy}}}. \tag{6.35}$$

When we combine the constants in Eq. (6.35), we have

$$\dot{D}_{\infty}(\text{air}) = 4.5\times10^{-7}\cdot C\cdot \bar{E}\,\frac{\text{mGy}}{\text{h}}. \tag{6.36}$$

Since the skin of a person in an infinite medium is irradiated from one side only and since soft tissue absorbs about 10% more energy per kilogram than air does, the dose rate to the basal cells of the skin in a semi-infinite medium is

$$\dot{D}_{\beta} = 0.5\cdot(1.1)\cdot\dot{D}_{\infty}(\text{air})\cdot e^{-(\mu_{\beta,t}\cdot 0.007)}. \tag{6.37}$$

If we combine Eqs. (6.36) and (6.37), we have the beta dose to the skin of a person immersed in a large cloud of concentration C Bq/m^3:

$$\dot{D}_{\beta} = 2.5\times10^{-7}\cdot C\cdot\bar{E}\cdot e^{-(\mu_{\beta,t}\cdot 0.007)}\,\frac{\text{mGy}}{\text{h}}. \tag{6.38}$$

Generally, if there are f_i betas of average energy $\bar{E}_i$ MeV whose absorption coefficient is $\mu_{\beta_i,t}$ each, then the beta dose rate is

$$\dot{D}_{\beta} = 2.5\times10^{-7}\cdot C\cdot\sum_i f_i\cdot\bar{E}_i\cdot e^{-(\mu_{\beta_i,t}\cdot 0.007)}\,\frac{\text{mGy}}{\text{h}}, \tag{6.39}$$

and if we divide by the concentration C, we obtain DCF:

$$\text{DCF(submersion)} = 2.5\times10^{-7}\cdot\sum_i f_i\bar{E}_i\cdot e^{-(\mu_{\beta_i,t}\cdot 0.007)}\,\frac{\text{mGy/h}}{\text{Bq/m}^3}. \tag{6.40}$$

EXAMPLE 6-12

Calculate the dose rate to the skin of a person immersed in a large cloud of ^{85}Kr at a concentration of 37 kBq/m^3 (10^{-6} μCi/mL).

Solution

Krypton-85 is a pure beta emitter that is transformed to ^{85}Rb by the emission of a beta particle whose maximum energy is 0.687 MeV and whose average energy is 0.252 MeV. The tissue absorption coefficient is calculated with Eq. (6.21):

$$\mu_{\beta,\text{tissue}} = 18.6\cdot(0.687-0.036)^{-1.37} = 33.5\,\frac{\text{cm}^2}{\text{g}},$$

and the skin dose is calculated with Eq. (6.38):

$$\dot{D}_{\beta} = 2.5\times10^{-7}\cdot C\cdot\bar{E}\cdot e^{-(\mu_{\beta,t}\cdot 0.007)}\,\frac{\text{mGy}}{\text{h}}$$

$$\dot{D}_\beta = 2.5\times10^{-7}\cdot\left(3.7\times10^{4}\right)\cdot 0.252\cdot e^{-(33.5\cdot 0.007)}$$

$$\dot{D}_\beta = 1.8\times10^{-3}\ \frac{\text{mGy}}{\text{h}}\left(0.18\ \frac{\text{mrads}}{\text{h}}\right).$$

In the traditional system of units, if the concentration is given in μCi/mL, Eqs. (6.39) and (6.40) become

$$\dot{D}_\beta = 9\times10^{5}\cdot C\cdot\sum_i f_i\bar{E}_i\cdot e^{-(\mu_{\beta_i,t}\cdot 0.007)}\ \text{mrads/h} \tag{6.39a}$$

and

$$\text{DCF (submersion)} = 9\times10^{5}\cdot\sum_i f_i\bar{E}_i\cdot e^{-(\mu_{\beta_i,t}\cdot 0.007)}\ \frac{\text{mrads/h}}{\mu\text{Ci/mL}}. \tag{6.40a}$$

Beta Volume Source

In an infinitely thick (thickness ≥ beta range) volume source, the rate of energy emission is equal to the rate of energy absorption. If C_v Bq/kg is the concentration of a beta emitter whose mean energy is $\bar{E}$ MeV/beta, then the dose rate inside the infinite volume is given by

$$\dot{D}_{\infty v},\ \frac{\text{mGy}}{\text{h}} = \frac{C_v\ \dfrac{\text{Bq}}{\text{kg}}\cdot 1\ \dfrac{\text{tps}}{\text{Bq}}\cdot\bar{E}\ \dfrac{\text{MeV}}{\text{t}}\cdot 1.6\times10^{-13}\ \dfrac{\text{J}}{\text{MeV}}\cdot 3.6\times10^{3}\ \dfrac{\text{s}}{\text{h}}}{10^{-3}\ \dfrac{\text{J/kg}}{\text{mGy}}} \tag{6.41}$$

$$\dot{D}_{\infty v} = 5.76\times10^{-7}\cdot C_v\cdot\bar{E}\ \frac{\text{mGy}}{\text{h}}. \tag{6.42}$$

Since the surface of an "infinitely thick" volume source is irradiated from one side only, the dose rate at the surface is

$$\dot{D}_{\text{surf}.\infty v} = \frac{1}{2}\cdot\dot{D}_{\infty v} = 2.88\times10^{-7}\cdot C_v\cdot\bar{E}\ \frac{\text{mGy}}{\text{h}}. \tag{6.43}$$

If there are f_i betas per transformation of $\bar{E}_i$ MeV each, then

$$\dot{D}_{\text{surf}.\infty v} = 2.88\times10^{-7}\cdot C_v\cdot\sum f_i\cdot\bar{E}_i\ \frac{\text{mGy}}{\text{h}}. \tag{6.44}$$

In traditional units, if C_v is in μCi/g, then Eq. (6.44) is transformed to

$$\dot{D}_{\text{surf}.\infty v} = 1.1\times10^{3}\cdot C_v\cdot\sum f_i\cdot\bar{E}_i\ \frac{\text{mrads}}{\text{h}}. \tag{6.44a}$$

EXAMPLE 6-13

A 5-L polypropylene (specific gravity = 0.95) bottle, 3-mm wall thickness, contains 5 mCi (185 MBq) ^{32}P aqueous waste. Calculate the beta dose rate at the outside surface of the bottle.

Solution

Phosphorus-32 is transformed to ^{32}S by the emission of a beta particle whose maximum energy is 1.7 MeV and whose average energy is 0.695 MeV. Since the solution is infinitely thick relative to the range of the beta particles, the dose rate at the liquid-wall interface is calculated with Eq. (6.44a):

$$\dot{D}_{\text{surf},\infty\text{v}} = 1.1\times10^{3}\cdot C_{\text{v}}\cdot\sum f_i\cdot\bar{E}_i\,\frac{\text{mrads}}{\text{h}}$$

$$\dot{D}_{\text{surf},\infty\text{v}} = 1.1\times10^{3}\cdot\frac{5\times10^{3}\,\mu\text{Ci}}{5\times10^{3}\,\text{g}}\cdot 0.695\ \text{MeV}$$

$$= 765\,\frac{\text{mrads}}{\text{h}}\left(7.7\,\frac{\text{mGy}}{\text{h}}\right).$$

The wall will attenuate the beta dose rate according to Eq. (6.19):

$$\varphi = \varphi_0 e^{-\mu_\beta\cdot t}.$$

In this case, the wall thickness is 0.285 g/cm^2, and $\mu_\beta = 9.18$ cm^2/g (from Example 6.10), and we have

$$\dot{D}_\beta = 765\cdot e^{-9.18\cdot0.285} = 56\,\frac{\text{mrads}}{\text{h}}\left(0.56\frac{\text{mGy}}{\text{h}}\right).$$

INTERNALLY DEPOSITED RADIONUCLIDES

Radiation Dose from Internally Deposited Emitters

Radiation dose from internal emitters cannot be measured directly; it can only be calculated. The calculated dose is based on both physical and biological factors. The physical factors include the type and energy of the radiation and the radiological half-life. The biological factors include the distribution of the radioisotope within the body and the kinetic behavior, such as absorption rates, turnover rates, and retention times in the various organs and tissues. The biological factors for internal dosimetry are derived from pharmacologically based biokinetic models of the in vivo behavior of the radioisotopes.

The calculation of the absorbed dose from internally deposited radioisotopes follows directly from the definition of the gray. For an infinitely large medium containing a uniformly distributed radionuclide, the concentration of absorbed energy must be equal to the concentration of energy emitted. The energy absorbed per unit tissue mass per transformation is called the *specific effective energy* (*SEE*). For practical health physics purposes, "infinitely large" may be approximated by a tissue mass whose dimensions exceed the range of the radiation from the distributed isotope. For the case of alpha and most beta radiations, this condition is easily met in practice, and the *SEE* is simply the average energy of the radiation divided by the mass of the tissue in which it is distributed:

$$SEE(\alpha \text{ or } \beta) = \frac{\bar{E}}{m} \frac{\text{MeV / t}}{\text{kg}}. \tag{6.45}$$

The computation of the radiation-absorbed dose from a uniformly distributed beta emitter within a tissue may be illustrated with the following example.

EXAMPLE 6-14

Calculate the daily dose rate to a testis that weighs 18 g and has 6660 Bq of ^{35}S uniformly distributed throughout the organ.

Solution

Sulfur is a pure beta emitter whose maximum-energy beta particle is 0.167 MeV and whose average energy is 0.0487 MeV. The beta dose rate from q Bq uniformly dispersed in m kg of tissue, if the specific effective energy is *SEE* MeV per transformation per kg, is

$$\dot{D}_\beta, \frac{\text{Gy}}{\text{d}} = \frac{q \text{ Bq} \cdot 1 \frac{\text{tps}}{\text{Bq}} \cdot SEE \frac{\text{MeV/t}}{\text{kg}} \cdot 1.6 \times 10^{-13} \frac{\text{J}}{\text{MeV}} \cdot 8.64 \times 10^4 \frac{\text{s}}{\text{d}}}{1 \frac{\text{J/kg}}{\text{Gy}}}. \tag{6.46}$$

Substituting Eq. (6.45) into Eq. (6.46) yields

$$\begin{aligned} \dot{D}_\beta &= \frac{q \text{ Bq} \cdot 1 \frac{\text{tps}}{\text{Bq}} \cdot \bar{E} \frac{\text{MeV}}{\text{t}} \cdot 1.6 \times 10^{-13} \frac{\text{J}}{\text{MeV}} \cdot 8.64 \times 10^4 \frac{\text{s}}{\text{d}}}{m \text{ kg} \cdot 1 \frac{\text{J/kg}}{\text{Gy}}} \\ &= \frac{6.66 \times 10^3 \cdot (1) \cdot 4.87 \times 10^{-2} \cdot \left(1.6 \times 10^{-13}\right) \cdot 8.64 \times 10^4}{(0.018) \cdot 1} \\ &= 2.5 \times 10^{-4} \frac{\text{Gy}}{\text{d}} \left(0.025 \frac{\text{rad}}{\text{d}}\right). \end{aligned} \tag{6.47}$$

Effective Half-Life

The total dose absorbed during any given time interval after the deposition of the sulfur in the testis (in Example 6.14) may be calculated by integrating the dose rate over the required time interval. In making this calculation, two factors must be considered, namely:

1. *in situ* radioactive decay of the radionuclide and
2. biological elimination rate of the radionuclide.

In most instances, biological elimination follows first-order kinetics. In this case, the equation for the quantity of radioactive material within an organ at any time t after deposition of a quantity Q_0 is given by

$$Q = \left(Q_0 e^{-\lambda_R t}\right)\left(e^{-\lambda_B t}\right), \tag{6.48}$$

where λ_R is the radioactive decay constant and λ_B is the biological elimination constant. The two exponentials in Eq. (6.48) may be combined:

$$Q = Q_0 e^{-(\lambda_R + \lambda_B)t}, \tag{6.49}$$

and, if

$$\lambda_E = \lambda_R + \lambda_B, \tag{6.50}$$

we have

$$Q = Q_0 e^{-\lambda_E t} \tag{6.51}$$

where λ_E is called the *effective* elimination constant. The effective half-life is then

$$T_E = \frac{0.693}{\lambda_E}. \tag{6.52}$$

From the relationship among λ_E, λ_R, and λ_B, we have

$$\frac{1}{T_E} = \frac{1}{T_R} + \frac{1}{T_B} \tag{6.53}$$

or

$$T_E = \frac{T_R \cdot T_B}{T_R + T_B}. \tag{6.54}$$

In Example 6.14, for ^{35}S, $T_R = 87.51$ days. T_B, the biological half-life in the testis, is reported to be 623 days. The effective half-life in the testis, therefore, is

$$T_E = \frac{87.51 \cdot 623}{87.51 + 623} = 76.7 \text{ days},$$

and the effective elimination rate constant is

$$\lambda_E = \frac{0.693}{76.7 \text{ days}} = 0.009 \text{ d}^{-1}.$$

It should be noted that the effective half-life of ^{35}S in the testis is less than either the radiological or the biological half-lives. This must be so because the quantity of a radionuclide in the body is continually decreasing due to radioactive decay and biological elimination. For this reason, the effective half-life can never be greater than the shorter of either the biological or radiological half-life.

Total Dose: Dose Commitment

The dose dD during an infinitesimally small period of time dt at a time interval t after an initial dose rate $\dot{D}_0$ is

dD = instantaneous dose rate · dt

$$dD = \dot{D}_0 e^{-\lambda_E t} dt. \tag{6.55}$$

The total dose during a time interval t after deposition of the radionuclide is

$$D = \dot{D}_0 \int_0^t e^{-\lambda_E t} \, \mathrm{d}t, \tag{6.56}$$

which, when integrated, yields

$$D = \frac{\dot{D}_0}{\lambda_E}\left(1 - e^{-\lambda_E t}\right) \tag{6.57}$$

For an infinitely long time—that is, when the radionuclide is completely gone—Eq. (6.57) reduces to

$$D = \frac{\dot{D}_0}{\lambda_E}. \tag{6.58}$$

For practical purposes, an "infinitely long time" corresponds to about six effective half-lives. It should be noted that the dose due to total decay is merely equal to the product of the initial dose rate, $\dot{D}_0$, and the average life of the radionuclide within the organ, $1/\lambda_E$. For the case in Example 6.14, the total absorbed dose during the first 5 days after deposition of the radiosulfur in the testis is, according to Eq. (6.57),

$$D = \frac{2.5 \times 10^{-4} \dfrac{\text{Gy}}{\text{d}}}{0.009 \text{ d}^{-1}}\left(1 - e^{-0.009 \text{ d}^{-1} \cdot 5 \text{ d}}\right)$$

$$= 1.2 \times 10^{-3} \text{ Gy} \qquad (0.12 \text{ rad}),$$

and the dose from complete decay is, from Eq. (6.58),

$$D = \frac{2.5\times10^{-4}\,\dfrac{\text{Gy}}{\text{d}}}{0.009\ \text{d}^{-1}} = 0.028\ \text{Gy}\quad(2.8\ \text{rads}).$$

The 0.028-Gy total dose absorbed from the deposition of the radiosulfur is called the *dose commitment* to the testes from this incident. This is defined as the absorbed dose from a given practice or from a given exposure. Although the dose commitment in this example was due to an internally deposited radionuclide, the dose commitment concept is applicable to external radiation as well as to radiation from internally deposited radionuclides.

In the example cited above, the testis behaved as if the radionuclide were stored in a single compartment. In many cases, an organ or tissue behaves as if the radioisotope were stored in more than one compartment. Each compartment follows first-order kinetics and is emptied at its own clearance rate. Thus, for example, cesium is found to be uniformly distributed throughout the body, although the body behaves as if the cesium were stored in two compartments. One compartment contains 10% of the total body burden and has a retention half-time of 2 days, while the second compartment contains the other 90% of the body's cesium content and has a clearance half-time of 110 days. The retention curve for cesium, therefore, is given by the equation[2]

$$q(\text{t}) = 0.1q_0 e^{-\left(\frac{0.693}{2\ \text{days}}\cdot t\right)} + 0.9q_0 e^{-\left(\frac{0.693}{110\ \text{days}}\cdot t\right)}, \tag{6.59}$$

where $q\,(t)$ is the body burden at time t after deposition of q_0 amount of cesium in the body. Ten percent of the total is deposited in compartment 1 and 90% is deposited in compartment 2. Generally, if there is more than one compartment, the body burden at any time t after deposition of q_0 units of a radionuclide is given by

$$q(t) = f_1 q_0 e^{-\lambda_1 t} + f_2 q_0 e^{-\lambda_2 t} + \cdots + f_n q_0 e^{-\lambda_n t}, \tag{6.60}$$

where $f_1, f_2, \ldots, f_n$ is the fraction of the total activity deposited in compartments $1, 2, \ldots, n$, and $\lambda_1, \lambda_2, \ldots, \lambda_n$ is the effective clearance rates for compartments $1, 2, \ldots, n$.

Since the activity in each compartment contributes to the dose to that organ or tissue, Eq. (6.57) becomes, for the multicompartment case,

$$D = \frac{\dot{D}_{10}}{\lambda_{1\text{E}}}\left(1 - e^{-\lambda_{1\text{E}}t}\right) + \frac{\dot{D}_{20}}{\lambda_{2\text{E}}}\left(1 - e^{-\lambda_{2\text{E}}t}\right) + \cdots + \frac{\dot{D}_{n0}}{\lambda_{n\text{E}}}\left(1 - e^{-\lambda_{n\text{E}}t}\right), \tag{6.61}$$

and when the radionuclide has completely been eliminated, Eq. (6.61) reduces to

$$D(t) = \frac{\dot{D}_{10}}{\lambda_{1\text{E}}} + \frac{\dot{D}_{20}}{\lambda_{2\text{E}}} + \cdots + \frac{\dot{D}_{n0}}{\lambda_{n\text{E}}}. \tag{6.62}$$

[2]Some retention curve equations can be found in "Age-dependent Doses to the Members of the Public from Intake of Radionuclides—Part 5 Compilation of Ingestion and Inhalation Coefficients." ICRP Publication 72. Ann. ICRP 26 (1). More information can also be obtained from the ICRP free educational downloads at http://www.icrp.org/page.asp?id=145.

Gamma Emitters

For gamma-emitting isotopes, we cannot simply calculate the absorbed dose by assuming the organ to be infinitely large because gammas, being penetrating radiations, may travel great distances within the tissue and leave the tissue without interacting. Thus, only a fraction of the energy carried by photons originating in the radioisotope-containing tissue is absorbed within that tissue. Although computer programs can make complex computational methods possible, simplifying the calculation of gamma-ray doses from internal radionuclides by assuming the body to be made of spheres and cylinders and then using simple calculation techniques to determine internal dose can often rapidly and easily approximate the dose to within a few percent. For example, in the case of a uniformly distributed gamma-emitting nuclide, the dose rate at any point p due to the radioactivity in the infinitesimal volume dV at any other point at a distance r from point p, as shown in Figure 6-8, is

$$d\dot{D} = C\Gamma \frac{e^{-\mu r}}{r^2} dV, \tag{6.63}$$

where C is the concentration of the isotope, Γ is the specific gamma-ray emission, and μ is the linear energy absorption coefficient. The dose rate at point p due to all the gamma rays emitted within the tissue is computed by integrating the contributions from all the infinitesimal volume elements:

$$\dot{D} = C\Gamma \int_0^v \frac{e^{-\mu r}}{r^2} dV. \tag{6.64}$$

For the case of a sphere, the dose rate at the center (Fig. 6-9) is

$$\dot{D} = 4C\Gamma \int_{r=0}^{r=R} \int_{\theta=0}^{\theta=\pi/2} \int_{\varphi=0}^{\varphi=\pi} \frac{e^{-\mu r}}{r^2} \cdot r\, d\theta \cdot r \cos\theta\, d\varphi \cdot dr. \tag{6.65}$$

Integrating with respect to each of the variables, we have, for the dose rate at the center of the sphere,

$$\dot{D} = C\Gamma \cdot \frac{4\pi}{\mu}\left(1 - e^{-\mu R}\right). \tag{6.66}$$

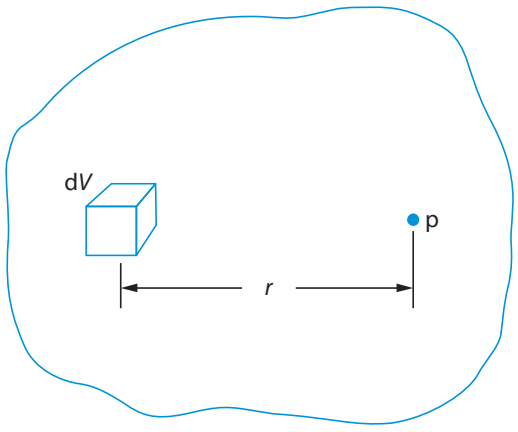

Figure 6-8. Diagram for calculating dose at a point p from the gamma rays emitted from the volume element dV in a tissue mass containing a uniformly distributed radioisotope.

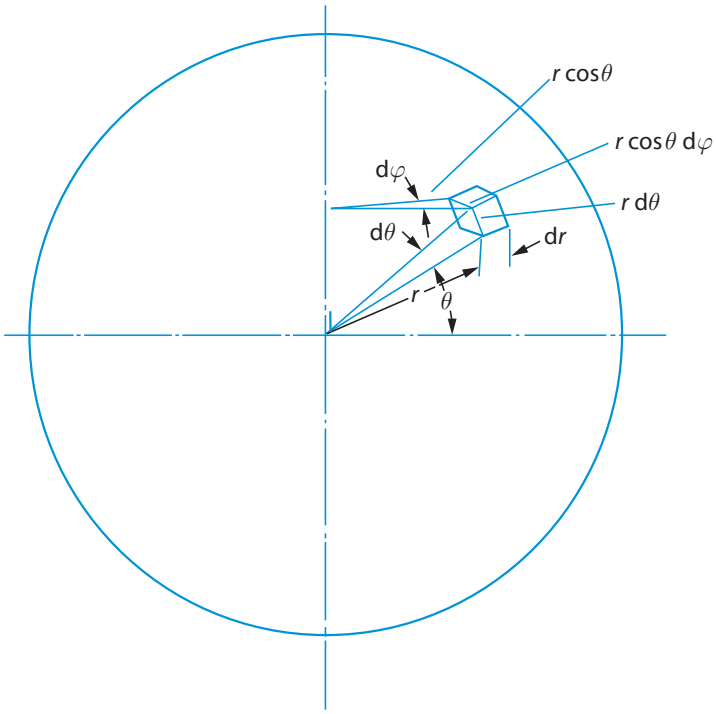

Figure 6-9. Geometry for evaluating Eq. (6.65) for the center of a sphere.

From an examination of Eqs. (6.63) to (6.66), it is seen that the factor that multiplies $C\Gamma$ depends only on the geometry of the tissue mass and hence is called the *geometry factor*.[3] The geometry factor g is defined by

$$g = \int_0^v \frac{e^{-\mu r}}{r^2} \mathrm{d}V. \tag{6.67}$$

Equation (6.64) may therefore be rewritten as

$$\dot{D} = C\Gamma g. \tag{6.68}$$

The definition of g in Eq. (6.67) applies to a given point within a volume of tissue. In many health physics instances, we are interested in the average dose rate rather than the dose rate at a specific point. For this purpose, we may define an average geometry factor as follows:

$$\bar{g} = \frac{1}{\mathrm{v}} \int g \mathrm{d}V. \tag{6.69}$$

[3]This material is mainly of historical importance in the evolution of internal dosimetry; it has been replaced by the MIRD system, which is discussed later in this chapter.

TABLE 6-2 Average Geometry Factors for Cylinders Containing a Uniformly Distributed Gamma Emitter

CYLINDER HEIGHT (cm)	RADIUS OF CYLINDER (cm)							
	3	5	10	15	20	25	30	35
2	17.5	22.1	30.3	34.0	36.2	37.5	38.6	39.3
5	22.3	31.8	47.7	56.4	61.6	65.2	67.9	70.5
10	25.1	38.1	61.3	76.1	86.5	93.4	98.4	103
20	25.7	40.5	68.9	89.8	105	117	126	133
30	25.9	41.0	71.3	94.6	112	126	137	146
40	25.9	41.3	72.4	96.5	116	131	143	153
60	26.0	41.6	73.0	97.8	118	134	148	159
80	26.0	41.6	73.3	98.4	119	135	150	161
100	26.0	41.6	73.3	98.5	119	136	150	162

The independence of the geometry factors material type is based on an estimation of $0 \leq \mu \leq 0.03\ \text{cm}^{-1}$.

Source: Reproduced with permission from Hine, G. J., and Brownell, G. L., eds. *Radiation Dosimetry*. New York, NY: Academic Press, Inc.; 1956.

For a sphere,

$$\bar{g} = \frac{3}{4}(g)_{\text{center.}} \tag{6.70}$$

At any other point in the sphere at a distance d from the center, the geometry factor is given by

$$g_p = (g)_{\text{center}} \left[0.5 + \frac{1 - \left(\frac{d}{R}\right)^2}{4\left(\frac{d}{R}\right)} \ln \frac{1 + \frac{d}{R}}{1 - \frac{d}{R}} \right]. \tag{6.71}$$

For a cylinder, the average geometry factor depends on the radius and height. Table 6-2 gives the numerical values of average geometry factors for cylinders of various heights and radii.

EXAMPLE 6-15

A spherical tank, capacity 1 m^3 and radius 0.62 m, is filled with aqueous ^{137}Cs waste containing a total activity of 37,000 MBq (1 Ci). What is the dose rate at the tank surface if we neglect absorption by the tank wall?

Solution

From Table 6-1, we find $\Gamma = 6.64 \times 10^{-13}$ (C/kg)-m^2/MBq-s with an *f* factor of 0.962. The absorption coefficient of water for the 0.661-MeV gammas from ^{137}Cs is listed (by interpolation between 0.6 and 0.8 MeV) in Table 5.4 as 0.0323 cm^2/g. Since the density of water is 1 g/cm^3, the linear absorption coefficient is 0.0323 cm^{-1}, or 3.23 m^{-1}.

First, convert the specific gamma ray constant, Γ, into dose units (Gy):

$$\Gamma = 6.64\times10^{-13}\,\frac{(\mathrm{C/kg})\cdot \mathrm{m}^2}{\mathrm{MBq}\cdot \mathrm{s}}\cdot\frac{0.962\ \mathrm{cGy}}{2.58\times10^{-4}\,\mathrm{C/kg}}\cdot\frac{\mathrm{Gy}}{100\ \mathrm{cGy}}\cdot\frac{3600\ \mathrm{s}}{\mathrm{h}} = 8.91\times10^{-8}\,\frac{\mathrm{Gy}\cdot \mathrm{m}^2}{\mathrm{MBq}\cdot \mathrm{h}}$$

The dose rate at the center of the sphere is found by substituting the respective values into Eq. (6.66):

$$\dot{D}_0 = C\Gamma\cdot\frac{4\pi}{\mu}\left(1-e^{-\mu R}\right)$$

$$\dot{D}_0 = 37\times10^{3}\,\frac{\mathrm{MBq}}{\mathrm{m}^3}\cdot 8.91\times10^{-8}\,\frac{\mathrm{Gy}\cdot \mathrm{m}^2}{\mathrm{MBq}\cdot \mathrm{h}}\cdot\frac{4\pi}{3.23\ \mathrm{m}^{-1}}\cdot\left(1-e^{-3.23\cdot0.62}\right)$$

$$\dot{D}_0 = 1.11\times10^{-2}\,\frac{\mathrm{Gy}}{\mathrm{h}}\quad\left(1.11\,\frac{\mathrm{rad}}{\mathrm{h}}\right).$$

From Eq. (6.71), we see that the surface dose rate is $0.5\cdot\dot{D}_0$. Therefore,

$$\dot{D}_{\mathrm{surface}} = 0.5\cdot\left(1.11\times10^{-2}\right) = 5.5\times10^{-3}\,\frac{\mathrm{Sv}}{\mathrm{h}}\quad\left(0.55\,\frac{\mathrm{rad}}{\mathrm{h}}\right).$$

Medical Internal Radiation Dose Methodology

To account for the partial absorption of gamma-ray energy in organs and tissues, the Medical Internal Radiation Dose (MIRD) Committee of the Society of Nuclear Medicine developed a formal system for calculating the dose to a "target" organ or tissue (*T*) from a "source" organ (*S*) (Fig. 6-10) containing a uniformly distributed radioisotope. *S* and *T* may be either the same organ or two different organs bearing any of the possible relationships to each other shown in Figure 6-10. The MIRD system separates the dose calculation into two basic components: the physical factors dealing with the radiation and the fraction of energy radiated by the deposited activity that is absorbed by the tissue, and the biological factors that are derived from physiologically based biokinetic models of the radionuclide. The fraction of the radiated energy that is absorbed by the target tissue is calculated by the application of Monte Carlo methods to the interactions and fate of photons following their emission from the deposited radionuclide.

Monte Carlo methods are useful in the solution of problems where events such as the interaction of photons with matter are governed by probabilistic rather than deterministic laws.

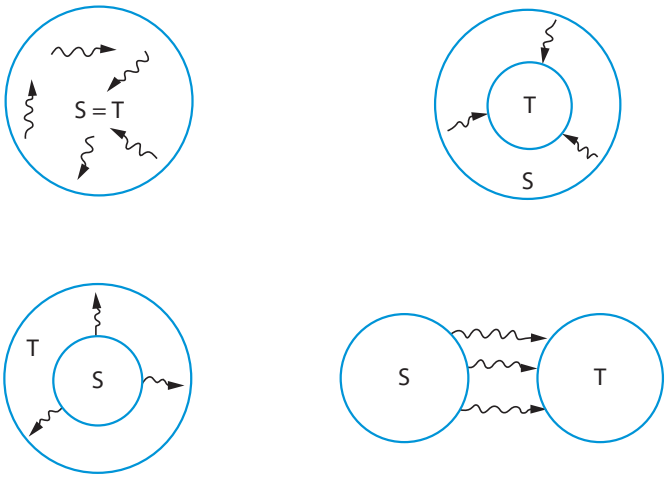

Figure 6-10. Possible relationships between source organ and target organ.

In Monte Carlo solutions, individual simulated photons (or other corpuscular radiation) are "followed" in a computer from one interaction to the next. The radioisotope is assumed to be uniformly distributed throughout a given volume of tissue (more sophisticated modeling can now vary the distribution). Since radioactive transformation is a random process occurring at a mean rate that is characteristic of the given radioisotope, we can start the process by randomly initiating a radioactive transformation. For any of these transformations, we know the energy of the emitted radiation, its starting point, and its initial direction. We also know the probability of each possible type of interaction within the organ and the energy transferred during each interaction. A situation is simulated by starting with a very large number of such nuclear transformations, following the history of each particle as it traverses the target tissue, and summing the total amount of energy that the particles dissipate within the target tissue. For a concentration of 1 Bq/cm^3 of tissue, for example, there would be one such start per cm^3/s. Since the sum of the initial energies of these particles is known, the fraction of the emitted energy absorbed by the target tissue, which is called the *absorbed fraction,* φ, can be calculated:

$$\varphi = \frac{\text{energy absorbed by target}}{\text{energy emitted by source}}. \tag{6.72}$$

Since the mean free paths of photons usually are large relative to the dimensions of the organ in which the photon-emitting isotope is distributed, the absorbed fraction for photons is less than 1. For nonpenetrating radiation, the absorbed fraction usually is either 1 or 0, depending on whether the source and target organs are the same or different.

Absorbed fractions for photons of various energies for point isotropic sources and for uniformly distributed sources in tissue and in water for spheres, cylinders, and ellipsoids have been calculated and published by MIRD in several supplements to the *Journal of Nuclear Medicine.* Tables 6-3 to 6-5 show some of these absorbed fractions.

TABLE 6-3 Absorbed Fractions for Uniform Distribution of Activity in Small Spheres[a] and Thick Ellipsoids[a]

MASS (kg)	φ									
	0.020 (MeV)	0.030 (MeV)	0.040 (MeV)	0.060 (MeV)	0.080 (MeV)	0.100 (MeV)	0.160 (MeV)	0.364 (MeV)	0.662 (MeV)	1.460 (MeV)	2.750 (MeV)
0.3	0.684	0.357	0.191	0.109	0.086	0.085	0.087	0.099	0.096	0.092	0.077
0.4	0.712	0.388	0.212	0.121	0.096	0.093	0.097	0.108	0.108	0.099	0.083
0.5	0.731	0.412	0.229	0.131	0.104	0.099	0.104	0.116	0.117	0.104	0.089
0.6	0.745	0.431	0.244	0.140	0.111	0.105	0.111	0.122	0.124	0.109	0.093
1.0	0.780	0.486	0.289	0.167	0.135	0.125	0.130	0.142	0.144	0.125	0.106
2.0	0.818	0.559	0.360	0.212	0.173	0.160	0.162	0.174	0.173	0.153	0.127
3.0	0.840	0.600	0.405	0.245	0.201	0.188	0.186	0.197	0.195	0.174	0.143
4.0	0.856	0.629	0.438	0.271	0.222	0.209	0.205	0.216	0.213	0.190	0.156
5.0	0.868	0.652	0.464	0.294	0.241	0.227	0.222	0.231	0.228	0.204	0.167
6.0	0.876	0.671	0.485	0.312	0.258	0.241	0.236	0.245	0.240	0.216	0.177

[a]The principal axes of the small spheres and thick ellipsoids are in the ratios of 1:1:1 and 1:0.667:1.333.

Source: Reprinted by permission of the Society of Nuclear Medicine from Brownell, G. L., Ellett, W. H., and Reddy, A. R. MIRD Pamphlet No. 3: Absorbed Fractions for Photon Dosimetry. *J Nuclear Med.* February 1968; 9(1 Suppl):29–39.

TABLE 6-4 Absorbed Fractions for Central Point Sources in Right Circular Cylinders[a]

MASS (kg)	0.040 (MeV)	0.080 (MeV)	0.160 (MeV)	0.364 (MeV)	0.662 (MeV)	1.460 (MeV)
2	0.528	0.258	0.224	0.240	0.229	0.200
4	0.645	0.336	0.290	0.295	0.288	0.253
6	0.712	0.391	0.335	0.333	0.326	0.286
8	0.757	0.435	0.370	0.363	0.354	0.311
10	0.789	0.471	0.399	0.387	0.376	0.332
20	0.878	0.593	0.501	0.472	0.453	0.401
30	0.917	0.668	0.568	0.528	0.504	0.446
40	0.940	0.721	0.618	0.571	0.543	0.480
50	0.954	0.761	0.658	0.605	0.575	0.509
60	0.964	0.792	0.691	0.633	0.602	0.533
70	0.971	0.818	0.719	0.658	0.625	0.553
80	0.977	0.838	0.743	0.679	0.646	0.572
90	0.981	0.856	0.763	0.698	0.664	0.588
100	0.984	0.871	0.781	0.714	0.680	0.603
120	0.989	0.894	0.811	0.742	0.708	0.629
140	0.992	0.911	0.834	0.765	0.730	0.651
160	0.994	0.924	0.852	0.784	0.749	0.669
180	0.994	0.933	0.866	0.800	0.765	0.685
200	0.994	0.939	0.877	0.813	0.777	0.698

[a]The principal axes of the right circular cylinders are in the ratio of 1:1:0.75.

Source: Reprinted by permission of the Society of Nuclear Medicine from Brownell, G. L., Ellett, W. H., and Reddy, A. R. MIRD Pamphlet No. 3: Absorbed Fractions for Photon Dosimetry. *J Nuclear Med.* February 1968; 9(1 Suppl):29–39.

EXAMPLE 6-16

The use of these absorbed dose fractions may be illustrated by their application to calculations of the dose rate to a 0.6-kg sphere made of tissue-equivalent material in which 1 MBq of ^{131}I is uniformly distributed.

Solution

In this case, energy will be absorbed from the beta particles and from the gamma rays. Since the range in tissue of the betas is very small, we can assume that all the beta energy is absorbed. For the gammas, however, only a fraction of the energy will be absorbed. The total energy absorbed from the ^{131}I is simply the sum of the emitted beta energy plus the fraction of the emitted gamma-ray energy that is absorbed by the sphere. This sum is called

TABLE 6-5 Absorbed Fractions for Central Point Sources in Spheres

MASS (kg)	φ									
	0.020 (MeV)	0.030 (MeV)	0.040 (MeV)	0.060 (MeV)	0.100 (MeV)	0.140 (MeV)	0.160 (MeV)	0.279 (MeV)	0.662 (MeV)	2.750 (MeV)
2	0.989	0.794	0.537	0.322	0.243	0.233	0.234	0.241	0.235	0.168
4	0.996	0.878	0.658	0.421	0.317	0.301	0.297	0.302	0.293	0.209
6	0.999	0.916	0.727	0.488	0.370	0.348	0.342	0.344	0.330	0.238
8	0.999	0.938	0.772	0.540	0.413	0.386	0.379	0.377	0.359	0.259
10	0.999	0.952	0.806	0.581	0.448	0.418	0.409	0.405	0.382	0.277
20	0.999	0.982	0.894	0.709	0.569	0.529	0.517	0.500	0.461	0.339
30	0.999	0.991	0.932	0.780	0.644	0.600	0.587	0.562	0.514	0.380
40	0.999	0.995	0.954	0.826	0.698	0.652	0.639	0.608	0.554	0.411
50	0.999	0.996	0.966	0.859	0.738	0.692	0.679	0.644	0.586	0.436
60	0.999	0.997	0.974	0.882	0.770	0.725	0.712	0.675	0.613	0.457
70	0.999	0.998	0.980	0.900	0.796	0.752	0.739	0.700	0.637	0.476
80	0.999	0.998	0.983	0.915	0.818	0.775	0.762	0.722	0.657	0.492
90	0.999	0.999	0.986	0.926	0.836	0.794	0.781	0.741	0.675	0.507
100	0.999	0.999	0.988	0.935	0.851	0.811	0.799	0.758	0.691	0.520
120	0.999	0.999	0.991	0.948	0.876	0.839	0.827	0.786	0.719	0.544
140	0.999	0.999	0.993	0.958	0.895	0.860	0.849	0.809	0.742	0.564
160	0.999	0.999	0.995	0.965	0.910	0.878	0.867	0.829	0.761	0.582
180	0.999	0.999	0.996	0.971	0.923	0.892	0.882	0.845	0.778	0.598
200	0.999	0.999	0.998	0.976	0.933	0.904	0.894	0.858	0.792	0.612

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the effective energy per transformation. The absorbed fraction of the gamma rays depends on the size of the absorbing medium and on the photon energy.

Using the absorbed dose fractions given in Table 6-3 and interpolating for gamma-ray energies lying between those listed in the table, we calculate the absorbed gamma-ray energy per ^{131}I transformation (as depicted in Figure 6-11 and based on ICRP 107 decay schemes[1]) as follows:

$$E_e(\gamma) = \sum_i E_{\gamma_i} \cdot n_i \cdot \varphi_i, \tag{6.73}$$

where

$E_e(\gamma)$ = absorbed gamma-ray energy, MeV/transformation,
$E_{\gamma i}$ = energy of the ith gamma photon, MeV,
n_i = number of photons of ith energy per transformation, and
φ_i = absorbed fraction of the ith photon's energy.

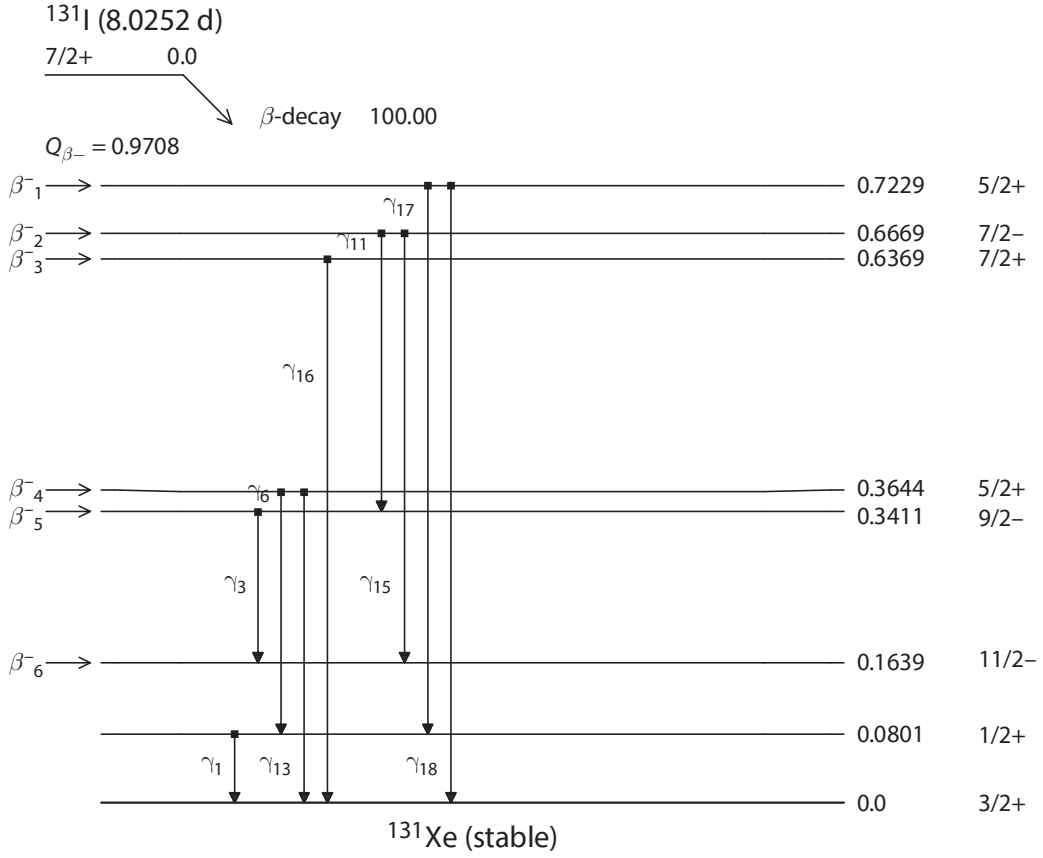


Figure 6-11. Transformation scheme for ^{131}I dosimetry. (Reprinted from: National Nuclear Data Center, Brookhaven National Laboratory, Upton, N.Y., USA.)

[1]ICRP, 2008. Nuclear Decay Data for Dosimetric Calculations. ICRP Publication 107. Ann. ICRP 38 (3)

PHOTON ENERGY, $E_{\gamma i}$ (MEV)	×	PHOTONS PER TRANSFORMATION, n_i	×	ABSORBED FRACTION, φ	=	ABSORBED ENERGY (MeV/T)
0.7229		0.0177		0.123		0.00157
0.6427		0.00217		0.124		0.00017
0.6370		0.0716		0.124		0.00566
0.5030		0.00359		0.123		0.00022
0.3258		0.00273		0.120		0.00011
0.1772		0.00269		0.112		0.00005
0.3645		0.815		0.122		0.03624
0.2843		0.0612		0.118		0.00205
0.08019		0.0262		0.111		0.00023
						$E_e(\gamma) = 0.0463$ Mev/t

We assume that 100% of the energy of emitted beta particles, Auger electrons, and conversion electrons is locally absorbed due to their short range. The mean beta energy per ^{131}I transformation, which corresponds to the effective energy for the betas, may be calculated by substituting the mean beta energies for ^{131}I listed in the table below into Eq.(6.74):

$$E_e(\beta) = \sum_i \bar{E}_{\beta i} \times n_{\beta i} \tag{6.74}$$

RADIATION TYPE	$E_{\beta i}$ (MeV)	FRACTION PER DECAY, n_i	ENERGY ABSORBED, $E_{\beta i} \times n_i$
β-1	6.936×10^{-2}	2.08×10^{-2}	1.44×10^{-3}
β-2	8.694×10^{-2}	6.45×10^{-3}	5.61×10^{-4}
β-3	9.662×10^{-2}	7.23×10^{-2}	6.99×10^{-3}
β-4	1.916×10^{-1}	8.96×10^{-1}	1.72×10^{-1}
β-6	2.832×10^{-1}	3.90×10^{-3}	1.10×10^{-3}
ce-K, γ-1	4.562×10^{-2}	3.14×10^{-2}	1.43×10^{-3}
ce-L, γ-1	7.473×10^{-2}	4.45×10^{-3}	3.32×10^{-4}
ce-K, γ-6	2.497×10^{-1}	2.50×10^{-3}	6.24×10^{-4}
ce-K, γ-13	3.299×10^{-1}	1.56×10^{-2}	5.16×10^{-3}
ce-L, γ-13	3.590×10^{-1}	2.44×10^{-3}	8.75×10^{-4}
Auger-L	3.430×10^{-3}	5.62×10^{-2}	1.93×10^{-4}
			$E_e(\beta) = 0.191$ MeV/t

Note for comparison that ICRP 107 gives a value of 0.1918 MeV/t for this calculation.

The effective energy E_e per transformation, that is, the amount of energy absorbed by the 0.6-kg tissue-equivalent sphere per ^{131}I transformation, is the sum of both the photon and beta energy deposition in the tissue,

$$E_e = E_e(\gamma) + E_e(\beta) \tag{6.75}$$
$$= 0.0463 + 0.191 = 0.237 \frac{\text{MeV}}{\text{t}}.$$

The daily dose rate to a mass of m kg that absorbs $E_e \frac{\text{MeV}}{\text{t}}$ from q Bq of activity within the mass is given by

$$\dot{D} = \frac{q\text{Bq} \cdot 1\frac{\text{tps}}{\text{Bq}} \cdot E_e \frac{\text{MeV}}{\text{t}} \cdot 1.6\times10^{-13} \frac{\text{J}}{\text{MeV}} \cdot 8.64\times10^{4} \frac{\text{s}}{\text{d}}}{m\ \text{kg} \cdot 1\frac{\text{J/kg}}{\text{Gy}}}. \tag{6.76}$$

If we substitute

$$q = 1\times10^{6}\ \text{Bq},$$
$$E_e = 0.237 \frac{\text{MeV}}{\text{t}}, \text{ and}$$
$$m = 0.6\ \text{kg}$$

into Eq. (6.76), we find the dose rate to be

$$\dot{D} = 5.5\times10^{-3} \frac{\text{Gy}}{\text{d}} \left(0.55 \frac{\text{rad}}{\text{d}}\right).$$

Note that had the contribution of the conversion and Auger electrons been neglected, the dose would still be within 5% of that calculated.

Let us now return to the MIRD method for internal dose calculation. Let us consider two organs in the body, one that contains the distributed radioactivity and is called the *source* S and the organ of interest T, the *target*, which is being irradiated by S. S and T may be either the same organ or two different organs bearing any of the possible geometric relationships shown in Figure 6-10.

The rate of energy emission by the radionuclide in the source at any time that is carried by the ith particle is given by

$$\chi_{ei} = A_s\,\text{Bq} \cdot 1\frac{\text{tps}}{\text{Bq}} \cdot \bar{E}_i \frac{\text{MeV}}{\text{particle}} \cdot n_i \frac{\text{particle}}{\text{t}} \cdot 1.6\times10^{-13} \frac{\text{J}}{\text{MeV}} \tag{6.77}$$
$$= 1.6\times10^{-13} A_s \cdot \bar{E}_i \cdot n_i \frac{\text{J}}{\text{s}}, \tag{6.78}$$

where

χ_{ei} = energy emission rate, J/s,
A_s = activity in source, Bq,
$\bar{E}_i$ = mean energy of the ith particle, MeV, and
n_i = number of particles of the ith kind per decay.

If the fraction of this emitted energy that is absorbed by the target is called, φ_i, then the amount of energy absorbed by the target due to emission from the source is given by

$$\chi_{ai} = \chi_{ei} \cdot \varphi_i = 1.6 \times 10^{-13} \cdot A_s \cdot \bar{E}_i \cdot n_i \cdot \varphi_i \frac{\text{J}}{\text{s}}. \tag{6.79}$$

Since 1 Gy corresponds to the absorption of 1 J/kg, the dose rate from the ith particle to a target that weighs m kg is given by

$$\dot{D} = \frac{\left(1.6 \times 10^{-13} \cdot A_s \cdot \bar{E}_i \cdot n_i \cdot \varphi_i\right)\frac{\text{J}}{\text{s}}}{1\frac{\text{J/kg}}{\text{Gy}} \cdot m \text{ kg}}. \tag{6.80}$$

If we let

$$\Delta_i = 1.6 \times 10^{-13} \cdot n_i \cdot \bar{E}_i \frac{\text{kg} \cdot \text{Gy}}{\text{Bq} \cdot \text{s}}, \tag{6.81}$$

then Eq. (6.80) can be written as

$$\dot{D}_i = \frac{A_s}{m} \cdot \varphi_i \cdot \Delta_i \frac{\text{Gy}}{\text{s}}. \tag{6.82}$$

Δ_i is the dose rate in an infinitely large homogeneous mass of tissue containing a uniformly distributed radionuclide at a concentration of 1 Bq/kg. Numerical values for Δ_i for each of the radiations generated by radioisotopes in infinitely large masses of tissue are included in the output data section of the decay schemes and nuclear parameters for use in radiation dose estimation that have been published by the MIRD Committee of the Society of Nuclear Medicine. Considering all types of the particles emitted from the source, the dose rate to the target organ is

$$\dot{D} = \frac{A_s}{m} \sum \varphi_i \Delta_i. \tag{6.83}$$

Since $\dot{D}$ is a function of A_s, which is a function of time, $\dot{D}$ too is a function of time. The dose commitment, that is, the total dose due to the complete decay of the deposited radionuclide, is given by integrating the dose rate with respect to time:

$$D = \int_0^\infty \dot{D}(t)\text{d}t = \frac{\sum \varphi_i \Delta_i}{m} \int_0^\infty A_s(t)\text{d}t. \tag{6.84}$$

The total number of nuclear transformations in target tissue S is called the time integrated activity in the source region, $\tilde{A}$ (previously called *cumulative activity*):

$$\tilde{A} = \int_0^\infty A_s(t)\text{d}t, \tag{6.85}$$

then the total dose to the target organ is given by

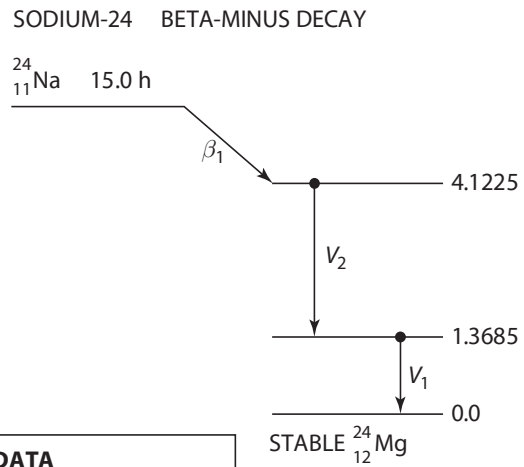
$$D = \frac{\tilde{A}}{m}\sum \varphi_i \Delta_i. \tag{6.86}$$

EXAMPLE 6-17

Calculate the total dose and initial dose rate to a 70-kg, 160-cm-tall reference man who is intravenously injected with 1-MBq ^{24}NaCl. Assume the ^{24}NaCl to become uniformly distributed within a very short time and to have a biological half-life of 10 days (240 hours[4]) for 99.7% of the ^{24}Na. The decay scheme and tables of input and output data are shown in Figure 6-12.

Solution

The decay scheme and the accompanying table of input data show one beta (actually >0.999) particle whose average energy is 0.5541 MeV, and two gamma rays (1.369 MeV



INPUT DATA

Radiation	%/disintegration	Transition energy (MeV)	Other nuclear parameters
Beta-1	99.9	1.392*	Allowed
All other betas	<0.1	—	—
Gamma-1	100.	1.3685	E2, $\alpha x < 0.00001$ (T)
Gamma-2	99.9	2.7539	E2, $\alpha x < 0.00001$ (T)
All other gammas	<0.1	—	—

Ref. Lederer, C. M. et al., *Table of Isotopes*, 6th ed.
*Endpoint energy (MeV). (T) = Theoretical value.

OUTPUT DATA

Radiation [1]	Mean number/disintegration (n_i)	Mean energy (MeV) ($\bar{E}_i$)	Δi $\left(\frac{\text{g-rad}}{\mu\text{Ci-h}}\right)$
Beta-1	0.999	0.5547	1.1803
Gamma-1	0.999	1.3685	2.9149
Gamma-2	0.999	2.7539	5.8599

Figure 6-12. Transformation scheme and input and output data for ^{24}Na dosimetry. (Reprinted by permission of the Society of Nuclear Medicine from Dillman LT. MIRD Pamphlet No. 4: Radionuclide Decay Schemes and Nuclear Parameters for Use in Radiation Dose Estimation. *J Nucl Med.* March 1969; 10(2 Suppl):1–32.)

[4]ICRP 30, Part 2, Vol. 4, issues 3–4, page 3. See ICRP 72, Vol. 26, issue 1, page 6 for additional listings of biokinetic models and biological half-lives.

and 2.754 MeV) per decay. The output data list the integral dose in an infinite medium, per unit of cumulated activity, in units of $\frac{\text{g}\cdot\text{rads}}{\mu\text{Ci}\cdot\text{h}}$ for each radiation. To convert from the old system of units found in the MIRD publications to the SI system, that is, to go from $\frac{\text{g}\cdot\text{rads}}{\mu\text{Ci}\cdot\text{h}}$ to $\frac{\text{kg}\cdot\text{Gy}}{\text{Bq}\cdot\text{s}}$, we use the following relation:

$$\frac{\text{kg}\cdot\text{Gy}}{\text{Bq}\cdot\text{s}} = \frac{\text{g}\cdot\text{rad}}{\mu\text{Ci}\cdot\text{h}} \cdot \frac{10^{-3}\,\frac{\text{kg}}{\text{g}}\cdot 10^{-2}\,\frac{\text{Gy}}{\text{rad}}}{3.7\times10^{4}\,\frac{\text{Bq}}{\mu\text{Ci}}\cdot 3.6\times10^{3}\,\frac{\text{s}}{\text{h}}}$$

$$\frac{\text{kg}\cdot\text{Gy}}{\text{Bq}\cdot\text{s}} = \frac{\text{g}\cdot\text{rad}}{\mu\text{Ci}\cdot\text{h}} \cdot 7.51\times10^{-14}. \tag{6.87}$$

To convert from SI units to traditional units:

$$\frac{\text{g}\cdot\text{rad}}{\mu\text{Ci}\cdot\text{h}} = \frac{1}{7.51\times10^{-14}} \cdot \frac{\text{kg}\cdot\text{Gy}}{\text{Bq}\cdot\text{s}} = 1.33\times10^{13} \cdot \frac{\text{kg}\cdot\text{Gy}}{\text{Bq}\cdot\text{s}}. \tag{6.88}$$

Now let us return to the problem. Since the ^{24}Na is cleared exponentially at an effective rate λ_E, the amount of activity in the source organ is given by

$$A_s(t) = A_s(0)\cdot e^{-\lambda_E t}, \tag{6.89}$$

where $A_s(0)$ is the initial activity in the source organ.

$$\tilde{A} = \int_0^\infty A_s(t)\mathrm{d}t = A_s(0)\int_0^\infty e^{-\lambda_E t}\mathrm{d}t = \frac{A_s(0)}{\lambda_E}. \tag{6.90}$$

Since

$$\lambda_E = \frac{0.693}{T_E} = \frac{0.693}{\left(\frac{T_R\cdot T_B}{T_R + T_B}\right)},$$

the biological half-life T_B is found in International Commission on Radiological Protection (ICRP) Publication 30 to be 240 hours, and the radioactive half-life T_R is 15 hours; therefore,

$$\tilde{A} = \frac{10^6\,\text{Bq}}{1.37\times10^{-5}\,\text{s}^{-1}} = 7.31\times10^{10}\,\text{Bq}\cdot\text{s}.$$

Now we must calculate $\sum \varphi_i \Delta_i$.

The absorbed fractions, φ_i, in a number of target organs and tissues, for photons ranging in energy from 0.01 to 4 MeV that originate in a number of different source organs and tissues, are tabulated in Appendix A of the *Journal of Nuclear Medicine*, Supplement

No. 3, August 1969.[5] Table 6-6 shows the absorbed fractions from a photon emitter that is uniformly distributed throughout the body, as in the case of ^{24} Na. The values of φ_i for the 1.369-MeV and 2.754-MeV gammas were found by interpolation between values in Table 6-6 and are listed below, together with Δ_i, which was found in the output data listing in Figure 6-12 and was converted to kg · fGy/Bq · s, where 1 fGy $=10^{-15}$ Gy:

RADIATION	E_i (MeV)	φ_i	Δ_i, $\frac{\text{kg}\cdot\text{fGy}}{\text{Bq}\cdot\text{s}}$	$\varphi_i\Delta_i$
Beta 1	0.554	1.000	88.64	88.64
Gamma 1	1.369	0.31	218.91	67.86
Gamma 2	2.754	0.265	440.08	116.62
				$\sum = 273.12 \frac{\text{kg}\cdot\text{fGy}}{\text{Bq}\cdot\text{s}}$

Substituting the values (and 10^{15} femtoGy (fGy) per Gy):

$$\tilde{A} = 7.31\times10^{10}\,\text{Bq}\cdot\text{s}, \quad \sum\varphi_i\Delta_i = 273.12\frac{\text{kg}\cdot\text{fGy}}{\text{Bq}\cdot\text{s}}, \quad \text{and} \quad m = 70\ \text{kg}$$

into Eq. (6.86) yields

$$D = \frac{7.31\times10^{10}\,\text{Bq}\cdot\text{s}}{70\ \text{kg}}\cdot 273.12\frac{\text{kg}\cdot\text{fGy}}{\text{Bq}\cdot\text{s}}$$
$$= 2.85\times10^{11}\,\text{fGy} \qquad (29\ \text{mrads}).$$

The initial dose rate may be found by substituting 10^6 Bq for A_s in Eq. (6.83):

$$\dot{D} = \frac{10^6\,\text{Bq}}{70\ \text{kg}}\cdot 273.12\,\frac{\text{kg}\cdot\text{fGy}}{\text{Bq}\cdot\text{s}}$$
$$= 3.9\times10^6\frac{\text{fGy}}{\text{s}} \quad \left(1.4\frac{\text{mrads}}{\text{h}}\right).$$

The physiological kinetics, on which the calculated dose from internally deposited radioactivity is based, are contained in the term for the time integrated activity in the source organ, $\tilde{A}$, while the balance of the right-hand side of Eq. (6.86) deals with physical data and measurements. The absorbed fraction φ_i represents the fraction of the energy that is absorbed by the total organ or tissue. According to Eq. (6.86), we must divide the total absorbed energy, $\Sigma\varphi_i\Delta_i$, by the mass of the target organ m. Rather than consider the fraction

[5]As the MIRD system evolved, the absorbed fraction was replaced by the *specific absorbed fraction*, which is discussed in the following paragraph.

TABLE 6-6 Absorbed Fractions (and Coefficients of Variation), Gamma Emitter Uniformly Distributed Throughout the Body[a]

TARGET ORGAN	PHOTON ENERGY (MeV)												Target Organ
	0.010		0.015		0.020		0.030		0.050		0.100		
	φ	$100\sigma_\varphi$	φ	$100\sigma_\varphi$	φ	$100\sigma_\varphi$	φ	$100\sigma_\varphi$	φ	$100\sigma_\varphi$	φ	$100\sigma_\varphi$	
Adrenals	0.270E–03	35.	0.228E–03	34.	0.175E–03	37.	0.209E–03	28.	0.131E–03	23.	0.101E–03	26.	Adrenals
Bladder	0.757E–02	6.6	0.762E–02	6.5	0.683E–02	6.6	0.625E–02	6.1	0.445E–02	5.6	0.352E–02	5.2	Bladder
GI (stom)	0.570E–02	7.6	0.507E–02	8.0	0.573E–02	7.1	0.560E–02	6.4	0.391E–02	5.8	0.273E–02	5.9	GI (stom)
GI (SI)	0.254E–01	3.6	0.236E–01	3.7	0.234E–01	3.6	0.209E–01	3.4	0.163E–01	3.1	0.120E–01	3.2	GI (SI)
GI (ULI)	0.541E–02	7.8	0.561E–02	7.5	0.647E–02	6.6	0.533E–02	5.9	0.374E–02	5.4	0.262E–02	5.7	GI (ULI)
GI (LLI)	0.350E–02	9.7	0.441E–02	8.5	0.457E–02	7.7	0.285E–02	7.9	0.256E–02	6.2	0.187E–02	6.3	GI (LLI)
Heart	0.756E–02	6.6	0.804E–02	6.3	0.769E–02	6.2	0.635E–02	6.0	0.469E–02	5.4	0.420E–02	5.0	Heart
Kidneys	0.410E–02	9.0	0.446E–02	8.5	0.412E–02	8.3	0.338E–02	7.4	0.233E–02	6.4	0.183E–02	6.6	Kidneys
Liver	0.260E–01	3.5	0.244E–01	3.6	0.249E–01	3.5	0.221E–01	3.3	0.154E–01	3.2	0.120E–01	3.2	Liver
Lungs	0.127E–01	5.1	0.142E–01	4.7	0.138E–01	4.4	0.122E–01	3.8	0.808E–02	3.4	0.551E–02	3.6	Lungs
Marrow	0.560E–01	1.4	0.594E–01	1.4	0.655E–01	1.3	0.740E–01	1.1	0.613E–01	1.1	0.329E–01	1.3	Marrow
Pancreas	0.134E–02	16.	0.103E–02	18.	0.828E–03	17.	0.780E–03	14.	0.567E–03	12.	0.449E–03	12.	Pancreas
Sk. (rib)	0.168E–01	4.4	0.206E–01	3.9	0.247E–01	3.4	0.263E–01	2.9	0.176E–01	2.9	0.764E–02	3.3	Sk. (rib)
Sk. (pelvis)	0.147E–01	4.7	0.160E–01	4.51	0.163E–01	4.3	0.224E–01	3.4	0.199E–01	3.0	0.103E–01	3.3	Sk. (pelvis)
Sk. (spine)	0.186E–01	4.2	0.190E–01	4.1	0.234E–01	3.7	0.253E–01	3.3	0.229E–01	3.0	0.144E–01	3.2	Sk. (spine)
Sk. (skull)	0.103E–01	5.6	0.115E–01	5.3	0.123E–01	5.	0.128E–01	4.6	0.722E–02	5.1	0.313E–02	6.0	Sk. (skull)
Skeleton (total)	0.144	1.4	0.153	1.3	0.167	1.2	0.188	1.1	0.153	1.1	0.810E–01	1.3	Skeleton (total)
Skin	0.258E–01	3.5	0.227E–01	3.5	0.169E–01	3.7	0.116E–01	3.3	0.758E–02	2.9	0.585E–02	3.1	Skin
Spleen	0.260E–02	11.	0.237E–02	12.	0.242E–02	11.	0.223E–02	9.1	0.149E–02	8.5	0.111E–02	8.7	Spleen
Thyroid	0.265E–03	35.	0.263E–03	34.	0.602E–04	48.	0.111E–03	36.	0.114E–03	27.	0.873E–04	29.	Thyroid
Uterus	0.999E–03	18.	0.109E–02	17.	0.122E–02	15.	0.924E–03	13.	0.712E–03	12.	0.611E–03	11.	Uterus
Trunk	0.604	0.47	0.589	0.48	0.566	0.50	0.500	0.55	0.358	0.67	0.245	0.79	Trunk
Legs	0.309	0.86	0.299	0.88	0.285	0.90	0.242	0.97	0.171	1.1	0.113	1.3	Legs
Head	0.488E–01	2.5	0.474E–01	2.5	0.440E–01	2.6	0.342E–01	2.7	0.200E–01	3.1	0.127E–01	3.1	Head
Total body	0.959	0.11	0.933	0.15	0.892	0.19	0.774	0.27	0.548	0.43	0.370	0.56	Total body

Adrenals	0.352E–04	36.	0.138E–03	35.	0.100E–03	42.	0.107E–03	43.	0.114E–03	43.			Adrenals
Bladder	0.327E–02	5.0	0.341E–02	6.6	0.274E–02	8.3	0.291E–02	8.4	0.231E–02	9.6	0.147E–02	12.	Bladder
GI (stom)	0.218E–02	7.0	0.258E–02	7.7	0.181E–02	9.8	0.199E–02	10.	0.212E–02	10.	0.119E–02	14.	GI (stom)
GI (SI)	0.106–01	3.4	0.114E–01	3.8	0.109E–01	4.2	0.915E–02	4.8	0.820E–02	5.2	0.409E–02	7.3	GI (SI)
GI (ULI)	0.256E–02	6.3	0.306–02	7.0	0.228E–02	8.9	0.209E–02	9.4	0.197E–02	10.	0.160E–02	12.	GI (ULI)
GI (LLI)	0.151E–02	7.6	0.184E–02	8.8	0.178E–02	9.7	0.181E–02	11.	0.157E–02	12.	0.673E–03	18.	GI (LLI)
Heart	0.337E–02	5.8	0.372E–02	6.6	0.301E–02	8.1	0.345E–02	7.8	0.312E–02	8.3	0.145–02	13.	Heart
Kidneys	0.171E–02	7.4	0.142E–02	9.7	0.161E–02	10.	0.152E–02	11.	0.154E–02	12.	0.904E–03	16.	Kidneys
Liver	0.111–01	3.4	0.101E–01	4.1	0.896E–02	4.7	0.912E–02	4.9	0.847E–02	5.1	0.560E–02	6.4	Liver
Lungs	0.507E–02	4.3	0.496E–02	5.2	0.466E–02	6.1	0.466E–02	6.5	0.427E–02	6.9	0.568E–02	6.4	Lungs
Marrow	0.221E–01	1.5	0.194E–01	1.0	0.182E–01	2.0	0.164E–01	2.2	0.156E–01	2.3	0.969E–02	3.0	Marrow
Pancreas	0.444E–03	14.	0.382E–03	17.	0.534E–03	19.	0.348E–03	22.	0.358E–03	24.	0.142–03	39.	Pancreas
Sk. (rib)	0.505E–02	4.1	0.435E–02	5.6	0.421E–02	6.3	0.405E–02	7.0	0.350E–02	7.7	0.338E–02	8.0	Sk. (rib)
Sk. (pelvis)	0.668E–02	3.9	0.569E–02	5.0	0.562E–02	5.7	0.511E–02	6.3	0.422E–02	7.0	0.256E–02	9.3	Sk. (pelvis)
Sk. (spine)	0.910E–02	3.6	0.763E–02	4.5	0.751E–02	5.1	0.610E–02	5.7	0.606E–02	5.9	0.341E–02	8.1	Sk. (spine)
Sk. (skull)	0.277E–02	6.3	0.304E–02	7.2	0.280E–02	8.0	0.254E–02	9.0	0.292E–02	8.8	0.224E–02	10.	Sk. (skull)
Skeleton (total)	0.550E–01	1.4	0.488E–01	1.7	0.456E–01	2.0	0.413E–01	2.2	0.396E–01	2.3	0.252E–01	3.0	Skeleton (total)
Skin	0.677E–02	3.5	0.757E–02	4.2	0.745E–02	4.8	0.759E–02	5.0	0.664E–02	5.5	0.123E–01	4.3	Skin
Spleen	0.798E–03	11.	0.116E–02	11.	0.914E–03	14.	0.903E–03	16.	0.740E–03	17.	0.368E–03	24.	Spleen
Thyroid	0.418E–04	42.							0.810E–04	46.			Thyroid
Uterus	0.408E–03	15.	0.473E–03	16.	0.517E–03	18.	0.323E–03	23.	0.364E–03	25.	0.238–03	33.	Uterus
Trunk	0.223	0.81	0.225	0.84	0.210	0.92	0.198	0.99	0.186	1.0	0.156	1.2	Trunk
Legs	0.102	1.3	0.101	1.4	0.965E–01	1.5	0.917E–01	1.6	0.846E–01	1.6	0.710E–01	1.8	Legs
Head	0.134E–01	3.2	0.147E–01	3.5	0.145E–01	3.8	0.130E–01	4.1	0.139E–01	4.1	0.127E–01	4.4	Head
Total body	0.338	0.57	0.340	0.60	0.321	0.67	0.302	0.73	0.284	0.77	0.240	0.90	Total body

[a]The digits following the letter E indicate the powers of 10 by which each number is to be multiplied; a blank in the table indicates that the coefficient of variation was greater than 50%; total body = head + trunk + legs.

Source: Reprinted by permission of the Society of Nuclear Medicine from Snyder, W. S., Ford, M. R., Warner, G. G., and Fisher, H. L., Jr. MIRD Pamphlet No. 5: Estimates of Absorbed Fractions for Monoenergetic Photon Sources Uniformly Distributed in Various Organs of a Heterogeneous Phantom. *J Nucl Med*. August 1969; 10(3 Suppl):7–52.

of energy absorbed by the target organ and then divide by the organ weight, it may be more convenient to use the *specific absorbed fraction* Φ.

$$\Phi = \frac{\text{absorbed fraction}}{\text{organ mass}} = \frac{\varphi}{m}. \tag{6.91}$$

This is the fraction of the absorbed energy per unit mass of target tissue from the *i*th particle emitted in the source organ. Specific absorbed fractions of photons of several energies for reference person, which were calculated by Monte Carlo methods, are tabulated in Appendix D. Specific absorbed fractions for beta or alpha radiation are easily calculated. In a large medium containing a uniformly distributed beta or alpha emitter, essentially all of the emitted energy is absorbed. For the case where the target and source are the same organ and where the range of the radiation from the deposited radioisotope is less than the smallest dimension of the organ in which it is deposited, the specific absorbed fraction in an organ of mass m may be closely approximated by

$$\Phi = \frac{\varphi}{m} = \frac{1}{m}. \tag{6.92}$$

When the target organ is widely separated from the source organ, that is, when the distance between them is greater than the range of beta or alpha particles, then the target absorbs no energy from the source, and $\Phi = 0$. For the case where the target tissue is a region surrounded by the source, the specific absorbed fraction in the target is

$$\Phi = \frac{1}{m(\text{source})}. \tag{6.93}$$

For example, if a beta emitter is uniformly distributed throughout the body, then the specific absorbed fraction to the liver from the radioactivity outside the liver, if the liver weighs 1.8 kg and the person weighs 70 kg, is

$$\Phi = \frac{1}{70 - 1.8} = 1.47 \times 10^{-2}\,\text{kg}^{-1}.$$

When the specific absorbed fraction is used, the absorbed dose to the person is given by

$$D = \tilde{A} \sum_i \Delta_i \Phi. \tag{6.94}$$

Since every organ in the body is a target for radiation from the source organ, the exact target–source relationship is identified explicitly by the symbol

$$(r_T \leftarrow r_S),$$

where r_T represents the target organ and r_S represents the source organ. Thus, the dose to target organ r_T from activity $\tilde{A}_S$ in source organ r_S is written as

$$D(r_T \leftarrow r_S) = \tilde{A}_S \sum_i \Delta_i \Phi(r_T \leftarrow r_S). \tag{6.95}$$

Using the specific absorbed fraction, we can define the quantity

$$S(r_T \leftarrow r_S) = \sum_i \Delta_i \Phi(r_T \leftarrow r_S). \tag{6.96}$$

Since S depends only on physical factors, such as the geometrical relationship between the source and the target, we can calculate the value of S for all of the target–source relationships of interest and for any radioisotope in the source organ. The dose to any target organ r_T from a source organ r_S is then

$$D(r_T \leftarrow r_S) = \tilde{A}_S \cdot S(r_T \leftarrow r_S). \tag{6.97}$$

Furthermore, since the radioactivity is usually widespread within the body, a target organ may be irradiated by several different source organs. The dose to the target, therefore, is

$$D(r_T) = \sum_S D(r_T \leftarrow r_S). \tag{6.98}$$

Tables of $S(r_T \leftarrow r_S)$, per unit cumulated activity for numerous target and source organs and for numerous radionuclides of interest, are published in MIRD Pamphlet No. 11. (Source organs are labeled with the subscript h and target k in MIRD 11.) Tables 6-7 and 6-8, which are excerpted from Pamphlet No. 11, give the values of S for ^{203}Hg and ^{99m}Tc. The use of the "S" tables in calculating internal dose is illustrated by Example 6.18.

EXAMPLE 6-18

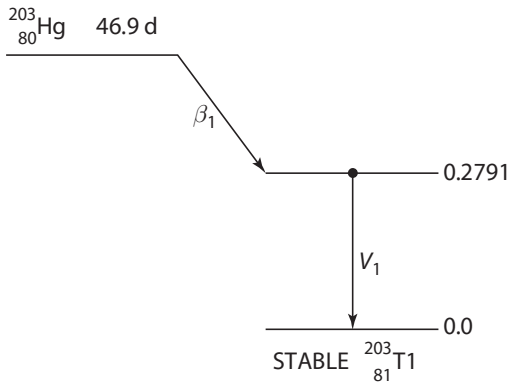
An accidental inhalation exposure to ^{203}Hg-tagged mercury vapor led to a deposition of 0.5 MBq (13.5 μCi) ^{203}Hg ($T_R = 47$ days) in the kidneys. Calculate the dose commitment to the kidneys from this inhalation.

Solution

The ICRP biokinetic model for inorganic mercury (ICRP Publication 30/2, 1980) assumes that after transfer from the blood, 8% of the absorbed Hg is in the kidneys and 92% is uniformly distributed throughout the body. Of all the body's Hg content, whether in the kidneys or elsewhere, 95% is assumed to be cleared with a biological half-life of 40 days and 5% with a biological half-life of 10,000 days.

The decay scheme for ^{203}Hg (Fig. 6-13) shows that the mercury emits a single group of beta particles whose maximum energy is 0.213 MeV and whose mean energy is listed in the output data as 0.058 MeV. A 0.279-MeV gamma ray is emitted after each beta transformation. The gamma ray, however, is internally converted in 18.3% of the transformations, thus leading to conversion electrons from the K, L, or M energy levels and, therefore, effective gamma-ray emission occurs in only 81.7% of the transformations.

Table 6-7 lists the absorbed dose per unit cumulated ^{203}Hg activity. For the kidneys as the source, S (kidneys ← kidneys) = 8.1×10^{-4} rad/μCi × h, and for the total body as the source, the dose to the kidney, S (kidneys ← total body) = 6.1×10^{-6} rad/μCi × h. The total dose to the kidneys is the sum of the doses due to the ^{203}Hg deposited in the kidneys, and



INPUT DATA

Radiation	%/dis-integration	Transition energy (MeV)	Other nuclear parameters
Beta-1	100.	0.213*	Allowed
Gamma-1	100.	0.2791	E2 + 38% M1, $\alpha\kappa = 0.162$, $K/L = 3.39$, $K/M = 11.3$

Ref.: Lederer, C. M. et al., *Table of Isotopes*, 6th ed.
* Endpoint energy (MeV).

OUTPUT DATA

Radiation (i)	Mean number/disintegration (n_i)	Mean energy (MeV) ($\bar{E}_i$)	Δi $\left(\frac{\text{g–rad}}{\mu\text{Ci–h}}\right)$
Beta-1	1.00	0.0580	0.1235
Gamma-1	0.817	0.2791	0.4857
K int. con. electron, gamma-1	0.132	0.1936	0.0544
L int. con. electron, gamma-1	0.0390	0.2648	0.0220
M int. con. electron, gamma-1	0.0117	0.2761	0.0069
K α-1 *x*-rays	0.0634	0.0729	0.0098
K α-2 *x*-rays	0.0350	0.0708	0.0053
K β-1 *x*-rays	0.0223	0.0826	0.0039
K β-2 *x*-rays	0.0063	0.0855	0.0011
L α *x*-rays	0.0273	0.0103	0.0006
L β *x*-rays	0.0256	0.0122	0.0007
L γ *x*-rays	0.0035	0.0143	0.0001
KLL Auger electron	0.0032	0.0570	0.0004
KLX Auger electron	0.0018	0.0683	0.0003
KXY Auger electron	0.0003	0.0796	0.0001
LMM Auger electron	0.0893	0.0084	0.0016
MXY Auger electron	0.271	0.0030	0.0017

Figure 6-13. Transformation scheme and input and output data for ^{203}Hg dosimetry. (Reprinted by permission of the Society of Nuclear Medicine from Dillman LT. MIRD Pamphlet No. 4: Radionuclide Decay Schemes and Nuclear Parameters for Use in Radiation Dose Estimation. *J Nucl Med*. March 1969; 10(2 Suppl):1–32.)

also of the radiomercury in the rest of the body. If 0.5 MBq (13.5 μCi) in the kidneys represents 8% of the total Hg in the body, then the total activity in the body is

$$\frac{0.5\,\text{MBq}}{0.08} = 6.25\,\text{MBq}\,(168.9\,\mu\text{Ci}).$$

Since 0.5 MBq is in the kidneys, the amount of ^{203}Hg distributed throughout the rest of the body is 6.25 − 0.5 = 5.75 MBq.

Of the 6.25 MBq deposited in the body, 95%, or 5.938 MBq, will be eliminated with an effective half-life, T_E, from Eq. (6.54):

$$T_E = \frac{T_R \cdot T_B}{T_R + T_B} = \frac{47\,\text{days} \cdot 40\,\text{days}}{47\,\text{days} + 40\,\text{days}} = 21.6\,\text{days},$$

and the remaining 5% of the deposited ^{203}Hg will have an effective half-life of 47 days. The corresponding effective clearance rate constants are

$$\lambda_1 = \frac{0.693}{21.6\ \text{days}} = 3.2 \times 10^{-2}\,\text{d}^{-1}$$

and

$$\lambda_2 = \frac{0.693}{47\ \text{days}} = 1.47 \times 10^{-2}\,\text{d}^{-1}.$$

TABLE 6-7 *S*, Absorbed Dose Per Unit Cumulated Activity (rad/μCi · h) Mercury-203 (Half-Life 1.12E–03 h)

TARGET ORGANS	SOURCE ORGANS									
	Adrenals	Bladder Contents	INTESTINAL TRACT				Kidneys	Liver	Lungs	Other Tissue (Muscle)
			Stomach Contents	SI Contents	ULI Contents	LLI Contents				
Adrenals	1.6E–02	3.6E–07	4.2E–06	2.7E–06	1.7E–06	8.4E–07	1.9E–05	9.0E–06	4.4E–06	2.7E–06
Bladder wall	2.1E–07	6.6E–04	5.0E–07	5.0E–06	3.8E–06	1.1E–05	5.9E–07	3.6E–07	1.0E–07	3.2E–06
Bone (total)	2.9E–06	1.3E–06	1.3E–06	1.8E–06	1.6E–06	2.3E–06	2.1E–06	1.6E–06	2.1E–06	1.9E–06
GI (stom. wall)	5.3E–06	5.2E–07	5.1E–04	6.5E–06	6.7E–06	3.2E–06	6.1E–06	3.5E–06	3.3E–06	2.5E–06
GI (SI)	1.6E–06	5.1E–06	4.7E–06	3.2E–04	3.0E–05	1.7E–05	5.1E–06	3.0E–06	3.9E–07	2.8E–06
GI (ULI wall)	1.7E–06	4.2E–06	6.3E–06	4.2E–05	5.5E–04	7.7E–06	5.2E–06	4.5E–06	4.9E–07	2.9E–06
GI (LLI wall)	4.6E–07	1.3E–05	2.3E–06	1.3E–05	5.4E–06	8.7E–04	1.5E–06	4.6E–07	1.8E–07	3.1E–06
Kidneys	2.1E–05	5.5E–07	6.3E–06	5.5E–06	5.0E–06	1.7E–06	8.1E–04	6.9E–06	1.7E–06	2.5E–06
Liver	8.9E–06	4.1E–07	3.6E–06	3.3E–06	4.6E–06	5.3E–07	7.0E–06	1.6E–04	4.4E–06	2.0E–06
Lungs	4.4E–06	5.6E–08	3.1E–06	4.8E–07	5.2E–07	1.6E–07	1.6E–06	4.5E–06	2.4E–04	2.4E–06
Marrow (red)	5.2E–06	3.0E–06	2.3E–06	5.8E–06	5.0E–06	6.9E–06	5.3E–06	2.3E–06	2.7E–06	2.9E–06
Other tissues (musc.)	2.7E–06	3.2E–06	2.5E–06	2.8E–06	2.7E–06	3.1E–06	2.5E–06	2.0E–06	2.4E–06	1.0E–05
Ovaries	1.0E–06	1.3E–05	8.1E–07	1.8E–05	2.1E–05	3.4E–05	2.2E–06	7.6E–07	2.3E–07	3.6E–06
Pancreas	1.5E–05	5.2E–07	3.2E–05	3.7E–06	4.1E–06	1.3E–06	1.2E–05	7.4E–06	4.7E–06	3.2E–06
Skin	1.1E–06	1.1E–06	9.0E–07	8.5E–07	8.6E–07	9.8E–07	1.1E–06	9.8E–07	1.1E–06	1.5E–06
Spleen	1.2E–05	3.3E–07	1.8E–05	2.8E–06	2.5E–06	1.5E–06	1.6E–05	1.7E–06	4.1E–06	2.6E–06
Testes	9.2E–08	8.4E–06	9.9E–08	6.3E–07	6.5E–07	3.7E–06	2.2E–07	1.6E–07	2.8E–08	2.1E–06
Thyroid	3.0E–07	9.0E–09	2.3E–07	4.8E–08	5.2E–08	2.0E–08	1.3E–07	3.8E–07	1.7E–06	2.4E–06
Uterus (nongrvd)	3.3E–06	2.8E–05	1.5E–06	1.7E–05	8.5E–06	1.2E–05	1.8E–06	7.2E–07	1.7E–07	4.0E–06
Total body	6.0E–06	3.7E–06	4.1E–06	6.0E–06	4.9E–06	5.2E–06	6.0E–06	6.1E–06	5.6E–06	5.5E–06

(*continued*)

TABLE 6-7 *S*, Absorbed Dose Per Unit Cumulated Activity (rad/μCi · h) Mercury-203 (Half-Life 1.12E–03 h) (*Continued*)

TARGET ORGANS	Ovaries	Pancreas	SKELETON			Skin	Spleen	Testes	Thyroid	Total Body
			Red Marrow	Cort. Bone	Tra. Bone					
Adrenals	7.7E–07	1.5E–05	4.4E–06	2.5E–06	2.5E–06	1.4E–06	1.2E–05	9.2E–08	3.0E–07	6.5E–06
Bladder wall	1.2E–05	3.0E–07	1.4E–06	9.3E–07	9.3E–07	1.0E–06	2.9E–07	8.8E–06	9.1E–09	6.2E–06
Bone (total)	2.1E–06	2.0E–06	1.0E–05	4.9E–05	4.0E–05	1.6E–06	1.6E–06	1.4E–06	1.5E–06	6.0E–06
GI (stom. wall)	1.4E–06	3.3E–05	1.9E–06	1.0E–06	1.0E–06	1.0E–06	1.7E–05	1.2E–07	1.1E–07	6.4E–06
GI (SI)	2.1E–05	3.3E–06	4.7E–06	1.4E–06	1.4E–06	9.0E–07	2.5E–06	8.0E–07	2.4E–08	6.6E–06
GI (ULI wall)	2.0E–05	4.0E–06	3.8E–06	1.3E–06	1.3E–06	8.9E–07	2.3E–06	6.4E–07	2.0E–08	6.3E–06
GI (LLI wall)	2.6E–05	1.0E–06	5.3E–06	1.9E–06	1.9E–06	9.4E–07	1.2E–06	5.1E–06	1.6E–08	6.2E–06
Kidneys	1.8E–06	1.1E–05	4.1E–06	1.6E–06	1.6E–06	1.2E–06	1.6E–05	1.2E–07	7.3E–08	6.1E–06
Liver	1.1E–06	7.8E–06	1.8E–06	1.2E–06	1.2E–06	1.1E–06	1.9E–06	7.7E–08	2.3E–07	6.0E–06
Lungs	1.4E–07	4.5E–06	2.2E–06	1.8E–06	1.8E–06	1.2E–06	4.0E–06	2.2E–08	1.8E–06	5.6E–06
Marrow (red)	7.2E–06	3.8E–06	1.3E–04	6.5E–06	3.5E–05	1.5E–06	2.5E–06	1.1E–06	1.7E–06	6.4E–06
Other tissues (musc.)	3.6E–06	3.2E–06	2.3E–06	1.9E–06	1.9E–06	1.5E–06	2.6E–06	2.1E–06	2.4E–06	5.5E–06
Ovaries	2.1E–02	6.0E–07	4.7E–06	1.4E–06	1.4E–06	8.0E–07	1.2E–06	0.0	1.9E–08	6.3E–06
Pancreas	9.1E–07	2.5E–03	3.0E–06	1.9E–06	1.9E–06	1.0E–06	3.4E–05	1.2E–07	1.8E–07	6.7E–06
Skin	1.5E–07	8.1E–07	1.2E–06	1.4E–06	1.4E–06	8.4E–05	9.6E–07	2.7E–06	1.5E–06	4.5E–06
Spleen	1.0E–06	3.5E–05	1.6E–06	1.3E–06	1.3E–06	1.1E–06	1.4E–03	8.6E–08	2.2E–07	6.2E–06
Testes	0.0	1.3E–07	6.1E–07	1.1E–06	1.1E–06	1.8E–06	1.3E–07	6.6E–03	2.8E–09	5.3E–06
Thyroid	1.9E–08	2.9E–07	1.4E–06	1.7E–06	1.7E–06	1.4E–06	2.2E–07	2.8E–09	1.1E–02	5.3E–06
Uterus (nongrvd)	3.7E–05	1.1E–06	4.0E–06	1.1E–06	1.1E–06	7.8E–07	7.4E–07	0.0	1.8E–08	6.7E–06
Total body	6.7E–06	6.6E–06	5.9E–06	5.6E–06	5.6E–06	4.5E–06	6.1E–06	5.5E–06	5.3E–06	5.6E–06

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TABLE 6-8 *S*, Absorbed Dose Per Unit Cumulated Activity, (rad/μCi · H) Technetium-99M (Half-Life, 6.03 h)

TARGET ORGANS	SOURCE ORGANS									
	Adrenals	Bladder Contents	INTESTINAL TRACT				Kidneys	Liver	Lungs	Other Tissue (Muscle)
			Stomach Contents	SI Contents	ULI Contents	LLI Contents				
Adrenals	3.1E–03	1.5E–07	2.7E–06	1.0E–06	9.1E–07	3.6E–07	1.1E–05	4.5E–06	2.7E–06	1.4E–06
Bladder wall	1.3E–07	1.6E–04	2.7E–07	2.6E–06	2.2E–06	6.9E–06	2.8E–07	1.6E–07	3.6E–08	1.8E–06
Bone (total)	2.0E–06	9.2E–07	9.0E–07	1.3E–06	1.1E–06	1.6E–06	1.4E–06	1.1E–06	1.5E–06	9.8E–07
GI (stom. wall)	2.9E–06	2.7E–07	1.3E–04	3.7E–06	3.8E–06	1.8E–06	3.6E–06	1.9E–06	1.8E–06	1.3E–06
GI (SI)	8.3E–07	3.0E–06	2.7E–06	7.8E–05	1.7E–05	9.4E–06	2.9E–06	1.6-06	1.9E–07	1.5E–06
GI (ULI wall)	9.3E–07	2.2E–06	3.5E–06	2.4E–05	1.3E–04	4.2E–06	2.9E–06	2.5E–06	2.2E–07	1.6E–06
GI (LLI wall)	2.2E–07	7.4E–06	1.2E–06	7.3E–06	3.2E–06	1.9E–04	7.2E–07	2.3E–07	7.1E–08	1.7E–06
Kidneys	1.1E–05	2.6E–07	3.5E–06	3.2E–06	2.8E–06	8.6E–07	1.9E–04	3.9E–06	8.4E–07	1.3E–06
Liver	4.9E–06	1.7E–07	2.0E–06	1.8E–06	2.6E–06	2.5E–07	3.9E–06	4.6E–05	2.5E–06	1.1E–06
Lungs	2.4E–06	2.4E–08	1.7E–06	2.2E–07	2.6E–07	7.9E–08	8.5E–07	2.5E–06	5.2E–05	1.3E–06
Marrow (red)	3.6E–06	2.2E–06	1.6E–06	4.3E–06	3.7E–06	5.1E–06	3.8E–06	1.6E–06	1.9E–06	2.0E–06
Other tissues (musc.)	1.4E–06	1.8E–06	1.4E–06	1.5E–06	1.5E–06	1.7E–06	1.3E–06	1.1E–06	1.3E–06	2.7E–06
Ovaries	6.1E–07	7.3E–06	5.0E–07	1.1E–05	1.2E–05	1.8E–05	1.1E–06	4.5E–07	9.4E–08	2.0E–06
Pancreas	9.0E–06	2.3E–07	1.8E–05	2.1E–06	2.3E–06	7.4E–07	6.6E–06	4.2E–06	2.6E–06	1.8E–06
Skin	5.1E–07	5.5E–07	4.4E–07	4.1E–07	4.1E–07	4.8E–07	5.3E–07	4.9E–07	5.3E–07	7.2E–07
Spleen	6.3E–06	6.6E–07	1.0E–05	1.5E–06	1.4E–06	8.0E–07	8.6E–06	9.2E–07	2.3E–06	1.4E–06
Testes	3.2E–08	4.7E–06	5.1E–08	3.1E–07	2.7E–07	1.8E–06	8.8E–08	6.2E–08	7.9E–09	1.1E–06
Thyroid	1.3E–07	2.1E–09	8.7E–08	1.5E–08	1.6E–08	5.4E–09	4.8E–08	1.5E–07	9.2E–07	1.3E–06
Uterus (nongrvd)	1.1E–06	1.6E–05	7.7E–07	9.6E–06	5.4E–06	7.1E–06	9.4E–07	3.9E–07	8.2E–08	2.3E–06
Total body	2.2E–06	1.9E–06	1.9E–06	2.4E–06	2.2E–06	2.3E–06	2.2E–06	2.2E–06	2.0E–06	1.9E–06

(*continued*)

TABLE 6-8 *S*, Absorbed Dose Per Unit Cumulated Activity, (rad/μCi · H) Technetium-99M (Half-Life, 6.03 h) (*Continued*)

TARGET ORGANS	Ovaries	Pancreas	SKELETON			Skin	Spleen	Testes	Thyroid	Total Body
			Red Marrow	Cort. Bone	Tra. Bone					
Adrenals	3.3E–07	9.1E–06	2.3E–06	1.1E–06	1.1E–06	6.8E–07	6.3E–06	3.2E–08	1.3E–07	2.3E–06
Bladder Wall	7.2E–06	1.4E–07	9.9E–07	5.1E–07	5.1E–07	4.9E–07	1.2E–07	4.8E–06	2.1E–09	2.3E–06
Bone (total)	1.5E–06	1.5E–06	4.0E–06	1.2E–05	1.0E–05	9.9E–07	1.1E–06	9.2E–07	1.0E–06	2.5E–06
GI (stom. wall)	8.1E–07	1.8E–05	9.5E–07	5.5E–07	5.5E–07	5.4E–07	1.0E–05	3.2E–08	4.5E–08	2.2E–06
GI (SI)	1.2E–05	1.8E–06	2.6E–06	7.3E–07	7.3E–07	4.5E–07	1.4E–06	3.6E–07	9.3E–09	2.5E–06
GI (ULI wall)	1.1E–05	2.1E–06	2.1E–06	6.9E–07	6.9E–07	4.6E–07	1.4E–06	3.1E–07	1.1E–08	2.4E–06
GI (LLI wall)	1.5E–05	5.7E–07	2.9E–06	1.0E–06	1.0E–06	4.8E–07	6.1E–07	2.7E–06	4.3E–09	2.3E–06
Kidneys	9.2E–07	6.6E–06	2.2E–06	8.2E–07	8.2E–07	5.7E–07	9.1E–06	4.0E–08	3.4E–08	2.2E–06
Liver	5.4E–07	4.4E–06	9.2E–07	6.6E–07	6.6E–07	5.3E–07	9.8E–07	3.1E–08	9.3E–08	2.2E–06
Lungs	6.0E–08	2.5E–06	1.2E–06	9.4E–07	9.4E–07	5.8E–07	2.3E–06	6.6E–09	9.4E–07	2.0E–06
Marrow (red)	5.5E–06	2.8E–06	3.1E–05	4.1E–06	9.1E–06	9.5E–07	1.7E–06	7.3E–07	1.1E–06	2.9E–06
Other tissues (Musc.)	2.0E–06	1.8E–06	1.2E–06	9.8E–07	9.8E–07	7.2E–07	1.4E–06	1.1E–06	1.3E–06	1.9E–06
Ovaries	4.2E–03	4.1E–07	3.2E–06	7.1E–07	7.1E–07	3.8E–07	4.0E–07	0.0	4.9E–09	2.4E–06
Pancreas	5.0E–07	5.8E–04	1.7E–06	8.5E–07	8.5E–07	4.4E–07	1.9E–05	5.5E–08	7.2E–08	2.4E–06
Skin	4.1E–07	4.0E–07	5.9E–07	6.5E–07	6.5E–07	1.6E–05	4.7E–07	1.4E–06	7.3E–07	1.3E–06
Spleen	4.9E–07	1.9E–05	9.2E–07	5.8E–07	5.8E–07	5.4E–07	3.3E–04	1.7E–08	1.1E–07	2.2E–06
Testes	0.0	5.5E–08	4.5E–07	6.4E–07	6.4E–07	9.1E–07	4.8E–08	1.4E–03	5.0E–10	1.7E–06
Thyroid	4.9E–09	1.2E–07	6.8E–07	7.9E–07	7.9E–07	6.9E–07	8.7E–08	5.0E–10	2.3E–03	1.5E–06
Uterus (nongrvd)	2.1E–05	5.3E–07	2.2E–06	5.7E–07	5.7E–07	4.0E–07	4.0E–07	0.0	4.6E–09	2.6E–06
Total body	2.6E–06	2.6E–06	2.2E–06	2.0E–06	2.0E–06	1.3E–06	2.2E–06	1.9E–06	1.8E–06	2.0E–06

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The cumulated activity in the kidney is given by extending Eq. (6.90) to several compartments, $c_1, c_2, \ldots, c_n$, within an organ or a tissue is given as

$$\tilde{A} = \frac{A_{c_1}(0)}{\lambda_{E_1}} + \frac{A_{c_2}(0)}{\lambda_{E_2}} + \cdots + \frac{A_{c_n}(0)}{\lambda_{E_n}}. \tag{6.99}$$

From Table 6-7, we find S (kidneys ← kidneys) to be 8.1×10^{-4} rad/μCi · h. To convert rad/μCi · h to Gy/Bq · d:

$$\frac{\text{Gy}}{\text{Bq}\cdot\text{d}} = \frac{\text{rad}}{\mu\text{Ci}\cdot\text{h}} \cdot \frac{1\text{Gy}}{100\ \text{rad}} \cdot \frac{1\mu\text{Ci}}{3.7\times10^4\,\text{Bq}} \cdot \frac{24\ \text{h}}{\text{d}}$$

$$\frac{\text{Gy}}{\text{Bq}\cdot\text{d}} = \frac{\text{rad}}{\mu\text{Ci}\cdot\text{h}} \cdot 6.5\times10^{-6}. \tag{6.100}$$

Using the conversion factor in Eq. (6.100), we find that

$$S(\text{kidney} \leftarrow \text{kidney}) = 8.1\times10^{-4}\,\frac{\text{rad}}{\mu\text{Ci}\cdot\text{h}} \cdot 6.5\times10^{-6} = 5.27\times10^{-9}\,\frac{\text{Gy}}{\text{Bq}\cdot\text{d}}.$$

The dose to the kidney from the mercury within the kidney is calculated from Eq. (6.97):

$$D(r_T \leftarrow r_S) = \tilde{A}_S \cdot S(r_T \leftarrow r_S)$$

$$D(\text{kidney} \leftarrow \text{kidney}) = \left[1.65\times10^7\,\text{Bq}\cdot\text{d}\right] \cdot \left[5.27\times10^{-9}\,\frac{\text{Gy}}{\text{Bq}\cdot\text{d}}\right] = 8.7\times10^{-2}\,\text{Gy}.$$

The contribution of the ^{203}Hg distributed throughout the rest of the body to the kidney dose is now calculated by multiplying the cumulated body activity, $\tilde{A}$, by the value S(kidney ← body) from Table 6–7. The cumulated activity in the body is

$$\tilde{A}(\text{body}) = \frac{A_{b_1}(0)}{\lambda_{E_1}} + \frac{A_{b_2}(0)}{\lambda_{E_2}}$$

$$\tilde{A}(\text{body}) = \frac{(0.95)\cdot5.75\times10^6\,\text{Bq}}{3.2\times10^{-2}\,\text{d}^{-1}} + \frac{(0.05)\cdot5.75\times10^6\,\text{Bq}}{1.47\times10^{-2}\,\text{d}^{-1}} = 1.903\times10^8\,\text{Bq}\cdot\text{d}.$$

The S (kidneys ← body) value from Table 6-7 is 6.1×10^{-6} rad/μCi · h. This value is converted to SI units with Eq. (6.100):

$$\frac{\text{Gy}}{\text{Bq}\cdot\text{d}} = \frac{\text{rad}}{\mu\text{Ci}\cdot\text{h}} \cdot 6.5\times10^{-6}$$

$$= \left(6.1\times10^{-6}\right)\cdot6.5\times10^{-6} = 3.97\times10^{-11}\,\frac{\text{Gy}}{\text{Bq}\cdot\text{d}}.$$

The dose to the kidneys from the mercury distributed throughout the rest of the body is calculated from Eq. (6.97):

$$D(r_T \leftarrow r_S) = \tilde{A}_S \cdot S(r_T \leftarrow r_S)$$

$$D(\text{kidneys} \leftarrow \text{body}) = \tilde{A}(\text{body}) \cdot S(\text{kidneys} \leftarrow \text{body})$$

$$= \left(1.903 \times 10^8 \,\text{Bq} \cdot \text{d}\right) \cdot 3.97 \times 10^{-11} \frac{\text{Gy}}{\text{Bq} \cdot \text{d}}$$

$$= 7.555 \times 10^{-3} \,\text{Gy}.$$

The total dose to the kidneys is the sum of the dose from each of the two sources:

$$D(\text{kidneys}) = D(\text{kidneys} \leftarrow \text{kidneys}) + D(\text{kidneys} \leftarrow \text{body})$$

$$D(\text{kidneys}) = 8.72 \times 10^{-2} \text{ Gy} + 7.555 \times 10^{-3} \text{ Gy} = 9.48 \times 10^{-2} \text{ Gy (9.48 rads)}.$$

Equation (6.99) tells us that the cumulated activity in an organ or tissue is given by

$$\tilde{A} = \frac{A_{s_1}(0)}{\lambda_{E_1}} + \frac{A_{s_2}(0)}{\lambda_{E_2}} + \cdots + \frac{A_{s_n}(0)}{\lambda_{E_n}}$$

and Eq. (6.97) tells us that the dose to any organ is given by the product of the time integrated activity (cumulated activity) in the source organ and the appropriate S factor:

$$D(r_T \leftarrow r_S) = \tilde{A}_S \cdot S(r_T \leftarrow r_S).$$

If we substitute the expression for $\tilde{A}$ from Eq. (6.99) into Eq. (6.97), we get

$$D(r_T \leftarrow r_S) = \left(\frac{A_{Sc_1}(0)}{\lambda_{Ec_1}} + \frac{A_{Sc_2}(0)}{\lambda_{Ec_2}} + \cdots + \frac{A_{Sc_n}(0)}{\lambda_{Ec_n}} \right) \cdot S(r_T \leftarrow r_S)$$

$$= \sum \frac{A_{c_i}}{\lambda_{Ec_i}} \cdot S(r_T \leftarrow r_S). \tag{6.101}$$

In Chapter 4, we showed that the average life of a radioisotope is simply the reciprocal of the transformation rate constant. Since the clearance of internally deposited radionuclides follows the same kinetics as radioactive transformation, it follows that the mean residence time of an internally deposited radionuclide is given by the reciprocal of the effective clearance rate constant, or its equivalent, 1.44 times the effective half-life:

$$\tau = \frac{1}{\lambda_E} = 1.44 \, T_E. \tag{6.102}$$

If the expression for $1/\lambda_E$ from Eq. (6.102) is substituted into Eq. (6.101), we obtain

$$D(r_T \leftarrow r_S) = \sum \tau_{Sci} A_{Si}(0) \cdot S(r_T \leftarrow r_S), \tag{6.103}$$

where τ_{Sci} is the residence time in the ith compartment of organ S.

The use of the mean residence time may be illustrated by the example that follows below.

EXAMPLE 6-19

Calculate the dose from an accidental intake of ^{137}Cs that led to an initial body burden of 1 MBq (which was determined by whole-body counting).

Solution

The retention curve[6] for ^{137}Cs is given by Eq. (6.59) as

$$q(\mathrm{t}) = 0.1 \cdot q_0 e^{-\left(\frac{0.693}{2\ \mathrm{d}} \cdot t\right)} + 0.9 \cdot q_0 e^{-\left(\frac{0.693}{110\ \mathrm{d}} t\right)},$$

and MIRD Pamphlet No. 11 (p. 209) gives the S(body ← body) value for ^{137}Cs together with its very short-lived daughter $^{137\mathrm{m}}$Ba as 1.4×10^{-5} rad/μCi · h (3.8×10^{-6} Gy/MBq h).

The mean residence time τ_1 in compartment 1 is calculated as follows:

$$\tau_1 = 1.44 \cdot 2\ \mathrm{days} = 2.88\ \mathrm{days}.$$

Since compartment 1 contained 10% of the activity, the time integrated (cumulated) activity in compartment 1 is

$$\tilde{A}_1 = 0.1 \cdot (1\ \mathrm{MBq}) \cdot 2.88\ \mathrm{d} \cdot \frac{24\ \mathrm{h}}{1\ \mathrm{d}} = 6.912\ \mathrm{MBq} \cdot \mathrm{h}.$$

The dose due to compartment 1 is calculated with Eq. (6.97):

$$D_1 = (6.912\ \mathrm{MBq} \cdot \mathrm{h}) \cdot 3.8 \times 10^{-6}\ \frac{\mathrm{Gy}}{\mathrm{MBq} \cdot \mathrm{h}} = 2.6 \times 10^{-5}\ \mathrm{Gy}.$$

For compartment 2, the mean residence time, τ_2, is

$$\tau_2 = 1.44 \cdot 110\ \mathrm{days} = 158.4\ \mathrm{days}$$

and

$$\tilde{A}_2 = 0.9 \cdot (1\ \mathrm{MBq}) \cdot 158.4\ \mathrm{d} \cdot \frac{24\ \mathrm{h}}{1\ \mathrm{d}} = 3.42 \times 10^3\ \mathrm{MBq} \cdot \mathrm{h}.$$

The dose from the radiocesium in compartment 2 is

$$D_2 = (3.42 \times 10^3\ \mathrm{MBq} \cdot \mathrm{h}) \cdot 3.8 \times 10^{-6}\ \frac{\mathrm{Gy}}{\mathrm{MBq} \cdot \mathrm{h}} = 1.3 \times 10^{-2}\ \mathrm{Gy}.$$

[6]For details on the biokinetic model, see ICRP, 1990. Age-dependent Doses to Members of the Public from Intake of Radionuclides—Part 1. ICRP Publication 56. *Ann. ICRP* 20(2).

The total dose is the sum of the doses from the two compartments:

$$D = D_1 + D_2 = \left[2.6\times10^{-5}\,\text{Gy}\right] + \left[1.3\times10^{-2}\,\text{Gy}\right] = 1.3\times10^{-2}\,\text{Gy}$$
$$= 13\,\text{mGy}\,(1.3\times10^{3}\ \text{mrads}).$$

ICRP Methodology

The MIRD methodology was developed to calculate doses from radionuclides that are administered for medical purposes. The ICRP methodology was developed to calculate doses from internally deposited radionuclides for health physics purposes. Since the objective for both methods is the same, namely to calculate a dose from an internal emitter, it is not surprising that the two methods are essentially the same. However, although they come to the same end point, the two computational methodologies use different terminology in constructing their formulations. Another difference between the two, which in most cases is trivial, is that the MIRD formulation calculates the dose in gray by integrating the dose rate over an infinitely long time after intake of the radionuclide, while the ICRP formulation calculates the equivalent dose in sievert (or rems when the traditional units are used) accumulated during 50 years after the intake for adults, and 70 years for children.

The dose to a single target from a single radionuclide in a single source organ (for multiple sources and multiple radionuclides, we merely sum the contributions of each source and each radionuclide) is given by MIRD, Eq. (6.97), as

$$D(r_T \leftarrow r_S) = \tilde{A}_S \times S(r_T \leftarrow r_S).$$

The ICRP formulation is given as

$$H_{50,T}\left(T \leftarrow S\right) = \left(U_S\,\text{transf.}\right)\cdot SEE\left(T \leftarrow S\right)\frac{\text{Sv}}{\text{trans.}}, \tag{6.104}$$

where

$H_{50,T}\,(T \leftarrow S)$ is the equivalent dose accumulated during 50 years after intake, U_S is the total number of disintegrations during 50 years after intake, and $SEE\,(T \leftarrow S)$ is the specific effective energy absorbed per gram of target tissue from each radiation, R, emitted from activity in the source organ.

It is calculated from the following equation:

$$SEE\left(T \leftarrow S\right)$$
$$= \sum_{\text{R}} \frac{Y_{\text{R}}\,\dfrac{\text{particles}}{\text{trans.}}\cdot\left(E\cdot w_{\text{R}}\right)\dfrac{\text{effective MeV}}{\text{particle}}\cdot 1.6\times10^{-13}\,\dfrac{\text{J}}{\text{MeV}}\cdot AF\left(T \leftarrow S\right)_{\text{R}}}{m_{\text{T}}\ \text{kg}}\cdot\frac{1\ \text{Sv}}{\text{eff. J}}\cdot\frac{}{\text{trans.}}$$
$$= \frac{\text{Sv}}{\text{trans}}, \tag{6.105}$$

where

Y_R = fractional yield, per disintegration, of the radiation under consideration,
w_R = radiation weighting factor (formerly symbolized by Q),
w_R = 1 for gamma and beta radiation, and 20 for alphas,
$AF(T \leftarrow S)_R$ = absorbed fraction from radiation R in T per transformation in S, and[7]
m_T is the mass of T, in kg.

Comparison of Eqs. (6.97) and (6.104) shows that the MIRD and ICRP equations are essentially the same. There are two differences between them: One is that the MIRD is the total lifetime dose, expressed in Gy (or rads in traditional units), while the ICRP dose is the equivalent dose, expressed in Sv (or rems in traditional units) for the 50-year period following intake. The equations show that the ICRP analogs of the MIRD formulation are the following:

MIRD	ICRP
$D(r_T \leftarrow r_S)$, total lifetime dose, Gy	$H_{50,T}(T \leftarrow S)$ 50-yr equivalent dose, Sv
$\tilde{A}$, time integrated activity, Bq · d	U_S, total number of transformations in 50 yrs
$S(r_T \leftarrow r_S), \frac{\text{Gy}}{\text{Bq} \cdot \text{d}}$	$SEE(T \leftarrow S), \frac{\text{Sv}}{\text{trans.}}$

The application of the ICRP notation is illustrated by the following example.

EXAMPLE 6-20

Ten percent of the activity in inhaled soluble ^{210}Po particles is transferred to the blood ($f_1 = 0.1$) in a worker. Of the polonium transferred to the blood, the distribution is as follows[8]:

ORGAN	FRACTION DISTRIBUTION	BIOLOGICAL HALF-LIFE
Liver	0.3	50 d
Kidney	0.1	50 d
Spleen	0.05	50 d
Red bone marrow	0.1	50 d
Other tissues	0.45	50 d

Calculate the dose to the spleen ($m = 150$ g) due to the inhalation of 1 Bq ^{210}Po particles.

[7]Tables of specific absorbed fractions may be found in M. Cristy and K. Eckerman, ORNL/NUREG/TM 8381.
[8]ICRP, 1993. Age-dependent Doses to Members of the Public from Intake of Radionuclides—Part 2, Ingestion Dose Coefficients. ICRP Publication 67. *Ann. ICRP* 23(3–4).

Solution

^{210}Po emits one 5.3-MeV alpha particle per transformation, in which 5.4 MeV are dissipated (5.3 MeV plus 0.1 MeV ^{206}Pb daughter recoil energy).

The effective half-life of ^{210}Po is (Eq. 6.54)

$$T_E = \frac{T_B \cdot T_R}{T_B + T_R} = \frac{50\,\text{d} \cdot 138\,\text{d}}{50\,\text{d} + 138\,\text{d}} = 36.7\ \text{d},$$

and the effective clearance rate constant is $\lambda_E = 0.693/T_E = 0.019$ per day.

Inhalation of 1 Bq ^{210}Po leads to a deposition in the spleen of

$$q\,(\text{spleen}) = 0.1 \cdot (1\text{Bq}) \cdot 0.05 = 0.005\ \text{Bq}.$$

The total number of transformations in the spleen, U_S, is found with the aid of Eq. (6.90):

$$U_S = \frac{A_S(0)}{\lambda_E} = \frac{0.005\,\text{Bq} \cdot 1\,\frac{\text{tps}}{\text{Bq}} \cdot 8.64 \times 10^4\,\frac{\text{s}}{\text{d}}}{0.019\ \text{d}^{-1}} = 2.3 \times 10^4\,\text{transformations}.$$

Since we are dealing with alpha particles, the absorbed fraction (AF) is equal to 1. The equivalent dose to the spleen due to the ^{210}Po in the spleen is calculated with Eq. (6.105):

$$H_{50}(T \leftarrow S)_R = U_S \frac{Y_R \cdot (E \cdot w_R)\,\text{AF}(T \leftarrow S)_R}{m_T} \cdot 1.6 \times 10^{-13} \cdot \frac{1\,\text{Sv}}{\text{J/kg}}.$$

$$H_{50}(\text{spleen} \leftarrow \text{spleen})$$

$$= 2.3 \times 10^4\,\text{trans.} \cdot \frac{\left(5.4\,\frac{\text{MeV}}{\alpha} \cdot 20\right) \cdot 1\,\frac{\alpha}{\text{trans.}} \cdot 1.6 \times 10^{-13}\,\frac{\text{J}}{\text{MeV}}}{0.15\ \text{kg}} \cdot \frac{1\,\text{Sv}}{\text{J/kg}}$$

$$= 2.6 \times 10^{-6}\,\text{Sv}.$$

The committed equivalent dose to the spleen is thus found to be 2.6×10^{-6} Sv Bq^{-1} of ^{210}Po inhaled.

EXTERNAL EXPOSURE: NEUTRONS

Exposure to neutrons is always from an external source. However, because one aspect of neutron-dose calculation simulates the dose from a uniformly distributed radionuclide, discussion was deferred until this point in the chapter.

Fast Neutrons

The absorbed dose from a beam of neutrons may be computed by considering the energy absorbed by each of the tissue elements that react with the neutrons. The type of reaction, of course, depends on the neutron energy. For fast neutrons up to about 20 MeV, the main mechanism of energy transfer is elastic collision. Thermal neutrons may be captured and initiate nuclear reactions. In cases of elastic scattering of fast neutrons, the scattered nuclei dissipate their energy in the immediate vicinity of the primary neutron interaction. The radiation dose absorbed locally in this way is called the *first collision dose* and is determined entirely by the primary neutron flux; the scattered neutron is not considered after this primary interaction. For fast neutrons, the first collision dose rate from neutrons of energy E is

$$\dot{D}_n(E) = \frac{\phi(E)E\sum_i N_i\sigma_i f_i}{1\dfrac{\text{J/kg}}{\text{Gy}}}, \tag{6.106}$$

where

$\phi(E)$ = flux of neutrons whose energy is E, in neutrons per cm^2 per second,
N_i = number of atoms per kilogram of the ith element,
σ_i = scattering across section of the ith element for neutrons of energy E, in barns $\times 10^{-24}$ cm^2, and
f = mean fractional energy transferred from neutron to scattered atom during collision with the neutron.

For isotropic scattering, the average fraction of the neutron energy transferred in an elastic collision with a nucleus of atomic mass number M is

$$f = \frac{2M}{(M+1)^2}. \tag{6.107}$$

The composition of soft tissue, for the purpose of radiation dosimetry, is given in Table 6-9. Table 6-10 lists the average fraction of the neutron energy transferred to each of the tissue elements.

TABLE 6-9 Synthetic Tissue Composition, Fraction of Total Mass

ELEMENT	ATOMIC NUMBER	ICRP SOFT TISSUE	ADIPOSE TISSUE (ICRP)	MUSCLE, SKELETAL (ICRP)	MUSCLE STRIATED, (ICRU)	ICRP 23	MIRD	ICRU 44 FOUR COMPONENT
H	1	0.104472	0.119477	0.100637	0.101997	0.10454	0.1	0.101172
C	6	0.23219	0.63724	0.10783	0.123	0.22663	0.1489	0.111
N	7	0.02488	0.00797	0.02768	0.035	0.0249	0.0347	0.026
O	8	0.630238	0.232333	0.754773	0.729003	0.63525	0.7139	0.761828
Na	11	0.00113	0.0005	0.00075	0.0008	0.00112	0.0015	—
Mg	12	0.00013	0.00002	0.00019	0.0002	0.00013	—	—
P	15	0.00133	0.00016	0.0018	0.002	0.00134	—	—

(*continued*)

TABLE 6-9 Synthetic Tissue Composition, Fraction of Total Mass (*Continued*)

ELEMENT	ATOMIC NUMBER	ICRP SOFT TISSUE	ADIPOSE TISSUE (ICRP)	MUSCLE, SKELETAL (ICRP)	MUSCLE STRIATED, (ICRU)	ICRP 23	MIRD	ICRU 44 FOUR COMPONENT
S	16	0.00199	0.00073	0.00241	0.005	0.00204	—	—
Cl	17	0.00134	0.00119	0.00079	—	0.00133	0.001	—
K	19	0.00199	0.00032	0.00302	0.003	0.00208	—	—
Ca	20	0.00023	0.00002	0.00003	—	0.00024	—	—
Fe	26	0.00005	0.00002	0.00004	—	0.00005	—	—
Zn	30	0.00003	0.00002	0.00005	—	0.00003	—	—
Density		1	0.92	1.04	1.04	1.04	1	1

Sources: ICRU 44: Tissue Substitutes in Radiation Dosimetry and Measurements, ICRU 1989 Report 44; ICRP 23: Report on the Task Group on Reference Man ICRP 1975 Publication 23; ICRU: Photon, Electron, Proton and Neutron Interaction Data for Body Tissues (Report 46).

TABLE 6-10 Atoms Per Unit Mass and Average Fraction of Neutron Energy Transferred

ELEMENT	FRACTION OF TISSUE MASS	ATOMIC MASS	*N* (atoms/kg)	*f*
H	0.1	1	6.02×10^{25}	0.5
C	0.1489	12	7.47×10^{24}	0.142
N	0.0347	14	1.49×10^{24}	0.124
O	0.7139	16	2.69×10^{25}	0.111
Na	0.0015	23	3.93×10^{22}	0.08
Cl	0.001	35.45	1.70×10^{22}	0.053

Tissue mass fractions used are from MIRD Pamphlet 3.

EXAMPLE 6-21

What is the absorbed dose rate to soft tissue in a beam of 5-MeV neutrons whose intensity is 2000 neutrons/cm^2/second?

Solution

The scattering cross sections of each of the tissue elements for 5-MeV neutrons are listed below. The scattering cross section for chlorine was computed using the abundance multiplied by the cross section. Cl-36 is 1.93 b with 76% abundance and Cl-37 is 2.25 b with 24% abundance, so the total chlorine cross section is

$$1.93(0.76) + 2.25(0.24) = 1.69 \text{ b}.$$

The neutron scattering values were obtained from Brookhaven National Labs National Nuclear Data Center Evaluated Nuclear Data File (ENDF)[9].

[9]http://www.nndc.bnl.gov/exfor/endf00.jsp

ELEMENT	N	σ, $\times 10^{-24}$, cm^2	f	$N_i\sigma_i f_i$
H	6.02×10^{25}	0.9	0.500	27.09
C	7.47×10^{24}	1.12	0.142	1.188
N	1.49×10^{24}	1. 2	0.124	0.301
O	2.69×10^{25}	0.78	0.111	2.32
Na	3.93×10^{22}	0.9	0.080	0.003
Cl	1.7×10^{22}	1.69	0.053	0.002

$$\sum N_i \sigma_i f_i = 30.9 \frac{\text{cm}^2}{\text{kg}}$$

Substituting the appropriate values into Eq. (6.106) yields

$$\dot{D}_n = \frac{2\times10^3 \frac{\text{neutrons}}{\text{cm}^2\cdot\text{s}} \cdot 5 \frac{\text{MeV}}{(\text{neutron})} \cdot 1.6\times10^{-13} \frac{\text{J}}{\text{MeV}} \cdot 30.9 \frac{\text{cm}^2}{\text{kg}}}{1\frac{\text{J/kg}}{\text{Gy}}}$$

$$= 4.9\times10^{-8} \frac{\text{Gy}}{\text{s}} \left(4.17\times10^{-6} \frac{\text{rad}}{\text{s}}\right),$$

$$4.9\times10^{-8} \frac{\text{Gy}}{\text{s}} \cdot 10^6 \frac{\mu\text{Gy}}{\text{Gy}} \cdot 3.6\times10^3 \frac{\text{s}}{\text{h}} = 178 \frac{\mu\text{Gy}}{\text{h}} \quad \left(17.8 \frac{\text{mrads}}{\text{h}}\right).$$

In the example above, the neutron beam was monoenergetic and thus only one neutron energy was considered. If a beam contains neutrons of several energies, then the calculation must be carried out separately for each energy group.

Thermal Neutrons

For thermal neutrons, two reactions are considered, namely, the $^{14}\text{N}(n, p)^{14}\text{C}$ reaction and the $^{1}\text{H}(n, \gamma)^{2}\text{H}$ reaction. For the former reaction, the dose rate may be calculated from the equation

$$\dot{D}_{n,p} = \frac{\phi N_\text{N} \sigma_\text{N} Q \cdot 1.6\times10^{-13} \frac{\text{J}}{\text{MeV}}}{1\frac{\text{J/kg}}{\text{Gy}}}, \tag{6.108}$$

where

ϕ = thermal flux, neutrons/cm^2/s,
N_N = number of nitrogen atoms per kg tissue, 1.49×10^{24}
σ_N = thermal absorption cross section for nitrogen, 1.568×10^{-24} cm^2, and
Q = energy released by the reaction = 0.63 MeV.

The latter reaction, $^1H(n, \gamma)^2H$, is equivalent to having a uniformly distributed gamma-emitting isotope throughout the body and results in an auto-integral gamma-ray dose. The specific activity of this distributed gamma emitter, the number of reactions per second per gram, is governed by the neutron flux and is given by Eq. (6.109):[10]

$$A = \phi N_H \sigma_H \frac{\text{“Bq”}}{\text{kg}}, \tag{6.109}$$

where

ϕ = thermal flux, neutrons/cm^2/s,
N_H = number of hydrogen atoms per kg tissue = 6.02×10^{25}, and
σ_H = absorption cross section for hydrogen = 0.332×10^{-24} cm^2.

EXAMPLE 6-22

What is the absorbed dose rate to a 70-kg person from a whole body exposure to a mean thermal flux of 10,000 neutrons/cm^2/s?

Solution

The dose rate due to the *n*, *p* reaction is calculated from Eq. (6.108):

$$\dot{D}_{n,p} = \frac{10^4 \frac{\text{neutrons}}{\text{cm}^2 \cdot \text{s}} \cdot 1.49 \times 10^{24} \frac{\text{atoms}}{\text{kg}} \cdot 1.568 \times 10^{-24} \frac{\text{cm}^2}{\text{atom}} \cdot (0.63 \text{ MeV}) \cdot 1.6 \times 10^{-13} \frac{\text{J}}{\text{MeV}}}{1 \frac{\text{J/kg}}{\text{Gy}}}$$

$$\dot{D}_{n,p} = 2.36 \times 10^{-9} \frac{\text{Gy}}{\text{s}}.$$

The dose rate per hour is then

$$\dot{D}_{n,p} = 2.36 \times 10^{-9} \frac{\text{Gy}}{\text{s}} \cdot \frac{3600 \text{ s}}{\text{h}} = 8.5 \frac{\mu\text{Gy}}{\text{h}} \left(0.85 \frac{\text{mrad}}{\text{h}}\right) \text{ from the } n, p \text{ reaction.}$$

The gamma ray “activity,” from the $^1H(n, \gamma)^2H$ interaction is calculated using Eq. (6.109):

$$A = 10^4 \frac{\text{neutrons}}{\text{cm}^2 \cdot \text{s}} \cdot 6.02 \times 10^{25} \frac{\text{atoms}}{\text{kg}} \cdot 0.332 \times 10^{-24} \frac{\text{cm}^2}{\text{atom}} = 2.0 \times 10^5 \frac{\text{“Bq”}}{\text{kg}}.$$

[10]Since the ^1H atom is not actually emitting the prompt gamma as a result of a decay, but as a result of a capture of the neutron, we have used “Bq” to represent the prompt capture gamma emission rate from ^1H.

The dose rate from this uniformly distributed gamma-ray emitter is calculated from Eq. (6.82):

$$\dot{D}_\gamma = \frac{A_S}{m} \cdot \varphi \cdot \Delta \frac{\text{Gy}}{\text{s}}.$$

The absorbed fraction, φ, for the 2.23-MeV gamma ray is found, by interpolating in Table 6-6 between the 2 and 4 MeV values, to be 0.278, and Δ, the dose rate in an infinitely large mass whose specific activity is $1\frac{\text{Bq}}{\text{kg}}$, is calculated from Eq. (6.81):

$$\Delta = 1.6\times10^{-13} \cdot 2.23\frac{\text{MeV}}{\gamma} = 3.57\times10^{-13}\frac{\text{Gy/s}}{\text{Bq/kg}}.$$

The auto-integral gamma-ray dose rate, therefore, is

$$\dot{D}_\gamma = 2.0\times10^{5}\frac{\text{Bq}}{\text{kg}} \cdot (0.278) \cdot 3.57\times10^{-13}\frac{\text{Gy/s}}{\text{Bq/kg}}$$
$$= 1.98\times10^{-8}\frac{\text{Gy}}{\text{s}} \quad \left(1.98\times10^{-6}\frac{\text{rad}}{\text{s}}\right)$$

or

$$71\frac{\mu\text{Gy}}{\text{h}} \quad \left(7.1\frac{\text{mrad}}{\text{h}}\right).$$

We cannot, in this case, add the auto-integral gamma-ray dose to the dose from the *n*, *p* reaction because an absorbed dose of 1 Gy of gamma radiation is not biologically equivalent to 1 Gy from proton radiation. This point, which deals with the relative biological effectiveness of the various radiations, is discussed in the next chapter.

SUMMARY

When ionizing radiation interacts with any medium (air, tissue, water, plastic, etc.), energy is transferred from the radiation field to the medium. The quantity that describes this energy transfer is the absorbed dose and is measured by the concentration of absorbed energy. Traditionally, this quantity was called a *rad*, and 1 rad was defined as the *absorption of 100 ergs of energy per gram of irradiated medium*. In the SI system, the quantity for absorbed dose is the *gray* (Gy); 1 gray is defined as an *absorption of one joule of energy per kilogram*: 1 Gy = 100 rads. Absorbed dose is a macroscopic quantity, and it applies to the average amount of energy absorbed per unit mass of absorbing medium.

The first quantitative unit that was used for radiation dose was the roentgen (R). Technically, the roentgen was a dose unit *only for air*; hence, it is a measure of X- or gamma-ray exposure, not radiation dose. However, it continues to be useful in radiation protection because an exposure of 1 R leads to a dose of approximately 1 rad to soft tissue. In SI units,

exposure to X- or gamma radiation is measured in coulombs of ions produced by the radiation per kilogram of air: 1 R = 2.58×10^{-4} C/kg.

Absorbed dose is defined in the same way for external radiation as it is for the dose from internally deposited radioisotopes. For external radiation, the dose can be measured. For internally deposited radionuclides, the dose cannot be measured. The dose from an internally deposited radionuclide is calculated with the aid of a physiologically based biokinetic dosimetric model for that radionuclide that considers the radiation characteristics of the nuclide and the biological characteristics of the deposited radionuclides. This method is formalized in the MIRD system for *m*edical *i*nternal *r*adiation *d*osimetry, where the calculation is based on the fraction of the energy emitted by the radionuclide that is absorbed in the tissue of interest and the mean residence time of the radioactivity in the organs or tissues. A parallel computational methodology for calculating internal dose for health physics purposes was developed by the ICRP. The ICRP formulation is based on calculating the energy absorbed per kg of irradiated target tissue from one decay of a radionuclide deposited in a source organ and then multiplying this by the total number of decays in the source organ during the 50-year period following intake of the radionuclide.

PROBLEMS

6.1. A 50-μC/kg (approximately 200 mR) pocket dosimeter with air-equivalent walls has a sensitive volume with the dimensions 0.5 in. (diameter) and 2.5 in. (length); the volume is filled with air at atmospheric pressure. The capacitance of the dosimeter is 10 pF. If 200 V are required to charge the chamber, what is the voltage across the chamber when it reads 50 μC/kg (~200 mR)?

6.2. An air ionization chamber whose volume is 1 L is used as an environmental monitor at a temperature of 27°C and a pressure of 700 torr. What is the exposure rate, in μC/kg/h and in mR/h, if the saturation current is 10^{-13} A?

6.3. A beam of 1-MeV gamma rays and another of 0.1-MeV gamma rays each produce the same ionization density in air. What is the ratio of 1:0.1 MeV photon flux?

6.4. Assuming that the specific heat of the body is 1 cal/g/°C, calculate the temperature rise due to a total body dose of 5 Gy.

6.5. Compute the dose rate, in mGy/h, at a distance of 50 cm from a small vial containing 10 mL of an aqueous solution of

(a) 2 GBq (54.1 mCi) ^{51}Cr

(b) 2 GBq (54.1 mCi) ^{24}Na, based on the transformation schemes shown below: