CHAPTER 29

# IONIZING RADIATION

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## INTRODUCTION

### Definition and General Description

Ionizing radiation, in general, is any electromagnetic or particulate radiation capable of producing ions, directly or indirectly, by interaction with matter. In the specific situation being considered here, the industrial environment — and usually in considering radiation protection matters — that portion of the electromagnetic spectrum having frequencies in the ultraviolet portion and lower is excluded (see Chapter 28). Stated in more explicit terms, the International Commission on Radiation Units and Measurements (ICRU)[1] defines ionizing radiation as: "any radiation consisting of directly or indirectly ionizing particles or a mixture of both. Directly ionizing particles are charged particles (electrons, protons, alpha particles, etc.) having sufficient kinetic energy to produce ionization by collision. Indirectly ionizing particles are uncharged particles (neutrons, photons, etc.) which can liberate directly ionizing particles or can initiate nuclear transformations." Thus, ionizing radiation encompasses consideration both of atomic particles having a variety of physical and electrical characteristics streaming at velocities from nearly zero, to values approaching the speed of light, and of electromagnetic radiations (photons) having a wide range of energies streaming at the speed of light. Photons, which have no mass, are referred to as "particles" for theoretical reasons. In this chapter "radiation" implies "ionizing radiation."

In the industrial environment, the radiations of primary concern are: x, gamma, alpha, beta and neutron. X and gamma radiations may be called x rays and gamma rays. Also, alpha and beta radiations are called alpha and beta particles. Except for very small amounts from natural background radiation, proton and some other kinds of radiation are not of concern unless there is equipment designed to specifically produce them. Research facilities, such as large accelerators, are not discussed in this chapter.

X and gamma radiations both are penetrating electromagnetic radiations having wavelengths much shorter than that of visible light but they are of different origin. X rays originate in the extra nuclear part of the atom, whereas gamma rays are emitted from the nucleus in the process of nuclear transition or during particle annihilation. (Annihilation is the process by which a negative electron and a positive electron, called a positron, combine and disappear with emission of electromagnetic radiation.)

Ordinarily, useful x rays are produced in an evacuated tube by accelerating electrons from a heated filament to a metal target with voltages of 50 to 500 kilovolts (kV). Sometimes much higher or somewhat lower voltages are used. The electrons interact with orbital electrons of atoms in the target causing energy level changes that result in the emission of "characteristic" x rays, and also with the nucleus of the atom to produce electromagnetic radiation having a "continuous" spectrum (called bremsstrahlung).

All radionuclides undergo a spontaneous transformation, called decay, during which radiation is emitted and a new nuclide, called a daughter (or decay product) is formed. The radiations are of a specific type (or types) and energy, or energy distribution, for each species of radionuclide. Tabulated data for many radionuclides are presented in *Radiological Health Handbook*.[2]

Gamma rays are emitted by the nucleus of certain radionuclides during their decay. Each such radionuclide emits one or more gamma rays having a specific energy. Gamma rays also are produced by neutron interactions with nuclei.

Alpha radiation consists of a stream of alpha particles, each particle being physically identical to the helium nucleus — two neutrons and two protons. They are emitted spontaneously during the radioactive decay of certain radionuclides, primarily those of higher molecular weight — bismuth and higher. Because of the comparatively large size and double positive charge of the alpha particles, alpha radiation does not penetrate matter readily. The more energetic radiation is completely stopped by the skin. Inside the body, however, it produces dense ionization in tissues.

Beta radiation consists of a stream of beta particles, which are either electrons of negative charge or electrons of positive charge, called positrons, which have been emitted by an atomic nucleus — or by a neutron in the process of transformation. Radionuclides that spontaneously emit beta particles span the entire range of the elements. These nuclides emit particles having a maximum energy characteristic of that nuclide, along with many other particles of lower energy.

Neutron radiation consists of a stream (flow) of neutrons. Radionuclides do not emit neutrons spontaneously, although a small number of very heavy radionuclides fission spontaneously with the emission of neutrons. Neutron radiation is produced by various nuclear reactions, by nuclear fission and by interactions of alpha or gamma radiation with certain nuclei. Since neutrons are uncharged, the radiation readily penetrates matter. Neutrons decay into a proton and an electron with a half-life of 11.7 minutes. Neutron energies, expressed in electron volts (eV) or the multiples kiloelectron volts (keV) and megaelectron volts (MeV), span a very wide range of values, and are commonly classified into three general groups — slow, intermediate and fast. The range of energies for each of these general groups is indefinite, different ranges being selected according to specific needs. One such classification is <1 eV, 1 eV to 0.1 MeV and >0.1 MeV, respectively.[3] There are also more specific classes, e.g., thermal, which are those essentially in thermal equilibrium with the medium in which they exist (mean value 0.025 eV at 20°C).

**Quantities and Units**

In quantitating radioactive materials, a unique situation exists because one property of primary interest is continually changing. As decay takes place, the activity, or number of nuclear disintegrations occurring in a given quantity of material per unit time decreases exponentially. Therefore, a time dependent factor, half-life, becomes part of any quantitative evaluation. Each radionuclide has a definite half-life, however the range of half-lives for different radionuclides is very great, from fractions of a second to billions of years. Since the mass of material does not change significantly during this decay, the quantity of a radionuclide or of a radioactive material is usually specified in terms of its activity, with the exception that mass may be used in some situations (e.g., nuclear fuel manufacturing) where half-lives of the useful radionuclides are very long. Since activity in a given specimen (and the radiation from it) may come from one or more radionuclides, each decaying exponentially, the composition and activity must be specified as of a definite date, the accuracy (year, month, day, minute, second) depending on the relation of the half-lives to previous or subsequent periods of interest. The activity is commonly expressed in curies (or its multiples) although for some measurements, disintegrations per minute (dpm) or per second (dps) are commonly used. One curie equals 3.7 x $10^{10}$ disintegrations per second.

Another unique and complex situation exists in quantitating the effect of radiation on living organisms. The different kinds of radiation interact in a wide variety of ways both with living organisms (e.g., body tissues) and inanimate things (e.g., shielding). Furthermore, the interactions may be different for different energies of the same type of radiation; and the spatial distribution of the interaction is not uniform. They all, however, impart energy to matter through which they pass; and for living organisms the absorbed dose — energy imparted in a volume element divided by the mass of irradiated material in that volume element — provides a common base for considering the degree of effect produced by specific amounts of any of the different types of radiation. The unit of absorbed dose is the rad. One rad equals 100 ergs per gram.

The biological effect for equal absorbed doses from different types and energies of radiation, however, is not constant. Therefore dose equivalent, which is the absorbed dose modified by pertinent factors, particularly the "quality factor," is used for radiation protection evaluations to take into account the difference in the biological effect of the different radiations. Values of the quality factor for commonly encountered radiations, suitable for general use, have been determined (see page 391). The special unit of dose equivalent is the rem, which is the product of absorbed dose in rads and the applicable quality factor. For special situations, a factor in addition to the quality factor may be used. Dose-limiting recommendations are expressed as maximum permissible dose equivalent in rems, commonly called "maximum permissible dose (MPD)." A similar concept is expressed as a Radiation Protection Guide (RPG) by the Federal Radiation Council.[4]

In some situations, it may be convenient and sufficiently accurate to express dose-limiting recommendations for x and gamma radiation (or to make related measurements) in terms of ionization in air, at the point of interest. The measure of ionization produced in air by x or gamma radiation is called "exposure," its special unit being the roentgen (R). It is customary, for radiation protection purposes, to consider that one R at the point of interest would be equivalent to a dose equivalent of one rem.

A comprehensive collection of data, graphs and tables will be found in the *Radiological Health Handbook.*[2] It should prove to be a useful adjunct to this chapter, since extensive tables and graphs are not included here.

**Glossary**

This glossary includes a limited number of terms used in radiation protection practice. These definitions are mostly from reference (3) which contains definitions of other terms. References (2) and (3) also include pertinent definitions.

*activity (A).* The number of nuclear disintegrations occurring in a given quantity of material per unit time.

*body burden.* The total quantity of a radionuclide present in the body.

*body burden, maximum permissible.* That body burden of a radionuclide which, if maintained at a constant level, would produce the maximum permissible dose equivalent in the critical organ.

*bremsstrahlung.* The electromagnetic radiation associated with the deceleration of charged particles. The term is also applied to the radiation associated with the acceleration of charged particles.

*controlled area.* A specified area in which exposure of personnel to radiation or radioactive material is controlled and which is under the supervision of a person who has knowledge of the appropriate radiation protection practices, including pertinent regulations, and who has responsibility for applying them.

*curie (Ci).* The special unit of activity. One curie equals 3.7 x $10^{10}$ disintegrations per second exactly. By popular usage, the quantity of any radioactive material having an activity of one curie.

*daughter*. A nuclide, stable or radioactive, formed by radioactive decay. A synonym for decay product.

*dose*. A general term denoting the quantity of radiation or energy absorbed in a specified mass. For special purposes, its meaning should be appropriately stated, e.g., absorbed dose.

*dose, absorbed*. The energy imparted to matter in a volume element by ionizing radiation divided by the mass of irradiated material in that volume element.

*dose equivalent*. The product of absorbed dose, quality factor, and other modifying factors necessary to express on a common scale, for all ionizing radiations, the irradiation incurred by exposed persons.

*dose equivalent, maximum permissible (MPD)*. The largest dose equivalent received within a specified period which is permitted by a regulatory agency or other authoritative group on the assumption that receipt of such dose equivalent creates no appreciable somatic or genetic injury. Different levels of MPD may be set for different groups within a population. (By popular usage, dose, maximum permissible, is an accepted synonym.)

*exposure*. A measure of the ionization produced in air by x or gamma radiation. It is the sum of the electrical charges on all of the ions of one sign produced in air when all electrons liberated by photons in a volume element of air are completely stopped in the air, divided by the mass of the air in the volume element.

*genetically significant dose (GSD)*. The dose which, if received by every member of the population, would be expected to produce the same total genetic injury to the population as do the actual doses received by the various individuals.

*half-life, radioactive*. For a single radioactive decay process, the time required for the activity to decrease to half its value by that process.

*half-value layer*. The thickness of a specified substance which, when introduced into the path of a given beam of radiation, reduces the value of a specified radiation quantity by one-half. It is sometimes expressed in terms of mass per unit area.

*isotopes*. Nuclides having the same atomic number but different mass numbers. NOTE: this term is often used inaccurately as a synonym for nuclide.

*nuclide*. A species of atom characterized by its mass number, atomic number, and energy state of the nucleus, provided that the mean life in that state is long enough to be observable.

*quality factor*. A linear energy transfer dependent factor by which absorbed doses are to be multiplied to obtain the dose equivalent.

*rad*. The special unit of absorbed dose. One rad equals 100 ergs per gram.

*radiation source*. An apparatus or a material emitting or capable of emitting ionizing radiation.

*Radiation Protection Guide (RPG)*. The radiation dose which should not be exceeded without careful consideration of the reasons for doing so; every effort should be made to encourage the maintenance of radiation doses as far below this guide as practicable.

*Radioactivity Concentration Guide (RCG)*. The concentration of radioactivity in the environment which is determined to result in organ doses equal to the Radiation Protection Guide.

*roentgen (R)*. The special unit of exposure. One roentgen equals $2.58 \times 10^{-4}$ coulomb per kilogram of air.

*sealed source*. A radioactive source sealed in a container or having a bonded cover, where the container or cover has sufficient mechanical strength to prevent contact with and dispersion of the radioactive material under the conditions of use and wear for which it was designed.

## PHYSICAL ASPECTS OF IONIZING RADIATION

### Electromagnetic Radiation

In the electromagnetic spectrum, gamma radiation spans an energy range from approximately $8 \times 10^{3}$ eV to $10^{7}$ eV, the corresponding frequencies being $2 \times 10^{18}$ to $2.5 \times 10^{21}$ hertz. X rays span a somewhat wider range of values, although there is no clear break at the lower energy boundary and at higher energies special equipment, such as an accelerator, is used for their production.

A beam of x rays from x-ray equipment encompasses a range of energies. The highest photon energy in the beam corresponds to the electron accelerating voltage, with the median being considerably below this value. The beam will include both photons having energies which are "characteristic" of the target material and photons having a continuous spectrum (bremsstrahlung), the proportion of the latter being greater at higher electron accelerating voltages. The energy spectrum, or quality of the beam, may be expressed either in terms of an "effective energy" or in terms of its half-value layer. The accelerating voltage may be constant or may come from a pulsating generator, which influences the photon energy distribution, the latter being designated in terms of peak voltage (kVp). Ordinarily, x-ray tubes and their housings are arranged so that there is shielding in all directions except for a "window" where the useful beam is emitted. The solid angle and shape of the useful beam is determined by the size of the window and by collimating devices, such as diaphragms and cones, made of shielding materials. Some low energy x rays are absorbed in the target, while others are removed from the useful beam by the material in the tube window and usually also by filters that preferentially absorb the less penetrating radiation. Accelerators used to produce high energy x rays are commonly arranged for beam emission to accomplish a specific purpose.

Sealed sources, consisting of a radionuclide encased in a metal capsule, are a common source of gamma radiation used in industry. They are used in radiography, measuring devices and a number of other special applications. The radionuclides in them are selected to provide radiation of the desired photon energy. Some emit photons of one energy, such as cesium-137 (.66MeV); others a range of energies, such as radium (.047 to 2.4 MeV due to retained daughters).

There are two basic processes by which electromagnetic radiation interacts with matter: scattering, in which the direction of the photon and its energy are altered; and absorption, in which the photon disappears with transfer of its energy to other radiations.

Along the path of a primary beam of photons, there are interactions between the electric fields of these photons and the electrons in the material being penetrated which cause "scattering" of some of the primary beam photons. Further reactions ensue with a resulting 360° angular distribution of scattered photons having a range of energies down to nearly zero. The shape of this angular, and associated energy, distribution is a function of the energy of the original photons. For a mathematical treatment of scattering, see *Principles of Radiation Protection.*[5]

Absorption of photons occurs primarily by three processes — the photoelectric effect, the Compton effect and pair production. These are also treated mathematically in the above book.[5] The photoelectric effect predominates for the lower energy photons, the Compton effect where the energy is greater than approximately 0.5 MeV, and for pair production a minimum of 1.02 MeV is required.

The photoelectric effect involves an interaction between incident photons and the electrons in the shells around the nuclei. Electrons are ejected from the atoms with an energy equal to the difference between the photon energy and the binding energy of the ejected electron. Subsequently, x rays or electrons are emitted as the shell vacancies are corrected, the x rays having a wide range of energies which are higher for the higher atomic number materials. The portion of photons interacting by the photoelectric process increases with increasing atomic number and decreasing energy.

The Compton effect involves interactions of photons incident on orbital electrons. The photon gives up part of its energy to the electron causing it to recoil and the balance of its energy goes into a scattered photon. From conservation of energy and momentum, the angular relationships of the recoil electron, scattered photon and incident photon can be determined. The original photon energy determines the distribution of these angles; at lower energies, the angles at which electrons are scattered is greater.

Pair production occurs by the interaction of a photon with the electric field surrounding a charged particle. The original photon disappears with the formation of an electron-positron pair. The photon energy must exceed 1.02 MeV and it is divided equally between the electron and the positron. The portion of incident photons which interact with a nucleus by pair production increases with increasing atomic number.

As a beam of photons traverses matter, the scattering and absorption of photons by all processes results in attenuation of the beam exponentially. This is expressed by the equation:

$$I = I_0 e^{-\mu x} \tag{1}$$

where $I$ is exposure rate at a depth $x$

$I_0$ is exposure rate at zero depth

$\mu$ is the attenuation coefficient

The value of the attenuation coefficient depends on the photon energy and the absorbing material. This coefficient may be expressed in terms of thickness, as a linear attenuation coefficient ($cm^{-1}$), or as a mass attenuation coefficient ($cm^2/g$) obtained by dividing the linear coefficient by the density $\rho$ of the absorbing material. Table 29-1 presents some of these values. More extensive tables are available in reference (2). In matter being traversed by a beam of electromagnetic radiation, there is actually a higher intensity of photons at any point, particularly at great depth,

TABLE 29-1
Mass Attenuation Coefficients

| Photon energy | Mass attenuation coefficient in $cm^2/g$ for — | | | | |
|---|---|---|---|---|---|
| MeV | Aluminum | Iron | Lead | Water | Concrete |
| 0.01 | 26.3 | 173 | 133 | 5.18 | 26.9 |
| 0.02 | 3.41 | 25.5 | 85.7 | 0.775 | 3.59 |
| 0.05 | 0.369 | 1.94 | 7.81 | 0.227 | 0.392 |
| 0.1 | 0.171 | 0.370 | 5.40 | 0.171 | 0.179 |
| 0.5 | 0.0844 | 0.0840 | 0.161 | 0.0968 | 0.087 |
| 1.0 | 0.0613 | 0.0599 | 0.0708 | 0.0707 | 0.0637 |
| 5.0 | 0.0284 | 0.0314 | 0.0424 | 0.0303 | 0.0290 |
| 10.0 | 0.0231 | 0.0298 | 0.0484 | 0.0222 | 0.0231 |

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TABLE 29-2
Dose Buildup Factor (B) for a Point Isotropic Source

| Material | MeV | $\mu x$* | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | 1 | 2 | 4 | 7 | 10 | 15 | 20 |
| Water | 0.255 | 3.09 | 7.14 | 23.0 | 72.9 | 166 | 456 | 982 |
| | 0.5 | 2.52 | 5.14 | 14.3 | 38.8 | 77.6 | 178 | 334 |
| | 1.0 | 2.13 | 3.71 | 7.68 | 16.2 | 27.1 | 50.4 | 82.2 |
| | 2.0 | 1.83 | 2.77 | 4.88 | 8.46 | 12.4 | 19.5 | 27.7 |
| | 3.0 | 1.69 | 2.42 | 3.91 | 6.23 | 8.63 | 12.8 | 17.0 |
| | 4.0 | 1.58 | 2.17 | 3.34 | 5.13 | 6.94 | 9.97 | 12.9 |
| | 6.0 | 1.46 | 1.91 | 2.76 | 3.99 | 5.18 | 7.09 | 8.85 |
| | 8.0 | 1.38 | 1.74 | 2.40 | 3.34 | 4.25 | 5.66 | 6.95 |
| | 10.0 | 1.33 | 1.63 | 2.19 | 2.97 | 3.72 | 4.90 | 5.98 |
| Aluminum | 0.5 | 2.37 | 4.24 | 9.47 | 21.5 | 38.9 | 80.8 | 141 |
| | 1.0 | 2.02 | 3.31 | 6.57 | 13.1 | 21.2 | 37.9 | 58.5 |
| | 2.0 | 1.75 | 2.61 | 4.62 | 8.05 | 11.9 | 18.7 | 26.3 |
| | 3.0 | 1.64 | 2.32 | 3.78 | 6.14 | 8.65 | 13.0 | 17.7 |
| | 4.0 | 1.53 | 2.08 | 3.22 | 5.01 | 6.88 | 10.1 | 13.4 |
| | 6.0 | 1.42 | 1.85 | 2.70 | 4.06 | 5.49 | 7.97 | 10.4 |
| | 8.0 | 1.34 | 1.68 | 2.37 | 3.45 | 4.58 | 6.56 | 8.52 |
| | 10.0 | 1.28 | 1.55 | 2.12 | 3.01 | 3.96 | 5.63 | 7.32 |
| Iron | 0.5 | 1.98 | 3.09 | 5.98 | 11.7 | 19.2 | 35.4 | 55.6 |
| | 1.0 | 1.87 | 2.89 | 5.39 | 10.2 | 16.2 | 28.3 | 42.7 |
| | 2.0 | 1.76 | 2.43 | 4.13 | 7.25 | 10.9 | 17.6 | 25.1 |
| | 3.0 | 1.55 | 2.15 | 3.51 | 5.85 | 8.51 | 13.5 | 19.1 |
| | 4.0 | 1.45 | 1.94 | 3.03 | 4.91 | 7.11 | 11.2 | 16.0 |
| | 6.0 | 1.34 | 1.72 | 2.58 | 4.14 | 6.02 | 9.89 | 14.7 |
| | 8.0 | 1.27 | 1.56 | 2.23 | 3.49 | 5.07 | 8.50 | 13.0 |
| | 10.0 | 1.20 | 1.42 | 1.95 | 2.99 | 4.35 | 7.54 | 12.4 |
| Lead | 0.5 | 1.24 | 1.42 | 1.69 | 2.00 | 2.27 | 2.65 | (2.73) |
| | 1.0 | 1.37 | 1.69 | 2.26 | 3.02 | 3.74 | 4.81 | 5.86 |
| | 2.0 | 1.39 | 1.76 | 2.51 | 3.66 | 4.84 | 6.87 | 9.00 |
| | 3.0 | 1.34 | 1.68 | 2.43 | 2.75 | 5.30 | 8.44 | 12.3 |
| | 4.0 | 1.27 | 1.56 | 2.25 | 3.61 | 5.44 | 9.80 | 16.3 |
| | 5.1097 | 1.21 | 1.46 | 2.08 | 3.44 | 5.55 | 11.7 | 23.6 |
| | 6.0 | 1.18 | 1.40 | 1.97 | 3.34 | 5.69 | 13.8 | 32.7 |
| | 8.0 | 1.14 | 1.30 | 1.74 | 2.89 | 5.07 | 14.1 | 44.6 |
| | 10.0 | 1.11 | 1.23 | 1.58 | 2.52 | 4.34 | 12.5 | 39.2 |

* $\mu x$ = mass absorption coefficient $(\mu/\rho)$ × shield thickness (cm) × shield density (g/cm²).
NOTE: For concrete use an average of aluminum and iron; e.g., B(con^c) = [B(iron) + B(Al)] ÷ 2.
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than would be predicted solely by attenuation (equation 1) because of the presence of x rays and secondary or scattered photons. This increase in exposure rate, called buildup (B), is not easily calculated. It can be included as a factor B in the attenuation equation $I = BI_0e^{-\mu x}$ and tabulated values of B for one set of conditions is shown in Table 29-2. Additional values appear in reference (2). To assure proper accuracy, it is usually necessary to consider this buildup factor.

Since absorption of energy is the physical quantity used in specifying absorbed dose, a mass energy absorption coefficient similar to the attenuation coefficient is useful.
Table 29-3 presents some of these values. More extensive tables will be found in *Physical Aspects of Irradiation.*[6]

TABLE 29-3
Mass Energy-absorption Coefficients

| Photon energy MeV | Mass energy-absorption coefficient in $cm^2/g$ for | | | |
|---|---|---|---|---|
| | Water | Air | Bone | Muscle |
| 0.01 | 4.89 | 4.66 | 19.0 | 4.96 |
| 0.02 | 0.523 | 0.516 | 2.51 | 0.544 |
| 0.05 | 0.0394 | 0.0384 | 0.158 | 0.0409 |
| 0.1 | 0.0252 | 0.0231 | 0.0368 | 0.0252 |
| 0.5 | 0.0330 | 0.0297 | 0.0316 | 0.0327 |
| 1.0 | 0.0311 | 0.0280 | 0.0297 | 0.0308 |
| 5.0 | 0.0190 | 0.0173 | 0.0186 | 0.0188 |
| 10.0 | 0.0155 | 0.0144 | 0.0159 | 0.0154 |

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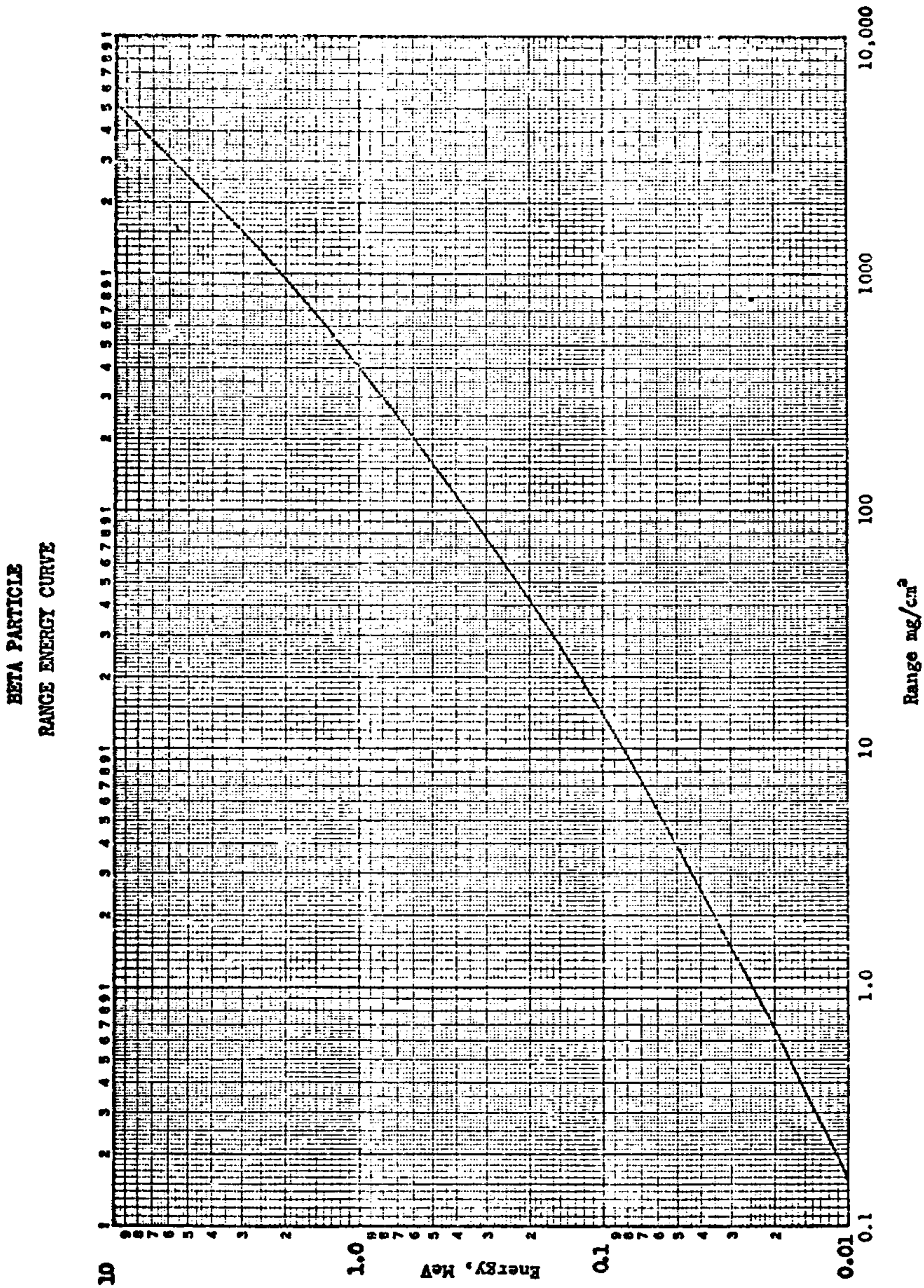
Shielding of radiation sources is commonly provided to reduce the exposure rate in occupied areas. For economy, the shield should be as close as possible to the radiation source. For discrete energies, the attenuation by a shield can be calculated from the attenuation equation, including buildup. In practice, extensive data on attenuation (or transmission), presented in graphic form, is available and used for shielding calculations. This data is presented in a variety of ways and selection of the most useful form will simplify shielding calculations. Because attenuation is exponential, the thickness of a "half-value layer" (HVL) for different shielding materials is a common and convenient form to present such data. A shield thickness of 2 HVL reduces exposure rate by a factor of 4, 3 HVL by a factor of 8, etc. Table 29-4 presents such data for the gamma radiation from several radionuclides, as well as their specific gamma ray constants (exposure rate constants). The latter are useful in determining exposure rate at varying distances, in air, from a point source using the inverse square relationship between exposure rate and distance from the source. Additional values are included in reference (2). Comprehensive shielding data for x rays appear in *Safety Standard for Non-medical X-ray and Sealed Gamma-ray Sources,*[7] and *Medical X-ray and Gamma-ray Protection for Energies up to 10 MeV.*[8]

**Particulate Radiation**

Beta radiation is emitted by a large percentage of the radionuclides, frequently accompanied by x or gamma radiation. Each nuclide, which decays by beta particle emission, emits beta particles having a maximum energy characteristic of that nuclide along with many other particles of lower energy. The average of these energies is much less than the maximum and for different nuclides the ratios of maximum to average span a wide range of values. For different nuclides, the range of maximum energies is from a few keV to slightly over 4 MeV. In contrast to electromagnetic radiation which is attenuated exponentially, beta radiation has a definite range as it traverses matter, the maximum being determined by its energy and the density of the material. If this distance is divided by density, a graph showing range in $mg/cm^2$ versus energy in MeV is applicable to all materials. Values for a given energy from the graph in Figure 29-1, divided by the density of the material being traversed ($mg/cm^3$), gives the thickness of that material (cm) which will completely stop that beta radiation. It should be noted that complete shielding for beta radiation is provided by reasonable thicknesses of commonly available materials. Correspondingly, measuring instruments must be selected which will not significantly impede the beta radiation, this being of particular importance at low energies. As beta radiation traverses matter, the electrons occasionally interact in a manner to produce electromagnetic radiation (bremsstrahlung), the amount of this radiation increasing as the beta energy and atomic number of the absorber increase.

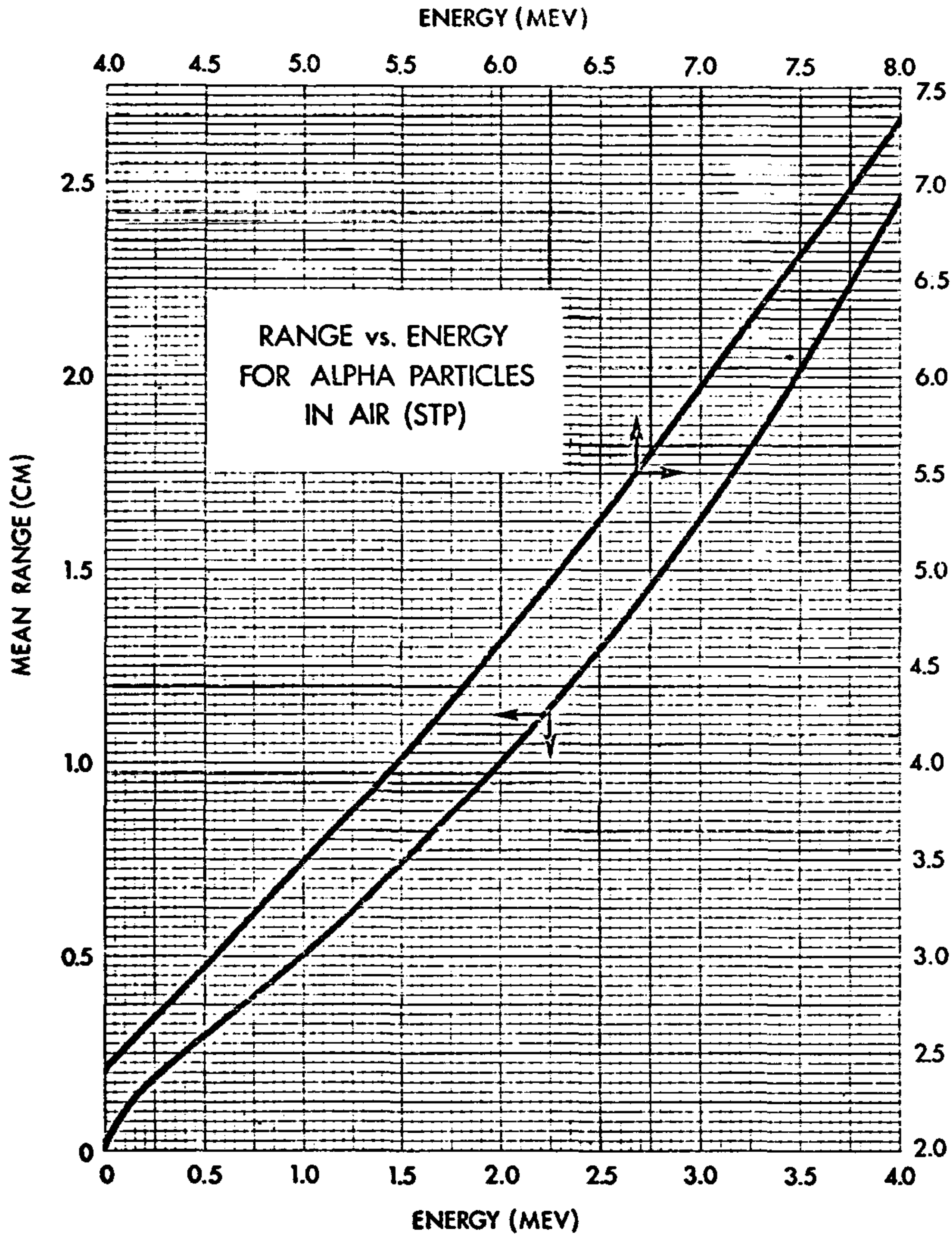
Alpha radiation is emitted primarily by the heavier radionuclides. The alpha particles are emitted at a specific energy characteristic of each nuclide. The energy range of these alpha particles from the different nuclides is predominantly between 4 and 8 MeV. Alpha radiation, like beta, has a definite range in the materials it traverses. The distances traversed, however, are much shorter than for beta. The horny layer of the skin completely stops alpha radiation and air stops it in a few centimeters. Figure 29-2 is a graph showing the range in air for different energy alpha particles.

Although neutrons are not emitted by radionuclides other than by a few that fission spontaneously, there are several types of radioactive neutron sources available and in use. Of course, neutron radiation exists around the core of any nuclear reactor and it will be produced in the event of an accidental nuclear criticality incident. The radioactive neutron sources are sealed sources, normally of relatively small size. Their neutrons are produced by interactions of alpha or gamma radiation with nuclei of appropriate materials (target materials). Alpha emitting radionuclides having a high specific activity are mixed with or alloyed with the target material; and in sources using gamma interactions, the target material usually surrounds the radionuclide. These sources emit neutrons with a maximum energy characteristic of the radionuclide and target material; and many more neutrons at lower energies, with a distinctive energy spectrum. Characteristics of some radioactive neutron sources are shown in Table 29-5. When neutron radiation traverses matter it is attenuated by elastic and inelastic scattering, capture, and induced nuclear reactions. The extent of these processes depends both on the energy (or energy spectrum) of the radiation and the specific nuclides in the matter being traversed. Moderation (slowing down) of the neutrons by elastic collisions progressively changes the energy spectrum. The probability of these interactions taking place is specified in terms of cross-sections,




Radiological Health Handbook. U.S. Dept. of Health, Education and Welfare, Public Health Service, 1970.

**Figure 29-1. Beta Particle Energy Range Curve.**




Radiological Health Handbook. U.S. Dept. of Health, Education and Welfare, Public Health Service, 1970.

**Figure 29-2. Alpha Particle Energy Range Curve.**

with their area expressed in units of barns (1 barn = $10^{-24}cm^2$). These various interactions result in the production of secondary radiations, particularly gamma rays, which must always be considered when neutron radiation is present. The complexities of neutron interactions with matter do not permit adequate treatment of energy absorption and shielding here (see *Protection Against Neutron Radiation).*[9]

**Dosimetry**

Dosimetry involves the evaluation of radiation, often complex as to its nature, energy, direction and quantity, in terms related to its effect on biological systems or other matter. Theoretically, it would seem that measurements could be made to completely describe the radiation field itself at any point of interest including time variations, and from this, the dose or other quantity of interest determined. In practice, however, measurements are made at the place of interest in a manner which relates the measurement directly to the quality of interest — usually absorbed dose or dose equivalent. To the maximum possible extent, there is a summation of the quantities of interest. Furthermore, due to the complexity of any biological response to irradiation by various types and quantities of radiation, as well as their measurement, it is customary to utilize environmental measurements for radiation protection purposes, with precise determinations of dose and dose distribution in biological systems limited to situations of special interest — usually abnormal exposures. Thus, dose-limiting recommendations, although specified in terms of dose equivalent in the body, are commonly evaluated by strictly environmental measurements.

TABLE 29-4
Data for Gamma-Ray Sources

| Radioisotope | Atomic Number | Half Life | Gamma Energy | Half-Value Layer[a] | | | Tenth-Value Layer[a] | | | Specific Gamma-Ray Constant |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | Conc. | Steel | Lead | Conc. | Steel | Lead | |
| | | | MeV | in | in | cm | in | in | cm | R cm$^2$/m Ci-h[b] |
| Cesium-137 | 55 | 27 y | 0.66 | 1.9 | 0.64 | 0.65 | 6.2 | 2.1 | 2.1 | 3.2 |
| Cobalt-60 | 27 | 5.24 y | 1.17, 1.33 | 2.6 | 0.82 | 1.20 | 8.2 | 2.7 | 4.0 | 13.0 |
| Gold-198 | 79 | 2.7 d | 0.41 | 1.6 | — | 0.33 | 5.3 | — | 1.1 | 2.32 |
| Iridium-192 | 77 | 74 d | 0.13 to 1.06 | 1.7 | 0.50 | 0.60 | 5.8 | 1.7 | 2.0 | 5.0[c] |
| Radium-226 | 38 | 1622 y | 0.047 to 2.4 | 2.7 | 0.88 | 1.66 | 9.2 | 2.9 | 5.5 | 8.25[d] |

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[a] Approximate values obtained with large attenuation.

[b] These values assume that gamma absorption in the source is negligible. Value is R/millicurie-hour at 1 cm can be converted to R/Ci-h at 1 meter by multiplying the number in this column by 0.10.

[c] This value is uncertain.

[d] This value assumes that the source is sealed within a 0.5 mm thick platinum capsule, with units of R/mgh at 1 cm.

TABLE 29-5
Data for Neutron Sources

| Source | Half-life | Max. neutron energy MeV | Avg. neutron energy MeV | Yield n/sec. x $10^{-6}$/ curie |
|---|---|---|---|---|
| $210_{Po-Be}$ | 138.4 d | 10.8 | 4.3 | 2.5 |
| Ra DEF-Be | 19.4 y | 10.8 | 4.5 | 2.5 |
| $226_{Ra-Be}$ | 1622 y | 13.2 | 3.6 | 15 |
| $239_{Pu-Be}$ | 24,400 y | 10.6 | 4.5 | 2.0 |

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There are a variety of detectors, with associated readout devices which are used for radiation monitoring or measurement. The most important are: Geiger-Müller (GM) tubes, ionization chambers, proportional counters, luminescent detectors, scintillation detectors, photographic emulsions, chemical reaction detectors, induced radiation detectors and fissionable materials. None are universally applicable and selection of the most appropriate detector or detectors for each radiation measurement (or type of measurement) becomes a matter of great importance. These detectors, with associated readout equipment, are used to perform two distinctly separate functions — to measure the radiation in the environment (monitoring or surveying); and to determine the activity or kind of radionuclide, or both, in solids or fluids, commonly samples from a larger quantity of material (analysis).

G-M tubes detect ionizing events which take place within their sensitive volume, each event causing an output voltage pulse. These pulses may be counted or summed in various ways, usually to give a count rate (counts per minute, cpm). Two types of G-M survey meters are in common use. One uses a cylindrical tube encased in a protective metal shield with an opening on one side over which various absorbers can be placed. The other has the opening at one end of the cylinder where the tube has a very thin "window." The output (cpm) is not proportional to exposure or absorbed dose rate for different types and energy of the radiation. Although scales on survey meters are frequently marked "R per hour," these values are only true for the calibrating radiation. Significant errors can occur from use where the radiation is different than the calibrating radiation. When G-M tubes are used for analysis, proper calibration is likewise essential.

Ionization (ion) chambers are commonly used to measure dose or dose rate (or exposure or exposure rate) from beta, gamma and x radiation. Ions formed by the radiation passing through a selected gas in a chamber are measured either by applying voltage continuously with measurement of the extremely low current flow or by using the chamber as a condenser which is first charged, then exposed to the radiation and the amount of discharge determined. The chamber walls, internal components and gas filling are usually either air equivalent or tissue equivalent.

Proportional counters usually consist of a gas filled cylinder (chamber) containing a central wire to which a potential is applied. The potential is selected so that the output voltage signals are proportional to the energy released by the radiation causing the ionization events in the chamber. This permits selective measurement of different radiations. These counters are commonly used to measure alpha or neutron radiation. The gas in the chamber may be either static or flowing. They may be used as survey meters or for analysis, including spectrometric analysis (measurement of radiation intensity as a function of energy). The alpha survey meters have a very thin "window" to minimize absorption of the radiation.

Luminescent detectors are solids in which energy changes produced by radiation are stored so that subsequent processing will cause them to emit a quantity of light proportional to the energy change. Commonly used materials are metaphosphate glass and calcium or lithium fluoride. The glass is processed by irradiation with ultraviolet and the fluorides by heating. The latter, called thermoluminescent dosimeters (TLD), are finding many uses because of good sensitivity with small pieces. They are used for personnel monitoring, including neutron exposure evaluation.

Scintillation detectors use the phenomenon of light production due to interaction of radiation with crystals or other phosphors (solid, liquid or gas). Light pulses from the scintillator are measured with a photomultiplier tube and its associated electronic equipment. Since the light output and in turn the electrical signal is proportional to the radiation energy absorbed in the scintillator, these devices find a wide variety of uses. They are used as survey meters and for analysis, including spectrometric analysis.

Radiation produces a latent image in photographic emulsions, resulting in darkening of the film when developed by usual techniques. Two general types of film are used, one in which the radiation (beta, gamma, x ray) produces a general blackening, and the other in which small tracks are produced by charged particles, usually protons from fast or thermal neutron interactions in the film. The blackening due to gamma and x rays is not proportional to air or tissue absorbed dose at different energies, and various absorbers are placed adjacent to the film to minimize this aberration. Blackening due to beta radiation varies a small amount with energy. A major use of film for dosimetry has been in personnel monitoring badges. Using both shielded and unshielded sections permits measurement of beta as well as gamma and x rays.

Chemical reaction detectors are systems in which radiation produces a chemical change in a material in such a manner that a chemical analysis or indicator will measure the amount of change. An example is a system using a chlorinated hydrocarbon, such as chloroform, with water and a dye indicator to measure the acid formed due to irradiation. These detectors are not used extensively because of their low sensitivity.

Induced radiation detectors are materials in which the radiation interacts to form radionuclides whose radiation can be measured. They are particularly useful for detecting or measuring neutron radiation. A typical example is the use of indium foil for detection of neutron radiation exposures. Proper selection of foil materials permits evaluation of a neutron energy spectrum. Fissionable materials also are useful for neutron radiation detection and measurement.

## CATEGORIES OF RADIATION EXPOSURE

### Natural Radiation

Individuals continually receive a dose from natural radiation that comes both from sources external to the body and from naturally occurring radionuclides deposited within the body. The external sources are primarily cosmic radiation and gamma radiation from materials naturally present in the ground and in building materials. From foods, drinking water and in the air, several radionuclides are deposited in the body including uranium and its decay products, thorium and its decay products, radiopotassium and radiocarbon. Natural radiation in the United States results in an estimated average annual dose equivalent to individuals of about 125 mrem (100 mrem external and 25 mrem internal). It is unlikely to be less than 100 mrem for any individual and unlikely to be more than 400 mrem for any significant number of people.[10]

### Environmental Radiation

In addition to natural radiation, environmental radiation from man-made sources adds a small increment of dose to the population generally. This dose comes from a wide variety and type of sources including: fallout from nuclear weapons testing; effluents from nuclear and other facilities processing or using radionuclides; luminous dial clocks or watches and signs; and electronic devices, such as television sets, using high voltages. The average annual dose equivalent to the population from these sources is estimated to be only a few percent of natural radiation, probably about five or six mrem per person per year.

**Medical Irradiation**

The planned exposure of patients to radiation is a category which involves a large percentage of the general population. Occupational exposures received incidentally by physicians and supporting staff are not considered part of this exposure category. Diagnostic and therapeutic procedures involve external irradiation with beta, gamma or x radiation, internal irradiation from ingested or injected radionuclides, and irradiation from implanted sealed sources. Doses to individuals vary over an extremely wide range but usually involve only partial body irradiation. Average annual dose equivalent to the population members from these sources has been estimated to be between 50 and 70 mrem per year.[10] Ordinarily, medical and occupational exposures are considered separately. With the exception of a high dose due to an occupational accident, necessary medical exposures are not restricted because of occupational exposures.

**Occupational Irradiation**

Occupational radiation exposures arise from practically every type of radiation and radiation source. The major groups of occupationally exposed personnel are medical or para-medical workers and workers in the expanding nuclear energy programs. However, there are many exposures to radiation or radioactive materials throughout industry, in underground mining, and in many types of research. On the basis of occupational radiation exposure records of the U. S. Atomic Energy Commission and its contractors for 1967, the average annual occupational exposure is estimated at about 500 mrem per person to 100,000 adults (95% of them received less than 1 rem each).[10] Assuming a similar dose to other workers, the estimated average annual dose equivalent to the population members is a fraction of a millirem per year.

## BIOLOGICAL ASPECTS OF IRRADIATION

**Somatic and Genetic Effects**

Irradiation of humans produces two types of effect — somatic and genetic. The somatic effect is the effect on tissues, organs or whole body. Independent of any somatic effect, irradiation of the gonads may cause genetic effects since mutations, which are caused by heritable changes in the germ plasm, may occur. Of course, only the irradiation prior to conception can have this influence.

Somatic effects vary over a wide range — from rapid death due to short term whole body exposures of 10,000 Roentgens or greater to slight reddening of the skin due to minimal exposure.

Effects, including those of particular concern —neoplasms, cataracts and life shortening—may also be delayed for long periods. Within the body, cells react with varying degrees of sensitivity. Tissues also respond differently, depending on dose equivalent rate. Dose fractionation has an ameliorating effect and there is repair of tissues and organs when time permits and the change is not irreversible. Partial body irradiation has much less effect than whole body irradiation. Age is a significant factor; for a given dose many effects are less as age increases. For equal absorbed doses, different types (and energies) of radiation do not produce the same degree of response.

In the study of biological effects, the variation due to different kinds of radiation is referred to as "relative biological effectiveness" (RBE) — the ratio of absorbed doses that produce equal effect, with cobalt-60 gamma rays or 200-250 kV x rays used as the reference. This ratio is reflected indirectly in the quality factor used in radiation protection practice. Those somatic effects (e.g., neoplasms) that are delayed for long periods of time may occur only in a small fraction of the exposed individuals — the *probability* of the effect occurring increasing with increased dose equivalent.

Genetic effects are of general concern because radiation-induced mutations are added to the "load" of defective genes present in the population. Because of the presence of defective genes in all members of the population, it is not possible to identify an abnormality in an offspring with possible mutations caused by irradiation of the parent. Thus, genetic effects relate to population groups, not individuals. Because of this, the radiation exposure to the entire population group is the matter of primary concern, and the genetically significant dose (GSD) has been established as a measure of this population exposure. Furthermore only gonadal exposures during the reproductive period of a lifetime have an influence. Thus, the age at which radiation exposures occur, as well as the dose equivalent, is of prime concern in relation to genetic effects.

**Acute and Chronic Exposures**

Practically all occupational irradiation involves chronic exposures, i.e., small weekly doses (e.g., <100 mrem) occurring over many months and years. Occasionally, due to an accident, an acute exposure may occur, i.e., a high dose (e.g., >25 rem) in a period of a day or less. Somatic response to acute exposure is different from and greater than that for an equal chronic exposure. In a lifetime of occupational exposure without any observable effect, an individual's total dose can be large enough so that an equal dose given in a few hours would be seriously disabling or fatal. Effects of acute exposures may be early, delayed or secondary, and late. Early effects as a result of an acute whole body exposure are shown in Table 29-6. There is less effect for partial body exposures. Delayed effects may occur some time after the early effects have been ameliorated, the extent depending on the dose. In addition to possible loss of hair, one such effect of general concern (often misunderstood) is sterility. Permanent sterility occurs only with absorbed doses to the

TABLE 29-6
Representative Dose-effect Relationships in Man for Whole Body Irradiation

| Nature of Effect | Representative absorbed dose of whole body x or gamma radiation (rads) |
|---|---|
| Minimal dose detectable by chromosome analysis or other specialized analyses, but not by hemogram | 5-25 |
| Minimal acute dose readily detectable in a specific individual (e.g., one who presents himself as a possible exposure case) | 50-75 |
| Minimal acute dose likely to produce vomiting in about 10% of people so exposed | 75-125 |
| Acute dose likely to produce transient disability and clear hematological changes in a majority of people so exposed. | 150-200 |
| Median lethal dose for single short exposure | 300 |

The dose entries in this table should be taken as representative compromises only of a surprisingly variable range of values that would be offered by well-qualified observers asked to complete the right hand column. This comes about in part because whole body irradiation is not a uniquely definable entity. Mid-line absorbed doses are used. The data are a mixed derivative of experience from radiation therapy (often associated with "free-air" exposure dosimetry), and a few nuclear industry accident cases (often with more up to date dosimetry). Also, the interpretation of such qualitative terms as "readily detectable" is a function of the conservatism of the reporter.

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gonads of 500-600 rads of x or gamma radiation; and a single dose of 50 rads may induce brief temporary sterility in many men and some women.[10] Late effects as the result of acute exposure, such as leukemia, may occur many years after exposure, their *probability* increasing as dose increases. From chronic exposures, there are no secondary or delayed effects and the possibility of late effects is minimal. If chronic occupational doses are within the NCRP dose limiting recommendations, the probability of any late effect is so small that it has not been possible to establish clearly whether any such somatic effect exists.

**Internal and External Radiation Sources**

External radiation sources, i.e., those sources which are located external to the body, present an entirely different set of conditions than radionuclides which have gained entrance to the body with their attendant continuous irradiation of the cells and tissues in which they exist. Such radionuclides are called internal radiation sources — sometimes internal emitters.

Entry of internal radiation sources into the body during occupational exposures is principally from breathing air containing particulate or gaseous radionuclides, although ingestion may be a significant mode. Absorption through the skin is significant for some compounds of a few radionuclides, particularly tritium; and implantation under the skin may occur as the result of accidental skin puncture or laceration. Once inside the body, radionuclides are absorbed, metabolized and distributed throughout the tissues and organs according to the chemical properties of the elements and compounds in which they exist. Their effects on organs or tissues depends on the type and energy of the radiation and residence time. Both radioactive decay and biological elimination remove radionuclides from the body and its organs, these removal rates frequently being expressed as half-lives. The net rate is designated as the "effective half-life." While metabolically similar, the degree of effect from different radioactive isotopes of the same element will vary according to the type and energy of the radiation they emit and their radioactive half-life. Acute or early effects do not occur from internal radiation sources, with the possible exception of a very large intake of certain radionuclides.

While radiation measurements can be made in the environment of workers which characterize their dose equivalent from external radiation sources, no comparable environmental radiation measurement will reveal dose equivalent from exposures to internal radiation sources. Instead, evaluation (and control) is based on activity concentrations in air or water, a specific relation between these concentrations and the resulting dose equivalents for each radionuclide having been determined from human experience when available, or from calculations. Thus, practical dose-limiting recommendations are expressed in terms of maximum permissible concentrations (MPC) for inhaled or ingested radionuclides. A similar concept is expressed as a Radioactivity Concentration Guide (RCG) by the Federal Radiation Council.[4] For essentially insoluble gases producing beta or gamma radiation, such as the inert gases argon and krypton, the amount of the radionuclide that becomes an internal radiation source is so small that the external irradiation from an infinite cloud surrounding the individual will produce the greater dose equivalent.

The effect from external radiation sources depends on the penetrating ability of the particular radiation. Thus, alpha radiation is of no concern externally, and beta is stopped in the outer tissues, the depth depending on energy. Very low energy x or gamma radiation is attenuated quite rapidly.

The effect of radiation on any organ or tissue is dependent on the total dose equivalent from both internal and external radiation sources. Thus, the total dose equivalent must be considered when comparisons with the MPD are made. Theoretically, it should be possible to sum these separate dose equivalents but in practice such quantitation is difficult, if not impossible. Therefore, it is customary to use the two different dose-limiting recommendations conservatively.

**Critical Organs and Tissues**

The various tissues and organs of the body are not affected equally by equal irradiation. Their responses vary considerably and for radiation protection purposes it is essential that dose equivalent to the most sensitive organs essential to well being be given primary consideration. For uniform whole body irradiation, the blood forming organs (red bone marrow), the lens of the eye, and the gonads are more susceptible to significant effects and these are designated as "critical organs." Of course, for individuals past reproductive age the gonads are not a critical organ. For those internal radiation sources that do not irradiate the body uniformly the distribution and metabolic pattern for each radionuclide will determine which organs and tissues receive the larger dose. Again, for radiation protection purposes, any essential organ or tissue which is likely to be affected the most by the radiation from internal radiation sources is of primary concern and these are also designated as critical organs. These critical organs (and tissues), sometimes designated as limiting organs, are: lung, GI tract, bone, muscle, fatty tissue, thyroid, kidney, spleen, pancreas and prostate.

The total activity (curies) of a radionuclide in the body is designated as the "body burden." Distribution may be inhomogeneous, with a large fraction in one or more organs or tissues. While the activity in the crititcal organ is the limiting factor, the body burden corresponding to the MPD for the critical organ indicates the total activity that should be present in the entire body. It is designated as the maximum permissible body burden.

## RADIATION PROTECTION CONSIDERATIONS

**Occupational and Public Exposures**

Occupational radiation exposures involve a select age group of healthy individuals. Their exposures occur for periods not exceeding approximately eight hours per day and 250 days per year. This group is a very small portion of the general population and they are trained in radiation protection practices. In contrast, the general population necessarily includes the unborn, the very young, the sick or disabled; and their exposures can be continuous — 24 hours per day, 365 days per year. For these, and other reasons, dose-limiting recommendations for the general population are set at lower limits than for occupational exposure, commonly by a factor of 10 or greater. Dose-limiting recommendations applicable to the public are designated frequently as "dose limits," those for occupational exposure as "maximum permissible dose equivalent (MPD)." The Federal Radiation Council designates both as RPG's. Only those individuals whose duties involve exposure to radiation should be classed as "occupationally exposed" and their training in radiation protection should be assured.

**Dose Assessment**

To accurately determine the true dose equivalent to the critical organs of all occupationally exposed individuals is a desirable objective which in practice becomes impractical, if not impossible. Activities in the workplace are varied in space and time, the energy and frequently the type of radiation varies, parts of the body being irradiated change with time, irradiation may occur from both internal and external radiation sources, and measurement devices have varying degrees of accuracy. Environmental measurements, however, can be made in a manner such that they provide a conservative evaluation of the dose equivalent to the critical organs and in turn assure that dose-limiting recommendations are not exceeded. This is accomplished by a combination of radiation surveys, area monitoring and personnel monitoring. If radiation surveys or other adequate data indicate that external irradiation will be less than one fourth of the applicable dose-limiting recommendation, personnel monitoring devices are not recommended. Above this, suitably selected personnel monitoring devices are required for evaluation of the radiation environment in which the individual works. The dose equivalents indicated by these are normally conservative with respect to any critical organ dose equivalent, and for general control purposes their readings can be compared to the applicable dose-limiting recommendation. Personnel monitoring devices are normally worn on the trunk of the body, but for some types of work they are required on extremities, particularly hands and forearms, as well. The dose-limiting recommendations permit higher doses here. Possible doses from small beams not intercepted by personnel monitoring devices must be evaluated by other means.

For exposures to airborne radioactive materials, the activity concentration in the breathing zone of the worker, averaged over a 40-hour weekly period, is compared with the tabulated values of maximum permissible concentrations (MPC) for the radionuclides of concern (see page 390). These concentrations, if breathed 40 hours per week indefinitely, will produce a dose equivalent in the critical organ equal to the dose-limiting recommendation.

If a valid determination of total dose equivalent to the whole body, the parts of the body, or the critical organ is required, such as after an abnormal exposure or to establish a monitoring procedure, a detailed evaluation based on all pertinent data should be made.

To convert absorbed dose to dose equivalent, the rounded practical values of the quality factor in Table 29-7 may be used. Methods of calculating a quality factor are described in reference (10). If neutron flux density and energy are measured or known, the dose equivalents may be found in Table 29-8.

TABLE 29-7
Practical Quality Factors
Radiation Type

| Radiation Type | Rounded QF |
|---|---|
| X rays, gamma rays, electrons or positrons, Energy >0.03 MeV | 1 |
| Electrons or positrons, Energy <0.03 MeV | 1 |
| Neutrons, Energy <10 keV | 3 |
| Neutrons, Energy >10 keV | 10 |
| Protons | 10 |
| Alpha particles | 20 |
| Fission fragments, recoil nuclei | 20 |

TABLE 29-8
Mean quality factors, QF[a], and values of neutron flux density which in a period of 40 hours results in a maximum dose equivalent of 100 mrem.

| Neutron Energy | $\overline{QF}$ | Neutron Flux Density |
|---|---|---|
| MeV | | $cm^{-2}\ s^{-1}$ |
| $2.5\times10^{-8}$ (thermal) | 2 | 680 |
| $1\times10^{-7}$ | 2 | 680 |
| $1\times10^{-6}$ | 2 | 560 |
| $1\times10^{-5}$ | 2 | 560 |
| $1\times10^{-4}$ | 2 | 580 |
| $1\times10^{-3}$ | 2 | 680 |
| $1\times10^{-2}$ | 2.5 | 700 |
| $1\times10^{-1}$ | 7.5 | 115 |
| $5\times10^{-1}$ | 11 | 27 |
| 1 | 11 | 19 |
| 2.5 | 9 | 20 |
| 5 | 8 | 16 |
| 7 | 7 | 17 |
| 10 | 6.5 | 17 |
| 14 | 7.5 | 12 |
| 20 | 8 | 11 |
| 40 | 7 | 10 |
| 60 | 5.5 | 11 |
| $1\times10^{2}$ | 4 | 14 |
| $2\times10^{2}$ | 3.5 | 13 |
| $3\times10^{2}$ | 3.5 | 11 |
| $4\times10^{2}$ | 3.5 | 10 |

[a]Maximum value of $\overline{QF}$ in a 30-cm phantom.

Tables 29-7 and 29-8 reprinted with permission of National Council on Radiation Protection and Measurements, from "NCRP Report No. 39" — (1971) Washington, D.C.

TABLE 29-9
NCRP Dose-limiting Recommendations

| | |
|---|---|
| **Maximum Permissible Dose Equivalent for Occupational Exposure** | |
| Combined whole body occupational exposure | |
| Prospective annual limit | 5 rems in any one year |
| Retrospective annual limit | 10-15 rems in any one year |
| Long term accumulation to age N years | $(N-18)\times5$ rems |
| Skin | 15 rems in any one year |
| Hands | 75 rems in any one year (25/qtr) |
| Forearms | 30 rems in any one year (10/qtr) |
| Other organs, tissues and organ systems | 15 rems in any one year (5/qtr) |
| Fertile women (with respect to fetus) | 0.5 rem in gestation period |
| **Dose Limits for the Public, or Occasionally Exposed Individuals** | |
| Individual or occasional | 0.5 rem in any one year |
| Students | 0.1 rem in any one year |
| **Population Dose Limits** | |
| Genetic | 0.17 rem average per year |
| Somatic | 0.17 rem average per year |
| **Emergency Dose Limits—Life Saving** | |
| Individual (older than 45 years if possible) | 100 rems |
| Hands and forearms | 200 rems, additional (300 rems total) |
| **Emergency Dose Limits—Less Urgent** | |
| Individual | 25 rems |
| Hands and forearms | 100 rems, total |
| **Family of Radioactive Patients** | |
| Individual (under age 45) | 0.5 rem in any one year |
| Individual (over age 45) | 5 rems in any one year |

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The dose to the whole body or to the critical organs from internal radiation sources continues as long as the radionuclide is present. When intake is stopped, the dose decreases with time — frequently exponentially. Where the effective half-life is long, the total dose equivalent is rather large in comparison to that produced during and shortly after the time of exposure. This total dose equivalent, integrated over a lifetime, is designated as the "dose commitment." It is useful in a number of different types of evaluations.

**Dose-Limiting Recommendations**

The National Council on Radiation Protection and Measurements (NCRP) is generally recognized as an authoritative source of radiation protection information, data and criteria in the United States. NCRP Report No. 39[10] discusses radiation protection criteria in detail and presents their dose-limiting recommendations, which are shown in Table 29-9. The NCRP comment on the occupational limits is: "There will be occasions when the measured or estimated actual dose equivalent exceeds the prospective limit of 5 rems in a year. No deviation from sound protection is implied if the retrospective dose equivalent does not exceed 10 to 12 rems for dose increments well distributed over time or even 15 rems for exceptionally well-distributed increments. Repetition of retrospective dose equivalents in excess of planned limits is controlled by the long-term occupational accumulated dose equivalent." The NCRP recommendations serves as the basis for various regulations and standards in which interpretations are made according to specific needs. Regulations of states, U.S. Atomic Energy Commission and other governmental agencies may not be the same as NCRP recommendations and must be consulted and used as applicable (see Chapter 9 and page 392).

Similarly, NCRP has provided tabulated values of maximum permissible body burdens and maximum permissible concentrations of radionuclides in air and water for occupational exposures. Some of these values are shown in *Radiological Health Handbook*.[2] The complete tabulation is in NCRP Report No. 22[11] and a similar tabulation, with the derivation data and methods, is in a report of the International Commission on Radiation Protection.[12] The various regulations also contain such tabulations, usually including values applicable to the general public.

## IRRADIATION BY EXTERNAL RADIATION SOURCES

**Exposure Control**

A basic concept in radiation protection practice is the establishment of a "controlled area." Access to these areas must be controlled and within them supervision and control of occupational exposures is provided. Emergence of beams and escape of radioactive materials from these areas are also controlled. These areas are identified by use of the standard radiation symbol[13] with associated warning notices. This symbol, a purple trefoil on a yellow background, also identifies any radiation source.

Since the useful beam of x-ray equipment may inflict a year's MPD in minutes or less, the design of industrial x-ray facilities must of necessity give proper consideration to the establishment of a suitably controlled area which will assure proper radiation protection for two groups of individuals — those who operate the equipment (occupationally exposed) and those in the environs, either normally or casually (not occupationally exposed). Where possible, the x-ray equipment should be within a room or other enclosure arranged with controls outside and having interlocks to prevent entry when equipment is energized. Shielding can then be provided so that the exposure rate outside the enclosure will be low enough to insure that the applicable MPD or dose limit will not be exceeded. Small devices or instruments using x rays, such as laboratory equipment, usually can be totally enclosed with adequate shielding, but accessibility to the inside of the shield requires special consideration (interlocks, etc.). Where work requires truly mobile or portable equipment, exposure time and distance from the equipment become the basic method for controlling exposure rates to values which will insure that no individual exceeds the applicable MPD or dose limit. Portable shielding can be an aid. *American National Standard Z54.1*-1963[7] classifies x-ray and sealed gamma-ray source installations into three types: exempt, enclosed and open. Shielding design and operational requirements are given. Although intended for medical installations, *Medical X-ray and Gamma-ray Protection for Energies up to 10 MeV*[8] may provide useful data; shielding data is also presented in reference (2). For accelerators, *American National Standard Radiological Safety in the Design and Operation of Particle Accelerators*[14] establishes safety requirements.

In addition to x-ray equipment, there may be other sources of x rays in industry, such as high voltage (>10kV) electron tubes, which may require shielding or other means of control to assure adequate radiation protection for workers (or the public).

Gamma radiation, usually from a sealed source, is used for a variety of purposes in industry. The larger sources produce beams comparable in exposure rate to x-ray equipment. Detailed descriptions cannot be given here, but rather a few general considerations. Gamma radiation cannot be turned off like x rays. This imposes a severe requirement on retention of the sealed source at a predetermined specific location where exposure control is assured or within appropriate shielding at all times. Procedures and surveys must guarantee this control. The integrity of the encapsulation or bonded cover of the sealed source must be assured at all times to prevent release of the radioactive material into the environment where it could be dispersed and inhaled or ingested. Periodic tests, such as smears of the sealed source or its container, should be made. Appropriate testing of radium sources is particularly important because any failure will release radon gas which, with its daughters, can contaminate the surrounding area. Exposure rates from sealed sources, in air, can be calculated from the specific gamma-ray constant (see page 382). As an approximation, the gamma exposure rate (R/hr) at 1 foot is 6CE, where C is the number of curies and E is the total energy per disintegration in MeV. As for x-ray installations, references (7) and (8) provide useful shielding information.

Beta radiation sources, which are frequently built into some piece of equipment such as a thickness gauge, must be shielded and arranged so that access to the beta radiation is prevented. Of particular concern is control of exposures during any maintenance procedures. Consideration must be given to any associated gamma radiation; and to the bremsstrahlung exposure rate, particularly for sources of high activity and energy. To permit escape of the beta radiation, the encapsulating material must be relatively thin, at least over the useful area of the source. Damage to this encapsulation will permit release of the radionuclide to the environment where it can be dispersed and inhaled or ingested.

Radioactive neutron sources are commonly small sealed sources of relatively substantial construction. Yield of neutrons is proportional to the activity in the source. See page 385 for neutron source data. Consideration must be given to gamma radiation as well as neutron radiation from them. If the radionuclide in them is radium, the gamma dose equivalent rate is higher than that from neutrons and the possibility of radon leakage must be recognized. Leakage of any of the radionuclides used in these sources presents a hazard of considerable magnitude which necessitates care in use and periodic testing. Commonly used shielding materials are concrete, polyethylene, boronated polyethylene or boron in other materials such as aluminum. Shielding and other useful data will be found in *Physical Aspects of Irradiation*[8] and *Protection Against Neutron Radiation*[9].

External radiation sources involve a wide variety of equipment which cannot be described here. Descriptions and useful data will be found in *Radiation Hygiene Handbook.*[15]

**Exposure Evaluation**

Applicable regulations (or NCRP recommendations) established the time period during which specific dose equivalents may be given to workers. Currently, most regulations permit a limit of 1.25 rem/quarter indefinitely to the whole body, gonads, bloodforming organs and lens of the eye; or, for individuals whose previous radiation history has been established, a limit of 3 rems/quarter* with an overriding yearly limitation of 5(N-18) rems, N being age in years. Separate quarterly and usually yearly limits, with higher values, are specified for extremities and skin. Exposure evaluations therefore must be related to these time periods, no matter whether measurements are made in rems (or R) per hour, per day, per week or per month. Administratively, daily or weekly limits are frequently used for general control.

The evaluations to establish dose equivalent from external radiation sources are accomplished by conducting radiation surveys and monitoring the environment in which the individuals work. The radiation survey establishes the parameters that must be measured and depicts whether occasional or essentially continuous surveillance with measuring instruments is required. Proper instruments must be selected for the survey so that all possible types and energies of the radiation will be measured with reasonable accuracy. A wide selection of instruments is available commercially[5,16,17] but caution must be exercised to be certain their specified capability will fulfill the required needs. A comprehensive discussion of instrumentation is in *Radiation Protection Instrumentation and Its Application.*[18] Proper calibration for the radiation to be measured is essential. The higher the instantaneous dose rates or potential dose rates in the work area and the greater the complexity of the operations, the greater the need for continuous or frequent surveillance measurements; and correspondingly, closer control over the exposure time of the workers. Where abnormal situations may occur, area monitoring by permanently installed instruments at key locations, with readouts under observation at a central location, will aid in detecting any significant changes of the radiation levels in the general environment. Use of alarms may be indicated in extreme cases.

Except where it can be assured that dose equivalent rates are consistently very low (<25% MPD), personnel monitoring devices must be used to measure the radiation incident on a worker's body (or extremities), integrated over preselected time periods. Film badges, available from commercial services,[16,17] have been used extensively with time periods usually being from a week to a month. Currently, there is increasing use of thermoluminescent dosimeters (TLD) for these and longer periods. Where dose equivalent rates are high and variable, the dose accumulated during minutes or hours of exposure becomes critical and pocket dosimeters (ionization chambers) provide a convenient means of such measurement. There are two types, those that require an instrument for readout and those that can be read directly. The latter are particularly useful when the worker can read them frequently and limit his work period or procedures accordingly. A film badge or TLD is normally used in addition to the pocket dosimeter to provide a back-up for an off-scale reading. There may be discrepancy between the two readings because of different response characteristics. A pocket-size instrument with an alarm sensitive to either dose or dose rate is available and can be used if operating conditions warrant. Indium foils may be added to film badges or other badges worn by workers if accidentally high neutron exposures may occur, the induced activity permitting a rapid qualitative check for a high neutron exposure.

External irradiation of the body tissues or organs may occur from radionuclides deposited on the skin or in the clothing. To detect (or measure) this requires a very careful probing over the entire body with a suitable instrument. A probe on a flexible cord is desirable; or where the contamination is limited to the hands or shoes, a "hand and foot counter" may be used.

Although periodic medical examinations are desirable for many reasons, they cannot be used as a means of exposure evaluation unless dose equivalents are many times the MPD (see Table 29-6). A relatively new cytogenetic technique involving a determination of chromosome irregu-

*This was a former NCRP recommendation.

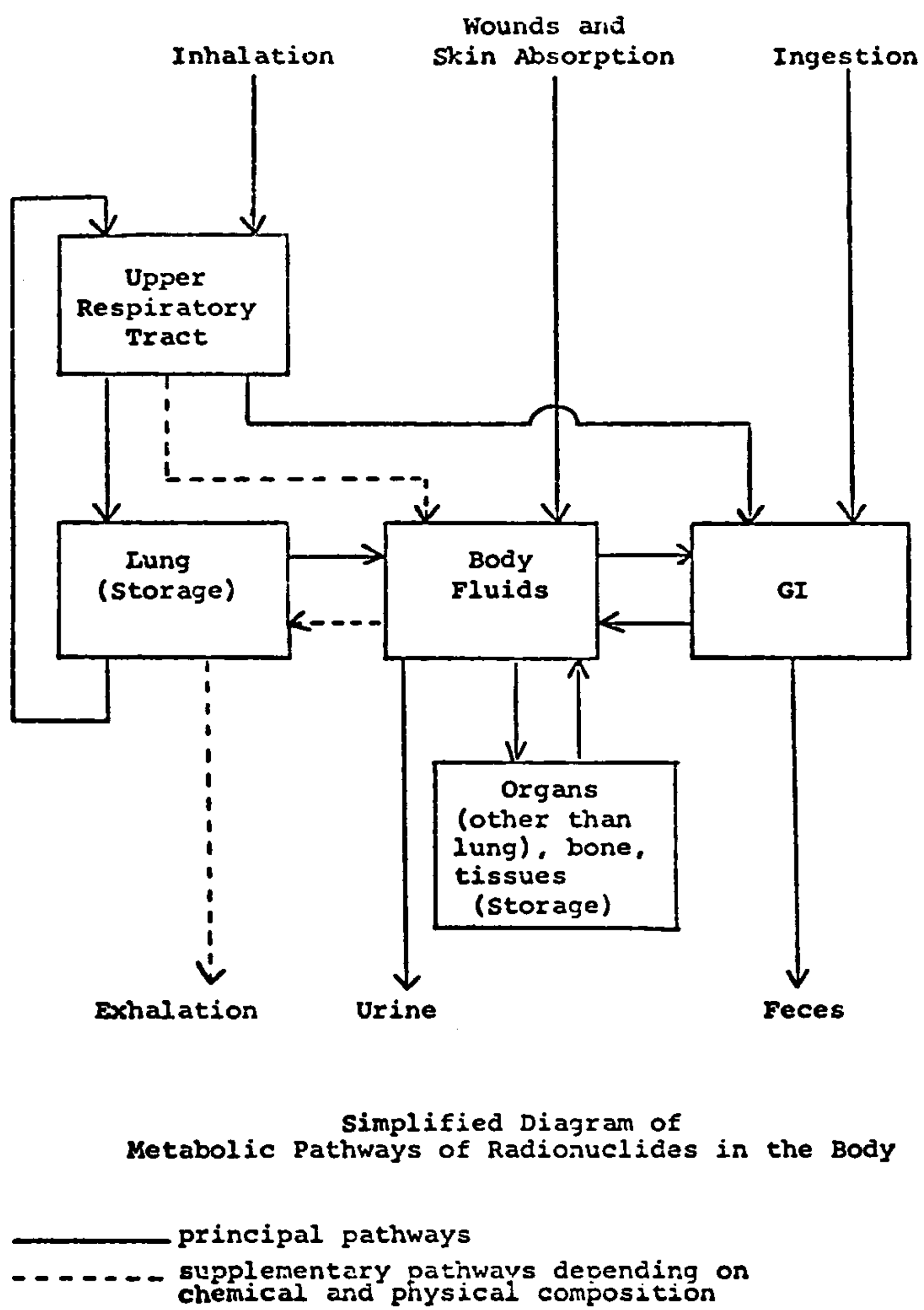


Figure 29-3. Radionuclide Pathways through the Body.

larities found in somatic human blood cells,[19] although nonspecific, can provide a means of measuring dose equivalents slightly above the MPD, but use of this technique is severely limited due to the many man-hours required for each determination.

General administrative practices for radiation monitoring are presented in *American National Standard Guide for Administrative Practices in Radiation Monitoring*.[20]

## IRRADIATION BY INTERNAL RADIATION SOURCES

### Mode of Entry

Internal radiation sources gain entry to the body by breathing gaseous or particulate airborne radioactive materials, by swallowing radioactive materials that have gotten into the mouth from contaminated lips, hands, foods, or liquids, and by absorption through or implantation under the skin. After entry, a rather complex distribution throughout the body may occur as indicated in Figure 29-3. Although there may be irradiation throughout the body, the organs or tissues where the residence time and concentration are greatest receive most of the dose equivalent from alpha and beta radiation, while gamma dose is more distributed.

When inhaled, a fraction of radioactive gases and particulates are retained and absorbed in accordance with chemical and physical properties (not radioactive properties), the balance being exhaled. The retained material is distributed along all respiratory passages — that deposited in the upper passages being subsequently swallowed after clearance by drainage or ciliary action. Retention of particulates in the several sections of the respiratory tract is a function of the particle size distribution, the larger particles ($>10$ $\mu$m dia.) not reaching the lung. Soluble materials, when deposited in the lung, are taken up in the blood stream, their subsequent distribution and excretion being determined by the metabolic pattern for that element. Insoluble materials are retained in the lung with a relatively slow clearance rate (e.g., $\sim$ 120 day half-life). Without specific data, ICRP recommends an assumption that 25% is exhaled, 50% is deposited in the upper respiratory passages and 25% is deposited in the lungs, with all of that in the upper passages and half of that in the lungs being swallowed.

Ingested materials, including those cleared from the respiratory tract, pass through the gastrointestinal tract, with their absorption and excretion being determined by solubility of the particular chemical compound and metabolic pattern of the element.

Embedded materials, unless very soluble, tend to remain in the tissues near the sight of entry with a slow clearance rate from that site. Those few materials which can be absorbed through the skin are promptly distributed throughout the body tissues.

The tabulated values of MPC for air and water take all of these various ramifications into account except for embedded materials, yet for exposure control and exposure evaluation purposes some of the above factors require consideration.

### Exposure Control

Work with radioactive materials that are not effectively contained necessitates the establishment of a well defined controlled area. Preferably it should be a room or other totally enclosed area which will prevent atmospheric dispersion of the radioactive materials to outside areas. Exhaust of air from the room through filters may be required. Movement of individuals or materials through the exit (entrance) should be controlled to prevent inadvertent transfer of radioactive materials outside. Any liquids containing radioactive materials should either be retained for safe disposal at an authorized location or be put into a drain which fulfills pertinent requirements for release of radioactive materials to the environment. Only workers, properly trained, or visitors properly controlled, should be permitted entry. The degree of these controls will vary over a wide range for different kinds of work depending on the activity (Ci) involved, the MPC of the radionuclides in air or liquid, the dispersion characteristics of the materials and the size or complexity of the operations.

Within the controlled area, the workers must be protected against breathing radionuclides in concentrations greater than the MPC averaged over a 40 hour week. Where exposure times are less or greater than 40 hours in a week, the tabulated MPC may be adjusted up or down proportionately. Although the MPC's are set for 40 hours per week exposures indefinitely, most regulations require that each week be considered separately. For work with very small amounts of material, e.g., less than the activity in a few maximum permissible body burdens, rather simple exposure controls are required — perhaps gloves and a lab coat. As the activity and dispersibility increase, protective measures progress through ventilated hoods, specially designed exhaust hoods, total enclosures and glove boxes. Some of these are described in reference (15). Use of personal protective equipment such as coveralls, gloves, head covers and shoe covers may be indicated (see Chapter 36). Where the concentration of airborne radioactive materials cannot be adequately controlled by exhaust and enclosures, respiratory protective equipment approved for radioactive materials is required.[21] Where abnormal concentrations may occur, continuous air sampling devices with an alarm may be needed. Contamination of surfaces with radioactive materials throughout the controlled area may require control, such measurements being made by counting smears taken on filter papers from 100 cm$^2$ areas. Clothing change rooms with shower facilities, located at the exit (entrance) of the controlled area, may be required to prevent spread of radioactive materials from the area and to assure removal of contamination from workers' bodies. Possible remaining contamination is checked with an instrument probe.

For a laboratory, many or all of these factors require consideration. *American National Stand-*

*ard Design Guide for a Radio-Isotope Laboratory (Type B)*[22] is a general guide to these requirements.

**Exposure Evaluation**

Except for an accident, internal radiation sources resulting from occupational exposures usually accumulate gradually in the body of workers. Breathing airborne radioactive materials either intermittently or continuously is a prime source. Thus, a knowledge of the radionuclides and their average activity concentration in the air breathed by workers becomes important in any assessment of dose equivalent to critical organs or the corresponding bodily intake in relation to the MPC. Usually it is sufficient to assure that the weekly average airborne radioactivity concentration in the breathing zone of workers is less than the MPC, although other possible modes of entry should be considered. Hence, where dispersible radioactive materials are used, an air sampling program becomes a necessity. This program may vary from an occasional spot check where concentrations are readily maintained at a small fraction of the MPC, to continuous sampling either at or near the breathing zone of workers where exposure may average near the MPC or equipment failures may produce abnormal conditions.

Air sampling techniques and instruments discussed in Chapters 13, 14 and 15 may be used for radioactive materials provided the sample is suitable for radioactivity analysis. Samplers using filter paper or membranes have found general application because the activity measurement can be made readily. For some situations, samplers which provide a size separation are used. The samplers may be portable instruments with a sampling head that can be positioned or held near the breathing zone of workers, or be permanently mounted equipment. *American National Standard Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities*[23] provides a complete guide to sampling airborne radioactive materials.

Radionuclides are removed from the body by excretion in the urine and feces, the rate and partitioning between the two depending on many factors including mode of entry into the body, elemental composition, solubility and rate of intake. For many radionuclides, sufficient knowledge of excretion patterns and rates are available so that measurements of excretion rates can be quantitatively related to intake rates, or activity in the body, or both. Data obtained from analysis of urine or feces — frequently called bioassays — can serve as an assessment of previous and current intake rates to verify air sampling data. By discontinuing current intake for a period, an estimate of the body burden or critical organ burden can be made from such excretion measurements — usually a series of measurements over a period of a week or longer. When used to assess current intake rates it is common practice to establish, for each radionuclide, an "investigation level" or "check point." These are discussed in *Recommendations of the International Commission on Radiological Protection, Report of Committee IV on Evaluation of Radiation Doses to Body Tissues from Internal Contamination Due to Occupational Exposure*,[24] with very conservative values listed. For excretion rates or activity concentrations in the excreta which are below an appropriate investigation level, it is assumed that exposure controls and evaluations have been adequate; but if above these levels, an examination of the adequacy of current practices is made.

A further means of assessing the current status of a worker with respect to intake and retention of radionuclides is an in vivo determination of body burden or critical organ burden by measuring the gamma radiation being emitted from his body (or critical organ). The simplest of these is measurement of radioiodine in the thyroid by placing an instrument adjacent to the thyroid. For most radionuclides, a "whole body counter" is required. It consists of one or more measuring instruments which scan the whole body, usually in a well shielded enclosure to minimize the effect of background radiation.[25] Such measurements require skilled personnel and very sensitive measuring devices, properly calibrated. Such measurements can be made for radionuclides emitting x rays, such as uranium-235, plutonium-239 and americium-241, as well as for radionuclides emitting higher energy gamma radiation.

As with external irradiation, medical examinations cannot be used to evaluate dose equivalent from internal radiation sources when exposures are at or below MPC, and the cytogenetic technique (page 392) may be useful in a qualitative way at slightly higher exposures.

## PARTICULAR CONDITIONS

**Nuclear Criticality Safety**

A few of the heavier radionuclides which are capable of sustaining a nuclear chain reaction require handling and processing techniques which will avoid the possibility of inadvertently forming a critical mass, with the attendant emission of intense radiation. The mass of these fissile materials at any one location or their geometric arrangement must be controlled with a high degree of confidence. Information about the basic limiting parameters used to assure nuclear criticality safety are in *American National Standard N16.1*[26] Additionally, all fissile materials are used under Atomic Energy Commission regulations and license — in which they are designated as "Special Nuclear Materials." The regulations designate the masses of fissile materials below which nuclear criticality safety controls are not required and licenses specify the limiting conditions of use for greater amounts. A standard symbol is used to identify fissile materials and areas in which they are used.[27] It is similar to the radiation symbol with circular bars around it.

**Nuclear Reactor Industry**

The nuclear reactor industry presents an array of radiation protection problems much too complex to discuss in this Chapter. During mining of uranium ore there are exposures to radon gas and its daughters, as well as to uranium dust;[28] and similar problems occur during the processing to extract the uranium from the ore.[29] The uranium enrichment (in uranium-235) process involves various chemical forms of uranium including uranium hexafluoride, a gas. After enrichment, nuclear criticality safety controls become mandatory for all subsequent handling and processing. Fuel manufacturing[30] involves a chemical conversion process and various treatments of the resulting solids prior to loading into the fuel rods. The fuel rods are sealed and subsequent handling involves no further exposure to airborne radioactive materials. Fuel manufacturing and much of the previous processing involves only relatively minor external irradiation problems. In a nuclear reactor facility, control of exposure to external radiation sources as well as control of fission and corrosion products which may become airborne becomes necessary. Fuel reprocessing plants take spent fuel, which is highly radioactive, and pass it through complex chemical processes to separate the fuel materials (including plutonium) from the fission products, so that both external irradiation and complex airborne radioactive materials, including gases, require control and evaluation. Fuel manufacturing may include fuel elements that contain plutonium — requiring sophisticated controls to prevent release of plutonium. Throughout the industry, control and evaluation of releases of radioactive materials to the environment is essential.

**Transportation**

Radioactive materials are shipped by all normal transportation methods including the U. S. Postal Service, railroads, airplanes, trucks and ships. There are some limitations on the types and quantities that will be accepted by some of these, particularly the Postal Service. Regulations of the U. S. Department of Transportation (DOT) are applicable to all interstate transport except for the Postal Service. There are separate regulations for the Coast Guard and Federal Aviation Agency, although these conform to the DOT Regulations. Most states have regulations applicable to intrastate transport and a few cities have some regulations. Turnpikes, bridges and tunnels operated by authorities may have separate regulations, limitations and requirements. Packaging of fissile materials and large quantities of radioactive materials are subject to Atomic Energy Commission Regulations, 10 CFR 71 (see Chapter 9). International transport is subject to regulations of the International Atomic Energy Agency.

In all cases the shipper is required to provide packaging which fulfills the requirements of the pertinent regulations except for small quantities that are exempted. For detailed packaging and labelling requirements, the regulations applicable to the mode of shipment should be consulted.[31,32]

**Records**

Since some diseases and attendant disability that may be caused by exposure to radiation or radioactive materials can occur after many years of exposure or many years after exposure, retention of suitable records relating to exposures and working conditions for long periods of time is desirable and usually required by regulations. Workmen's Compensation Laws usually permit filing claims for radiation injury many years after exposure occurred, and for these adequate records will be required. A comprehensive presentation of information about records and their retention is in *American National Standard N2.2.*[33]

**Regulations**

The use of radiation and radioactive materials is subject to official regulation by various governmental agencies. Such regulations are discussed in Chapter 9. It is essential that any user of radiation or radioactive materials become intimately familiar with the details of all current rules and regulations applicable to his operations. Licenses, permits or notifications are commonly required. Before preparing a license application the person who will subsequently issue the license should be consulted. Because of the length, detailed provisions and variability of these rules and regulations, only a few general provisions have been mentioned in this chapter.

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