Chapter 7

Radioactive Cloud-dose Calculations

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LIST OF SYMBOLS

Symbols used frequently in Chaps. 7 and 8 are listed here. The equation or section number indicates the first appearance of the symbols. The noted units for each symbol are those that are typically used in the calculations discussed in the text although other units may be used in specific instances.

A	Atomic weight: relative weight of an atom based on a weight of 16 for oxygen, Eq. 7.14
B'	Breathing rate (m^3/sec), Eq. 7.98
B _x	Dose buildup factor (dimensionless), Eq. 7.74
b	A height above the ground ($\text{g}/\text{cm}^2, \text{m}$) for beta- and gamma-ray equations, respectively, Eqs. 7.26 and 7.65a
c	Energy-dependent parameter in equations for estimating beta dose (dimensionless), Eq. 7.25
c _m	Concentration of radioactive material in the muscle ($\mu\text{c}/\text{g}$), Eq. 7.123
c _g	Concentration of isotopes in food-stuffs ($\mu\text{c}/\text{g}$), Eq. 7.116
c	Concentration of radioactive material in the food of an animal ($\mu\text{c}/\text{g}$), Eq. 7.123
D	Dose to an organ (rad), Eq. 7.90

D'	Dose rate to an organ from internal irradiation (rad/sec), Eq. 7.88
βD_d	Beta dose (rad) at a depth, d (g/cm^2), in body tissue, Eq. 7.25
βD_∞	Beta dose from an infinite cloud (rad), Eq. 7.19
$\beta D'_\infty$	Beta dose rate from an infinite cloud (rad/sec), Eq. 7.15
$\beta D'_{\infty b_1}$	Beta dose rate (rad/sec) at a height b_1 (g/cm^2) above an infinite plane upon which w curies/ m^2 of beta-emitting material has been deposited, Eq. 7.26
γD	Gamma-ray dose (rad), Eq. 7.32
γD_0	Gamma dose that would be received in the absence of shielding (rad), Eq. 7.32
$\gamma D'_r$	Gamma dose rate (rad/sec) at a distance r (m) from a point source, Eq. 7.33
$\gamma D'_\infty$	Gamma dose rate from an infinite cloud (rad/sec), Eq. 7.35
d	An effective depth (g/cm^2), Eq. 7.25
$\bar{E}$	Energy absorbed in an organ from one disintegration (Mev), Eq. 7.88
$E_i(\mu r)$	Exponential integral $\int_{\mu r}^{\infty} [\exp(-\mu r)/r] dr$ (dimensionless), Eq. 7.64
E_0	Maximum energy of the beta-ray spectrum (Mev), Eq. 7.25a
$\bar{E}_\beta$	Average beta energy per disintegration (Mev/dis), Eq. 7.15

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$\bar{E}_\gamma$	Average gamma energy emitted at each disintegration (Mev/dis), Eq. 7.33	$\bar{I}_1, \bar{I}_2, \bar{I}_T$	Integrals used in the solution of equations for gamma dose from a continuous release maintained over a long time period, Eqs. 7.61a and 7.61b and Sec. 7-5.2.5
${}_a\bar{E}_\gamma$	Average gamma energy absorbed in an organ per disintegration (Mev/dis), Eq. 7.89		
${}_0\bar{E}_\gamma$	Average gamma energy emitted in an organ per disintegration (Mev/dis), Eq. 7.89	k	Component in the buildup factor $(1 + k\mu x)$ for gamma rays where $k = (\mu - \mu_a)/\mu_a$ (dimensionless), Eq. 7.33
f_a	Fraction of a soluble radionuclide which is translocated to a critical organ (dimensionless), Eq. 7.87	M	Mass of an organ (g), Eq. 7.88
		M_e	Amount of food eaten per day (g/day), Eq. 7.116
f_w	Fraction of an ingested isotope deposited in a critical organ or in a specific tissue (dimensionless), Eq. 7.116	M_g	Mass of edible portion of vegetation (g/m ²), Eq. 7.122
		M_m	Total mass of muscle in an animal (g), Eq. 7.123
f_1	Fraction of radionuclide passing from the gastrointestinal tract to the blood (dimensionless), Eq. 7.87	M_R	Mass of a radioactive material (g), Eq. 7.14
		m	Slope of logarithmic plot of retention of radioactive material in the body or in an organ vs. time, Eq. 7.95
f_2	Fraction of the total body radionuclide burden in the critical organ (dimensionless), Eq. 7.88	N	Number of atoms, Eqs. 7.1 and 7.8
f'_2	Fraction of radionuclide passing from the blood to a critical body organ (dimensionless), Eq. 7.87	N_0	The number of atoms in one gram atomic weight of a substance (6.02486×10^{23} atoms/gram atomic weight), Eq. 7.14
$f(\sigma_z, h, h_i),$ $f_1(\sigma_z, h, h_i),$ $f_2(\sigma_z, h, h_i)$	Factors used in equations describing diffusion beneath a capping inversion (dimensionless), Eqs. 7.51 and 7.52	n	Slope of logarithmic plot of radioactivity from material in the body or in an organ vs. time, Eq. 7.95
$f_B(t)$	Function describing the rate of biological elimination of a mixture of radionuclides from an organ (dimensionless), Eq. 7.91	$P_{\beta 1}$	Radioactive source term in power units for beta radiation (Mev/sec), Eq. 7.22
$f_R(t)$	Function describing the radioactive decay in a mixture of radionuclides in an organ (dimensionless), Eq. 7.91	$P_{\beta 2}$	Radioactive source term in power units for beta radiation (erg/sec), Eq. 7.24
G_1, G_2	Integrals used in the evaluation of the gamma dose from radioactive materials deposited on the ground, Eqs. 7.68 and 7.68a	$P_{\gamma 1}$	Total gamma energy release rate from a source (Mev/sec), Eq. 7.44
		$P_{\gamma 2}$	Total gamma energy release rate from a source (watts), Eq. 7.45
h, h_i	Height of release and inversion base, respectively, (m), Fig. 7.17	Q_0	Initial source strength (curie), Eq. 7.22 (also given as Q). Primes (Q') indicate rate of source release (curies/sec)
I	Gamma-ray intensity (photons/m ² sec), Eq. 7.64	Q_x	Amount of source material remaining in a cloud after a travel distance, x (curie), Eq. 7.22
I_1, I_2, I_T	Integrals used in the solution of the equations for gamma-ray dose from a cloud, Eqs. 7.41a and 7.41b and Sec. 7-5.2.2		

q	Quantity of radioactive material used to express strength of a point source or quantity in the body (curie or μc); appears in various sections and in a variety of uses	λ_R	Radioactive-decay constant equal to $0.693/T_R$ (1/sec), Eq. 7.92
		λ_v	Fraction of an isotope removed from vegetation by radioactive decay, weathering, growth, etc. (1/day), Eq. 7.116
qf ₂	Quantity of radioactive material in a body organ (curie or μc), Eq. 7.88	λ_w	Weathering constant for vegetation (1/day) Eq. 7.123
R	A radial distance (m), Eq. 7.60	λ_1	Elimination constant for short component in lung retention (1/time), Eq. 7.104
S	Source strength per unit area of an infinite plane (photons/m ² -sec), Eq. 7.64	λ_2	Elimination constant for long component in lung retention (1/time), Eq. 7.104
s	Volume of a building (m ³), Eq. 7.83; also a distance between a receptor and a cloud center-line path (m), Sec. 7-5.2.2	μ	Total gamma-ray absorption coefficient ($\mu = \mu_a + \mu_s$) (1/m or cm ² /g), Eq. 7.32
T _B	Biological half-life (time), Eq. 7.92	μ_a	Energy-absorption coefficient for gamma rays (1/m or cm ² /g), Sec. 7-5.1
T _R , T _{1/2}	Radioactive half-life (time), Eqs. 7.3 and 7.14	μ_s	Scattering absorption coefficient for gamma rays (1/m or cm ² /g), Sec. 7-5.1
t	Time (sec, hr); appears in various sections and in a variety of uses	μ_x	Absorption coefficient for wall material (cm ² /g), Eq. 7.74
$\bar{u}$	Mean value of the wind component in the x-direction (m/sec); appears in numerous equations. In practice $\bar{u}$ is usually replaced by the average value of the wind speed indicated by an anemometer	ν	Apparent absorption coefficient for beta radiation in tissue (cm ² /g), Eq. 7.25
V	Ventilation rate in a building (m ³ /time), Eq. 7.83	ρ	Atmospheric density at ambient conditions of temperature and pressure (g/m ³), Eq. 7.17
V _d	Deposition velocity (m/sec), Eq. 7.28	ρ_0	Atmospheric density at S.T.P. (g/m ³), Eq. 7.17
w	Thickness of shielding afforded by building walls (g/cm ²), Eq. 7.74	σ_y, σ_z	Standard deviation of the distribution of material in a continuous plume in the y and z directions (m), Eq. 7.47
α	Normalizing constant in the beta-dose estimation equations, Eq. 7.25	$\sigma_{x1}, \sigma_{y1}, \sigma_{z1}$	Standard deviations of the distribution of material in a puff (instantaneously generated cloud) (m), Sec. 7-5.2.2
η	Factor change in building ventilation rate following cloud passage (dimensionless), Eq. 7.85	τ	A time interval (sec, hr); appears in various sections and in a variety of uses
θ	Angular width of a sector (radians), Eq. 7.60	χ	The concentration at a point at a given instant (curie/m ³), Eq. 7.15
λ, λ_R	Disintegration or decay constant for model in which a constant fraction is removed per unit time (1/time), Eq. 7.1	$\bar{\chi}$	Time-averaged concentration at a point (curie/m ³), Eq. 7.27
λ_B	Biological elimination constant equal to $0.693/T_B$ (1/sec), Eq. 7.92	ψ	Exposure; the integral of concentration over a specified time interval ($\psi = \int_0^{t_{\text{exp}}} \chi dt$ in typical units of curie sec/m ³), Eq. 7.19

ω

Quantity of deposited radioactive material per unit area (curie/ m^2), Eq. 7.26

7-1 INTRODUCTION

In previous chapters the transport and diffusion of materials released to the atmosphere and the mechanisms of removal have been discussed. In radiation-dosage calculations these concepts must be related to the effects that occur at the receptor, whether the receptor be man, animal, or property. Such effects are frequently a complex function of the actual atmospheric distribution of the material in time and space at the location of the receptor. This distribution depends upon the physical properties of the material emitted and the method of release as well as upon the meteorological situation following release. The effects usually depend upon many sociological and biological factors that vary considerably among individuals, groups, and entire populations.

The techniques of radiation-dosage calculation discussed in this chapter are presented as an introduction to some of the major elements in this field. As in any brief résumé of a technological discipline, many topics cannot be discussed and many subtleties are lost. Therefore this chapter must be viewed as a demonstration of the more important aspects of dosage calculation and not as a complete, self-contained working manual.

Releases of radioactive material to the atmosphere can occur almost instantaneously or over a long period of time. The resulting downwind exposure of people and objects to the airborne material will be of short or long duration, depending upon the release time, the wind speed, the nature of the radioactive material, the geometry of the source, and the spreading of the cloud as it travels. In many cases the effects of a radioactive cloud on the receptor are a function of the total radiation received from the cloud with variations in the concentration during the exposure period generally being of minor significance. In this circumstance the integral of the concentration vs. time curve at the point of receipt is a measure of the total effect. The important release parameter thus is the total quantity released

without regard to the rate as long as the diffusion parameters remain constant.

An exception to this generalization will occur when the total radiation dose approximates that dose expected to have an immediate biological effect. For example, a gamma dose to the whole body that could be lethal if delivered within a few hours might be expected to produce less serious consequences if delivered over a period of days or weeks.

With many chemically toxic materials, the maximum allowable concentration in the atmosphere for long-term exposure differs by only one or two orders of magnitude from the concentration that can be expected to produce damage from short-term exposure. It is possible to produce damage by a single inhalation if the concentration is sufficiently high. Since the peak short-term concentration in a cloud can exceed the average concentration by several orders of magnitude (e.g., in high-concentration puffs that reach the ground from a stack during unstable conditions), it may be necessary to consider the effects of instantaneous peak concentrations as well as the effects of average concentrations.

In the event of an accidental release of radioactive materials to the atmosphere, the radiation dose received by people and animals can be considered in two phases: the initial or acute phase while the airborne material is passing the receptor and the latent or recovery phase following the passage of the cloud.

Radiation doses in the initial phase will include the external radiation from the cloud passage, some contribution (before recovery action is taken) from the materials deposited on the ground, body, or other surfaces, and the radiation from any material inhaled during cloud passage. These doses must be considered in conjunction with any other possible doses, such as those from the direct radiation from a nuclear excursion or from fission products retained in a containment vessel. Although the dose from inhaled material will be received over a period of time following the accident, it is included in the acute phase because the total amount inhaled will depend upon conditions in the cloud and not upon later actions.

The dose received in the latent phase will consist of external radiation from material deposited on the ground or other surfaces and

internal radiation from ingestion of foodstuffs or liquids that have become contaminated by the cloud passage. Possibly an additional internal dose could be received during this phase by inhalation of resuspended dust in the contaminated area or from the adherence of such dust to the body. This additional dose would be expected to be significant only in highly contaminated areas.

The doses received in the initial phase will depend upon the location of people or animals and the speed of evacuation as well as upon the quantity and physical characteristics of the material released and the meteorological situation causing the dispersal. The doses received in the latent phase will depend upon administrative actions taken to limit access and to provide protective devices and upon the limits chosen for such actions. The risk or probability of injury to people is therefore a function of the doses received in the initial phase before evacuation; property damage resulting from denying individuals access to or the use of their property will depend upon the residual contamination in the latent phase.

At short time periods after release, before appreciable mixing or diffusion has occurred, the physical limits of the cloud will be quite sharply defined. The cloud will have a limited size and steep concentration gradients. Consequently the effect of the release at near downwind locations may be quite critically affected by minor topographical features (i.e., buildings, hills, and trees), by minor fluctuations in meteorological variables (i.e., wind direction and wind speed), and by release parameters (i.e., finite size of source and rate of release).

As the cloud moves farther from the point of release, continued diffusion will cause the concentration gradients to decrease. At these greater distances the cloud will be relatively insensitive to local irregularities but will be affected by major topographical features, gross changes in meteorological conditions, and depletion by ground deposition en route. The combination of uncertainties in these several parameters can produce major uncertainties in the estimate of effects at all distances downwind.

The greatest overall uncertainty in estimating the effects of a release arises from the dy-

namic character of the atmosphere. Frequent shifts of wind direction and changes in stability and wind speed may cause the plume to move toward unanticipated areas. Present practice is to estimate the dose under several meteorological conditions, while varying the physical and chemical composition of the cloud over a reasonable range, in order to illustrate the potential influence of these factors on the results.

Radiation can be received by a downwind receptor in a variety of ways. The radiation received can be classed as external or internal according to the location of the radioactive material irradiating the tissue. External radiation is received from the radiations emitted as the cloud passes the receptors, from material deposited on surfaces in the region of the receptor, and from material deposited on the body as the cloud passes. As noted earlier, additional external radiation can be received directly from the devices originating the radioactive source.

Internal exposure can result from inhaling the radioactive material in the air as the cloud passes or by ingesting contaminated water or foodstuffs. In each case it is necessary to consider not only the dose to the organs serving as the portal of entry, such as the lungs or gastrointestinal tract, but also the dose to other organs of the body through metabolic transfer of the radioisotopes. Parameters describing the rate of uptake, such as breathing rate and retention in the lung, along with parameters describing the residence time in the specific organ and the transfer between organs must be included in the calculation. The final estimate of internal dose will inherently include the uncertainties in these parameters, as well as those involving the initial air concentration and the movement of the radioactive materials through the ecological systems in the area.

Radioactive material taken into the body by inhalation, by ingestion, or through a wound will continuously deliver a dose to the body (or portions of it) until it has been removed by biological elimination or radioactive decay. The total dose delivered to the person during his lifetime by a specific intake is known; the dose commitment for the incident.

The calculational model presented here is based upon the estimation of internal or ex-

ternal exposure from each important radiation type. The calculations start with the amount and nature of the material released, the air concentration integrated over the time of interest, and the deposition pattern as determined from meteorological conditions. In many cases an exact solution of the geometrical problem thus presented is not warranted by the accuracy with which other factors are known, and an approximate solution may be used.

Calculation of potential doses that could be received during the acute phase of an accidental release will provide an estimate of the time available to evacuate an area or to complete other emergency actions that will minimize or limit exposure to people in the area. The dose estimated for the latent phase will provide a basis for determining the stringency needed in recovery measures and the time period during which such measures may be required.

7.2 SOURCES OF AIRBORNE MATERIALS

An important concept in the relation between plant operations and potential effects on a receptor in the environs is the so-called "source term," which describes the quantities of material released to the atmosphere. Also important, although frequently not so well quantitized, are the physical and chemical characteristics of the material emitted and their influence on the subsequent behavior of the material.

In its broadest sense the source term includes a description of the mode of emission. The material may be released as an almost instantaneous puff or it may be released continuously; the initial source configuration may be represented by a point, a line, or a finite area or volume; the material may be less or more dense than air, or it may have an appreciable exit velocity. The effects of these source conditions on the ultimate disposition of the released materials is important and has been the subject of previous chapters.

Radioactive materials can be released to the atmosphere either routinely during normal operation or as a result of an abnormal operating condition or accident. In general, it is possible to determine the characteristics of routine wastes through direct observation of

operating installations so that the source term under these conditions can be as well known as desired (although some uncertainty may exist in the design of a new facility). For abnormal operating conditions or accidents, however, there may be wide uncertainties as to both the quantity of material emitted and its physical and chemical characteristics.

It is general practice in the nuclear industry to treat off-gases to remove radioactive materials to a very low level during routine operation and to provide backup devices to minimize the probability of release in the event of abnormal conditions. Since it is impossible to remove all unwanted materials from the off-gases, it is necessary to select some limits as bases for equipment design. The lowest feasible concentration measurement is determined by instrument sensitivity and the statistical nature of radioactive disintegrations. Thus estimates of receptor effect are of prime importance in establishing safe and economical limits for discharge.

The quantities and types of radioactive materials reaching the atmosphere are a function of the type of installation, the specific design, and the operating mode. The actual source term can be determined only from the characterization of a particular installation. The following description of sources is general and intended to provide some orientation rather than specific numbers for use.

7.2.1 Classification of Airborne Materials

Atmospheric contaminants can be in the form of gases or particles. Gases are monomolecular distributions of the contaminant formed by evaporation or sublimation; particles are agglomerates of many molecules in crystalline or amorphous form and in the solid or liquid state.

In general, gases will mix in the same manner as the atmosphere in which they are dispersed although at high concentrations heavier gases may have a net settling force owing to the lower density of the surrounding mixture. The deposition characteristics of a gas will depend upon its chemical reactivity and, perhaps, on the difference between the boiling point of the source material and the ambient temperature of surfaces. Noble gases, such as xenon, krypton, or radon, will not deposit on

surfaces although they have limited solubility in water and other fluids. They are eliminated from the atmosphere primarily by radioactive decay. Other gases will deposit with varying degrees of efficiency. For example, elemental ^{131}I will deposit very strongly on many types of surfaces including vegetation when in the form of iodine vapor. Such gases will be eliminated from the atmosphere by deposition, by reaction with or adsorption on natural aerosol particles in the atmosphere, by photochemical reactions resulting in a more reactive compound, by washout with precipitation, or by biological processes. Since the removal rates of these processes depend upon the chemical form of the gas, predictions of interactions in the environs require data on the deposition tendency of the particular effluent and on its change with time under the environmental conditions of the atmosphere. Only a few materials have been studied to any appreciable degree; therefore such estimates require empirical verification.

Particles may be composed of liquids or solids. When in liquid form, they are referred to as mists or, in high concentrations of sub-micron particles, as fogs. Solid particulates may be dusts formed mechanically by grinding, crushing, and drilling; fumes may be formed by combustion, sublimation, and condensation; smokes are usually formed by combustion of organic material. In general, the particle sizes of fumes are less than $1\ \mu$, smokes are usually less than $0.5\ \mu$ (Silverman, 1959) and may range as low as tens of Angstrom units. The general term "aerosol" is used to denote these disperse systems in air in analogy to the term "hydrosol" applied to disperse systems in water.

The fate of aerosols once they are released to the atmosphere depends to a large degree on the dynamic properties of the aerosol which, in turn, depend upon the particle size, shape, density, and, in some important cases, upon the absolute number of particles per unit volume. Among the more important processes influencing such particles are gravitational settling, agglomeration, and impaction. A discussion of some of this information is included in Sec. 5-3 as well as in the following paragraphs.

Gravity will act upon particles to provide a net settling force. Owing to the high surface area per unit mass of these small particles and

the resulting high air resistance, the particles after a short time will reach a constant velocity at which the force of gravity is exactly balanced by the drag forces. This velocity is referred to as the terminal velocity of the particle. The drag forces will vary with the character of the flow around the particle and, hence, with the particle shape and velocity. Figure 7.1 illustrates the approximate terminal velocity for

Fig. 7.1 — Terminal velocities for spheres of various sizes and densities in air at S.T.P.

spheres of several densities as calculated by equations given by Drinker and Hatch (1954). The decrease in slope above about $100\ \mu$ is in the intermediate region between streamline flow and turbulent flow where the equations result from approximate relations between the drag coefficient and the Reynolds number. For particles of other than spherical shape, the drag resistance of the air would generally be increased so that the terminal velocity for an irregular particle containing the same mass as the sphere would be smaller. In the atmosphere the forces exerted on small particles by turbulent motions are large compared with the force of gravity, and consequently such particles can remain suspended for long periods of time. The dependence of fall velocity on atmospheric den-

sity, interpreted in terms of height in the atmosphere, has been shown in Fig. 5.4.

Particles will increase in size by agglomeration when the number of particles per unit volume is very high as in the case of an aerosol generated by volatilization in a limited space. In a freshly formed fume, the particle sizes are very small so that agitation by Brownian movement causes frequent collisions and formation of larger particles through agglomeration. As the particle size increases, the rate of collision and agglomeration decreases, and the larger particles are removed by gravitational settling or, in an agitated atmosphere, by impaction on surfaces. Such flocs of particles frequently have bulk densities considerably lower than the theoretical density of the parent material. This process places an upper limit of about 10^5 to 10^6 particles/cm³ on the number of particles that can exist in a stable aerosol, i.e., one that can last for a reasonable period of time (Magill, Holden, and Ackley, 1956). Agglomeration rate, shape, and bulk density may be affected by the composition of the fluid in which the process occurs, e.g., the presence of steam can have a marked effect. The agglomeration of radioactive material may be different from the agglomeration of stable material because the radiation field associated with the radioactive material will produce ions that can produce statically charged particles and thereby affect the forces of attraction between particles.

When a stream of air containing particles flows around an obstacle in its path, particle inertia will resist the change in motion. If the particle inertia is sufficiently high and the flow direction change is great enough, the particle will not swerve sufficiently to miss the object and will impact upon it. The efficiency of impaction (collision) is defined as the ratio of the number of impacting particles to the number that would have passed through the same area if the object had not been present. The overall collection efficiency of the object will depend upon both the impaction efficiency and the fraction of the impacting particles that are retained. Small particles or particles of lower density will have lower impaction efficiencies than will large or dense particles. Theory predicts that inertial forces for small particles are so low that collection by direct impact becomes negligible but that collection is enhanced for

very small particles by diffusion. Deposition of very small particles may occur owing to electrostatic forces and adsorption. Inertial forces are believed to play an important part in deposition from turbulent mediums such as the atmosphere.

7-2.2 Characteristics of Radioactive Materials

The behavior of radioactive contaminants in the atmosphere is governed by their chemical and physical nature, i.e., the dynamics of transport, diffusion, deposition, and condensation will be nearly the same for these materials as for their nonradioactive counterparts. A reservation as to overall behavior is necessary since the radiation from a radioactive particle produces ionization in the surrounding air and leaves charges on particles and it is possible that this factor could change processes that are dependent upon charge effects. The radioactive properties of individual radioisotopes or particular mixtures of radioisotopes are of importance in determining the quantity and nature of the radioactive materials reaching the receptor as well as the resulting radiation dose.

7-2.2.1 Radioactive Decay. A radioactive atom is characterized by an unstable nucleus that tends to a more stable condition through the emission of a nuclear particle and/or a quantum of energy. If a charged particle is emitted, the resulting atom has a different atomic number and is therefore a different element. Such disintegrations occur according to the laws of chance. The decrease in the number of radioactive atoms in a time period is therefore

$$dN = -\lambda_R N dt \quad (7.1)$$

where N is the number of radioactive atoms, dt is the time period, and λ_R is a proportionality constant called the disintegration constant with the units of reciprocal time. λ_R is a constant for a given radioactive species and represents the fraction of the atoms present which disintegrate per unit time. If Eq. 7.1 is integrated using the condition that $N = N_0$ at zero time, the familiar exponential radioactive-decay equation results.

$$N = N_0 \exp(-\lambda_R t) \quad (7.2)$$

From Eq. 7.2 it can be shown that one-half the radionuclides originally present will decay in a time given by

$$T_R = \frac{\ln 2}{\lambda_R} = \frac{0.693}{\lambda_R} \quad (7.3)$$

where T_R is the half-life of the radionuclide and is characteristic of the particular isotope.

In many cases the original radionuclide (parent) decays to a nucleus that is also unstable (daughter). The number of atoms of the daughter at some time after starting with an initial quantity of the parent (N_1^0 atoms) is given by

$$N_2 = N_1^0 \frac{\lambda_{R1}}{\lambda_{R2} - \lambda_{R1}} [\exp(-\lambda_{R1}t) - \exp(-\lambda_{R2}t)] \quad (7.4)$$

where the subscripts 1 and 2 refer to the parent nucleus and the daughter nucleus, respectively. Equation 7.4 indicates that, if the half-life of the parent is very long with respect to that of the daughter (i.e., if λ_{R1} is small with respect to λ_{R2}), an equilibrium condition is reached after some time, depending upon the half-life of the daughter, where the number of atoms of N_2 are such that the disintegration rate of the daughter equals the disintegration rate of the parent. Since the source of supply of the daughter atoms is the disintegration of the parent, the daughter atoms are, on the average, decaying at the same rate at which they are being supplied. Such a condition is known as secular equilibrium between the parent and the daughter.

$$\lambda_{R1} N_1 = \lambda_{R2} N_2 \quad (7.4a)$$

If the parent has a longer half-life than the daughter, $\exp(-\lambda_{R2}t)$ becomes negligible in comparison to $\exp(-\lambda_{R1}t)$ after some period of time. Equation 7.4 then reduces to

$$N_2 = N_1^0 \frac{\lambda_{R1}}{\lambda_{R2} - \lambda_{R1}} \exp(-\lambda_{R1}t) = N_1 \frac{\lambda_{R1}}{\lambda_{R2} - \lambda_{R1}} \quad (7.5)$$

If expressed as disintegration rate rather than number of atoms, Eq. 7.5 becomes

$$\lambda_{R2} N_2 = (\lambda_{R1} N_1) \frac{\lambda_{R1}}{\lambda_{R2} - \lambda_{R1}} \quad (7.6)$$

This is known as transient equilibrium since the amount of daughter to the parent is a constant although the absolute quantities of the two radioisotopes present are continuously decreasing with time at a rate determined by the decay constant of the parent. Note that the disintegration rate of the daughter exceeds that of the parent once this equilibrium is established.

Such chains in which radioactive daughters produce radioactive second daughters, third daughters, etc. can extend to many numbers. In the chain originating with ^{238}U , for example, there are 10 steps before stable ^{207}Pb is reached. Bateman (1910) gave equations for solving such a radioactive chain of any length with the condition that only the parent material is present at time zero.

$$N_n = N_1 [C_1 \exp(-\lambda_{R1}t) + C_2 \exp(-\lambda_{R2}t) + \dots + C_n \exp(-\lambda_{Rn}t)] \quad (7.7)$$

$$\text{where } C_1 = \frac{\lambda_{R1} \lambda_{R2} \dots \lambda_{R(n-1)}}{(\lambda_{R2} - \lambda_{R1})(\lambda_{R3} - \lambda_{R1}) \dots (\lambda_{Rn} - \lambda_{R1})}$$

$$C_2 = \frac{\lambda_{R1} \lambda_{R2} \dots \lambda_{R(n-1)}}{(\lambda_{R1} - \lambda_{R2})(\lambda_{R3} - \lambda_{R2}) \dots (\lambda_{Rn} - \lambda_{R2})}$$

$$C_3 = \frac{\lambda_{R1} \lambda_{R2} \dots \lambda_{R(n-1)}}{(\lambda_{R1} - \lambda_{R3})(\lambda_{R2} - \lambda_{R3}) \dots (\lambda_{Rn} - \lambda_{R3})}$$

7-2.2.2 Activation. Radioactive materials can be produced by nuclear reactions caused by bombarding the nucleus with subatomic particles, such as energetic alpha particles, deuterons, electrons, or neutrons with considerably lower kinetic energy. Since the nuclear reactor depends upon a large number of neutrons being present during the chain reaction, radioactive materials will be produced in structural members as well as in coolant, gases, or any other material in the vicinity of the reactor core.

Of greatest interest at the present time are activation reactions caused by thermal neutrons, i.e., neutrons with kinetic energy equivalent to that caused by thermal energies, because most of the present generation of reactors operate with a preponderance of neutrons slowed to this energy and because the probability of a nucleus capturing a neutron with this energy is considerably higher than for more energetic

neutrons. Most of the thermal-neutron reactions are of the type in which the nucleus captures a neutron and immediately emits a gamma photon so that the end result is a nucleus of the same element as the parent with a mass of one mass unit greater (n, γ reaction). With fast neutrons the reactions are more likely to produce prompt emission of a particle, such as an alpha particle or one or more neutrons.

Such reactions can be written symbolically as, for example, the reaction of a thermal neutron with ^{40}Ar :

This indicates that an ^{40}Ar nucleus has absorbed a neutron, emitted a quantum ($h\nu$) of gamma radiation, and produced an atom of ^{41}Ar . The ^{41}Ar is unstable and decays by emission of a beta particle with a half-life of 1.8 hr to a stable ^{41}K . Frequently the nucleus resulting from such a reaction is stable so that no radioactive material results.

The measure of the probability of a nucleus capturing a particle and undergoing the reaction is the activation cross section usually designated as σ_A . The cross section will vary with the energy of the bombarding particle and is different for each nucleus being bombarded. Cross sections are expressed in units of barns, which have the dimensions of 10^{-24} cm^2 .

For a material exposed to a given neutron flux (in neutrons/ $\text{cm}^2\text{-sec}$), the number of interactions, and thus the net number of atoms of radioactive material produced (dN_R), in time dt is

$$dN_R = (nv) \sigma_A N_T dt - \lambda_R N_R dt \quad (7.8)$$

where N_T is the number of target atoms exposed and λ_R is the radioactive-decay constant. The second term accounts for the number of radioactive atoms lost by radioactive decay. Solution of this equation with the condition that the number of radioactive atoms present at the start of the irradiation is zero gives

$$N_R = \frac{(nv) \sigma_A N_T}{\lambda_R} [1 - \exp(-\lambda_R t)] \quad (7.9)$$

Thus the number of atoms of radioactive material will increase at a decreasing rate

that is dependent upon the half-life of the radioactive material until an equilibrium is attained between the rate of formation and the rate of decay. At equilibrium

$$N_R \lambda_R = (nv) \sigma_A N_T \quad (7.10)$$

In other words, at equilibrium the rate of decay of the material produced is equal to the rate at which it is formed.

7.2.2.3 Fission. Fission is a special type of nuclear reaction in which the nucleus of the bombarded atom splits into two pieces and releases energy in the form of heat and radiations. Fission can be induced in most elements with an atomic number above about 30 by bombardment with sufficiently energetic particles. Only elements with atomic number above 90 can be fissioned by neutrons of low or moderate energy ($<10 \text{ Mev}$) with appreciable cross sections. Some isotopes are fissionable with thermal neutrons (^{235}U , ^{233}U , and ^{239}Pu are most important); others, such as ^{238}U , require neutrons with an energy of 1 Mev or greater. Of particular interest is the fact that the fission process produces neutrons that in themselves can produce further fissions. Thus it is possible to arrange a mass of fissionable material so that these additional neutrons produce a chain reaction. Since the fission cross section is high for slow neutrons, a thermal reactor is designed to produce further fissions by absorbing much of the energy of the neutrons before they enter the fuel. In a fast reactor the neutrons are utilized with only slight degradation of energy so that the cross sections for fission are lower but the parasitic captures in structural or other materials are also lower. This permits more efficient use of the neutrons and greater freedom in choice of materials for construction.

In the fission process the nucleus is split asymmetrically into two fragments of different mass. In ^{235}U , for example, only about 0.01% of the fissions result in a symmetrical division producing two fragments of identical size.

The average frequency occurrence of each mass chain is referred to as the fission yield and varies to some extent with the energy of the bombarding neutrons and the mass and type of the fissionable nucleus. A fission-yield curve for ^{235}U for thermal neutrons is presented as a

function of the chain mass in Fig. 7.2 (Plutonium Project, 1946, Petruska, Thode, and Tomlinson, 1955, and Bolles and Ballou, 1959). Note that the total yield adds to 200% since the basis is the yield per 100 atoms fissioned, i.e., each fission will yield one atom on the lower part of the mass scale and one atom on the higher part. With high-energy neutrons as the bombarding particles, the pronounced dip between the high-mass and low-mass fractions tends to become less pronounced and practically disappears when 14-Mev neutrons are used.

Fig. 7.2 — Yield of fission-product mass chains from ^{235}U fission by thermal neutrons.

When ^{239}Pu fissions, the low-mass portion of the curve is displaced about four mass units from ^{235}U , but the higher mass region is only slightly affected. This has considerable significance in reactor operation because a sizable portion of the power may be produced by ^{239}Pu fissions in slightly enriched reactors. (The ^{239}Pu is produced by neutron capture in ^{238}U during operation.) As a result, the quantities of strontium and other fission products of similar mass number may be lower than would be estimated from ^{235}U fission, and the quantities of ruthenium and fission products of similar mass number may be higher.

The total amount of energy released per fission is about 200 Mev. Thus the production

of 1 watt of thermal power in a reactor corresponds to a fission rate of 3.1×10^{10} fissions/sec, and a burst of 1 Mw-sec corresponds to a total of 3.1×10^{16} fissions. Since the energy of the explosion of 1 ton of TNT is taken as 10^9 calories (Glasstone, 1962), the fission of 1.3×10^{20} atoms will release an amount of energy equivalent to the explosion of 1 ton of TNT.

The formulas for computing the quantities of individual fission products present are similar to those for neutron activation and decay. For the first member of the chain,

$$\lambda_{R1}N_1 = \gamma F [1 - \exp(-\lambda_{R1}t)] \times \exp(-\lambda_{R1}T) \quad (7.11)$$

where λ_1N_1 = the disintegration rate of the first member

γ = the fission yield of the chain

F = fission rate in fissions per second

t = the time over which the fission rate occurs

T = the time since the fission rate stopped

Similarly, for the second member

$$\lambda_{R2}N_2 = \frac{\gamma F \lambda_{R1} \lambda_{R2}}{\lambda_{R2} - \lambda_{R1}} \left\{ \frac{[1 - \exp(-\lambda_{R1}t)]}{\lambda_{R1}} \times \exp(-\lambda_{R1}T) - \frac{[1 - \exp(-\lambda_{R2}t)]}{\lambda_{R2}} \times \exp(-\lambda_{R2}T) \right\} \quad (7.12)$$

where λ_{R1} and λ_{R2} are the decay constants of the first and second members. These equations assume that the isotope is removed only by radioactive decay. In some cases, particularly with reactors with high neutron fluxes, some fission products with high neutron-capture cross sections may be eliminated by such capture.

Inspection of Eqs. 7.11 and 7.12 indicates that the short-lived isotopes will soon build to an equilibrium level where the rate of decay is equal to the rate of formation by fission. As in the radioactive-decay equations, the time required to reach such an equilibrium depends upon the half-life of the isotope produced. The longer lived isotopes, such as ^{90}Sr or ^{137}Cs , will continue to increase with the life of the core as the reactor operates. As a result, the mixture of fission products in a reactor is

weighted much more heavily toward the longer lived materials than are the fission products from a nuclear excursion or a nuclear detonation.

Way and Wigner (1951) by considering the statistics of fission and resulting fission products derived a useful rule of thumb for the quantities of radioactive materials produced. In their formulation the beta disintegration rate for times in excess of 10 sec following a single fission is given by

$$\beta(t) = 5.2 \times 10^{-6} t_d^{-1.2} \quad (7.13)$$

where $\beta(t)$ is the number of beta particles emitted per second and t_d is the time since fission in days.

Figure 7.3 shows the total quantity of fission products remaining after a fission burst equivalent to an energy release of 10 tons of TNT (1.3×10^{21} fissions) or 42,000 Mw-sec as calculated by Bolles and Ballou (1959). For comparison the decay curve for the gross fission mixture from a typical power reactor operated at 1 Mw for a period of 1000 days at an average power of 10.4 Mw/ton of fuel is included. (Quantities have been normalized to the quantity produced per megawatt.) For the reactor curve the assumption that plutonium replaced ^{235}U was made in order to hold power constant, and it was assumed that 4% of the fissions were caused by fast neutrons.

Table 7.1 presents the inventory of specific fission products for (1) decay periods of 1 day and 90 days following reactor operation of 1095 days and (2) decay periods of 1 day and 100 days following a very-short-term excursion of 4,200 Mw-sec, an energy release one-tenth of that chosen for Fig. 7.3.

7.2.3 Nuclear Power

Nuclear power in this discussion is considered the application of the fission reaction in a nuclear reactor regardless of the purpose.

7.2.3.1 Fuel Cycle. The primary fuel for nuclear reactors is ^{235}U , an isotope which comprises about 0.72% by weight of natural uranium. In some reactors, such as military propulsion reactors, the ^{235}U is separated from the ^{238}U to provide a smaller, more compact source of power. In most reactors, however, the ^{235}U is used with a large quantity of ^{238}U either as slightly enriched uranium or, in the case of the

heavy-water-moderated reactors and graphite-moderated reactors, as natural uranium in the proportions provided by nature.

In the majority of the U. S. reactors, the ^{238}U is enriched 2 to 5% to provide sufficient reactivity to operate with a less efficient but cheaper moderator such as ordinary water. The production of ^{239}Pu is induced by neutron capture in ^{238}U and subsequent decay of ^{239}U to ^{239}Pu . Plutonium-239 is in itself fissionable and can be used as a fuel to replace ^{235}U by separation at the end of the irradiation period and incorporation into a new fuel batch. In a similar fashion thorium introduced into the reactor can produce ^{233}U by capture of a neutron. Uranium-233 is also fissionable and can be used as a fuel.

In most thermal reactors the conversion reaction is relatively inefficient so that the discharged fuel contains only 20 to 50% as much ^{239}Pu or ^{233}U as the initial ^{235}U content. It is possible to raise the conversion to a factor greater than 100% (i.e., more fissionable material is produced than is used). This process is called breeding, and the fast reactors show the greatest promise for providing such multiplication of available fuel.

The entire fuel cycle starts with the mining of uranium (or thorium) and proceeds through the refining of the ore and concentrates to the desired uranium compound. At this point the fuel elements for reactors are fabricated with natural uranium; an enrichment step is interposed for those reactors requiring a higher ^{235}U percentage. After the fuel elements are used in the reactor, the remaining uranium and the plutonium are separated, sent to fuel fabrication, and recycled; fission products are processed for use or storage. A second input to the cycle can occur with thorium or depleted uranium used as targets to produce ^{233}U or additional plutonium.

The fuel cycle is illustrated in Fig. 7.4

7.2.3.2 Airborne Wastes from Fuel-element Fabrication. Uranium and thorium are mined in underground operations or in open pits. Since the uranium and thorium and their daughter products are radioactive, some radioactive wastes will be produced at all steps in the chain of producing fuel or targets. Both uranium and thorium have very long half-lives. Therefore the mass associated with a given quantity of

Fig. 7.3—Quantities of fission products remaining after very short and long term reactor operation.

Table 7.1—DECAY OF INDIVIDUAL FISSION PRODUCTS*

Isotope	Isotope half-life	Reactor (1095 days operation)		Excursion (4200 Mw-sec)	
		Decay, curies/Mw		Decay, curies	
		1 day	90 days	1 day	100 days
⁸⁵ Kr	4.36 hr	140		38	
⁸⁵ Kr	10.57 years	360	350	0.03	0.03
⁸⁹ Sr	53 days	23,000	6,600	21	6.1
⁹⁰ Sr	28 years	2,600	2,600	0.2	0.2
⁹¹ Sr	9.7 hr	5,300		700	
⁹⁰ Y	64.8 hr	2,700	2,600	0.05	0.2
⁹¹ Y	51 min	3,500		300	
⁹¹ Y	58.3 days	30,000	11,000	21	8.3
⁹² Y	3.5 hr	1,100		330	
⁹³ Y	10 hr	7,800		790	
⁹⁵ Zr	65 days	50,000	19,000	270	97
⁹⁷ Zr	17 hr	18,000		950	
⁹⁵ Nb	90 hr	1,000	410	0.09	0.2
⁹⁵ Nb	35 days	51,000	32,000	0.5	12
⁹⁷ Nb	60 sec	17,000		950	
⁹⁷ Nb	74 min	20,000		990	
⁹⁹ Mo	68 hr	40,000		470	
⁹⁹ Tc	6.04 hr	40,000			
¹⁰³ Ru	39.8 days	40,000	8,600	24	4.4
¹⁰⁵ Ru	4.5 hr	610		38	
¹⁰⁶ Ru	1 year	19,000	16,000	0.4	0.3
¹⁰³ Rh	57 min	40,000	8,600	24	4.4
¹⁰⁵ Rh	45 sec	170		38	
¹⁰⁵ Rh	36.5 hr	16,000		130	
¹⁰⁶ Rh	30 sec	19,000	16,000	0.4	0.3
¹⁰⁹ Pd	13.6 hr	2,600		4.1	
¹¹² Pd	21 hr	330		2.1	
¹⁰⁹ Ag	40 sec	2,600		4.1	
¹¹¹ Ag	7.6 days	1,300			
¹¹² Ag	3.2 hr	390		2.4	
¹²¹ Sn	27.5 hr	150		1.8	
¹²⁵ Sn	9.4 days	420	0.6	0.7	
¹²⁵ Sb	2.0 years	200	190	0.0005	0.007
¹²⁷ Sb	93 hr	2,600		10	
¹²⁸ Sb		820			
¹²⁹ Sb	4.6 hr	180		30	
¹²⁷ Te	105 days	580	340	0.015	0.05
¹²⁷ Te	9.3 hr	2,800	330	8.7	5.1
¹²⁹ Te	33.5 days	3,000	460	1.6	0.2
¹²⁹ Te	74 min	3,100	460	31	0.02
¹³¹ Te	30 hr	2,600		45	
¹³¹ Te	25 min	580		46	
¹³² Te	77 hr	38,000		280	
¹³⁰ I	12.6 hr	235			
¹³¹ I	8 days	28,000	13		0.02
¹³² I	2.4 hr	35,000		285	
¹³³ I	20.5 hr	22,000		930	
¹³⁵ I	6.68 hr	4,700		490	
¹³¹ Xe	~12 days	270	4	0.05	
¹³³ Xe	2.3 days	1,500		8.2	
¹³³ Xe	5.27 days	55,000		160	
¹³⁵ Xe	15 min	1,300		160	
¹³⁵ Xe	9.2 hr	14,000		1300	
¹³⁴ Cs	2.1 years	2,000	1,800		
¹³⁶ Cs	12.9 days	910	8		
¹³⁷ Cs	27 years	3,600	3,600	0.1	0.1
¹³⁷ Ba	2.63 min	3,300	3,300	0.1	0.1
¹⁴⁰ Ba	12.8 days	45,000	370	130	0.7
¹⁴⁰ La	40.5 hr	49,000	420	44	0.8
¹⁴¹ La	3.7 hr	750		140	
¹⁴¹ Ce	32.8 days	48,000	7,200	51	6.7
¹⁴³ Ce	33 hr	29,000		700	
¹⁴⁴ Ce	290 days	38,000	30,000	4.9	3.9
¹⁴² Pr	119 hr	900			
¹⁴³ Pr	13.7 days	46,000	550	44	1.0

Table 7.1—(Continued)

Isotope	Isotope half-life	Reactor (1095 days operation)		Excursion (4200 Mw-sec)	
		Decay, curies/Mw		Decay, curies	
		1 day	90 days	1 day	100 days
¹⁴⁴ Pr	17.5 min	38,000	30,000	4.9	3.9
¹⁴⁵ Pr	4.5 hr	1,700		280	
¹⁴⁷ Nd	11.3 days	19,000	72	67	0.2
¹⁴⁷ Pm	2.6 years	7,600	7,400	0.04	0.6
¹⁴⁸ Pm	42 days	620	140		
¹⁴⁸ Pm	5.3 days	4,100	160		
¹⁴⁹ Pm	54 hr	12,000		150	
¹⁵¹ Pm	27.5 hr	3,400		70	
¹⁵³ Sm	47 hr	6,300		16	
¹⁵⁶ Eu	14 days	4,800	78	0.2	0.04

* From Bolles and Ballou (1959), and private communication with I. F. Stuart and A. S. Szegh, August 4, 1966.

radioactivity is high. Although this tends to minimize radiation problems, in the actual fuel the large quantities handled and the association with relatively short-lived daughter products, particularly in the early stages of refining, provide the potential for airborne release of radioactive material.

Airborne dusts containing the parent materials and daughter products are generated, but little is released to the atmosphere. The chief problem in the operation is with the gaseous radon daughter and its decay products. Extensive ventilation is required to limit the concentration in the mines to tolerable levels.

Uranium mill operations include crushing, grinding, and, in some mills, roasting. All these operations are potential sources of airborne dusts or gaseous daughter products. Usually equipment to clean the air is provided along with forced ventilation through stacks. A possible additional source of airborne radioactivity may be the wind pickup and transport of materials from mill stockpiles and tailing piles. Since the milling operations are carried out on the ore with the daughter products in equilibrium with the product, steps in the milling process produce the greatest likelihood that the more hazardous daughter products will be separated and released to the atmosphere.

After the milling operation the concentrates are subjected to further chemical and metallurgical purification to produce essentially pure uranium metal or uranium compounds. Since the majority of the daughter products have been removed earlier, airborne materials from these purification operations consist primarily of dusts or fumes of uranium or thorium gen-

Fig. 7.4—Nuclear fuel cycle.

erated by mechanical or chemical means. These can be controlled by conventional industrial ventilation and air-cleaning equipment. Similarly the fuel-fabrication steps are primarily mechanical, and dusts are essentially the uranium or thorium compounds involved in the process.

The enrichment step is accomplished by converting the uranium to UF_6 (which is gaseous at a slightly elevated temperature) and processing this compound in a gaseous-diffusion plant. The UF_6 is very reactive chemically and, if released, combines with water in the air to produce a cloud of UO_2F_2 and HF . Since impurities can affect the process, care is taken to minimize the possibilities of leaks. Accidental releases are possible, however, in which case the UF_6 would be converted to small particles of uranium compounds and gases that could disperse downwind. If a release occurs at the product end of the process, the higher concentrations of ^{234}U and ^{235}U will result in more radioactivity per unit mass (relative to natural uranium) and a somewhat greater radiation hazard. In the event of a large

release, the chemical toxicity of the fluorine compounds may well be the major hazard.

7-2.3.3 Nuclear Reactors. Radioactive products that could be released to the atmosphere during reactor operation include the fissionable material and the fission and activation products. The actual releases during normal operation depend upon the design of the fuel and the cooling system, and wide variations are possible depending upon the reactor type and use. Reactors can be classified in a number of ways. For the purpose of this chapter, it appears useful to classify reactors by the relation between the cooling system and the atmosphere: the once through direct cycle, the direct cycle, and the indirect cycle. In the first type the coolant (usually air or water) flows through the reactor removing heat and is then discharged to the environment. In the second type the coolant circulates, but the vapor is used directly to drive the energy-recovery system with any entrained gases discharged through the turbine condenser. In the indirect-cycle type, the coolant passes through a heat exchanger and produces steam or other heated gas in the secondary loop which drives the energy-recovery system.

Table 7.2 listing some present or proposed reactors under these classifications indicates the variability of other characteristics possible.

Although corrosion and activation products must be considered a potential source for release of radioactive material during routine operations, the major problem is in maintaining the integrity of the fuel and cladding so that fission products do not escape to the coolant and then to the wastes or the environs. During irradiation the structural changes that occur cause the fuel to swell, thus producing cracks and breaks in the fuel material. With metal fuels operating temperatures are generally limited by phase changes. All of these changes tend to place stresses on the fuel cladding so that eventually the clad may break or lose its integrity. The fuel is designed to operate for a given period with only minimal failures, but even so the escaping fission products may make a major contribution to the radioactivity released in the effluents during normal operations.

In the reactors using once-through cooling systems, any activation or impurities in the coolant, as well as corrosion and fission products picked up by the coolant stream, will be

Table 7.2—REPRESENTATIVE REACTORS CLASSIFIED BY TYPE OF COOLING CYCLE

Use	Coolant	Moderator	Fuel form	Power, Mw(t)	Reactor
Once Through					
Experimental	Air	Graphite	Metal	3.8	X-10*
²³⁹ Pu production	Air	Graphite	Alloy		Britain*
²³⁹ Pu production	Water	Graphite	Metal		Hanford
Experimental	Air	Graphite	Metal	20	Brookhaven
Direct Cycle					
Power	Water	Water	Oxide	To 3000	Boiling-water reactors
Power	Water	Water	Oxide	700	Dresden 1
Power	Water	Water	Oxide	50	Vallecitos Boiling Water Reactor*
Power	Water	Water	Oxide	240	Humboldt Bay
Development (superheat)	Steam	Water	Oxide	12	EVESR, Vallecitos
Power (superheat)	Steam	Water	Oxide	38	BONUS
Power (superheat)	Steam	Water	Oxide	190	Pathfinder
Power	Heavy water	Heavy water	Oxide	575	Marviken (Sweden)
Indirect Cycle					
Power	Water	Water	Oxide	To 3000	Pressurized-water reactors
				505	Shippingport
				600	Yankee
				615	Indian Point
Power and ²³⁹ Pu production	Water	Graphite	Metal		NPR, Hanford
Power	Heavy water	Heavy water	Oxide	694	Douglas Point (Canada)
Power	CO ₂	Graphite	Alloy	1533	EDF-3 (France)
Power	CO ₂	Graphite	Alloy	1453	Dungeness (United Kingdom)
Power	He	Graphite	Carbide	46	AVR (Germany)
Power	He	Graphite	Carbide	116	Peach Bottom
Power	Sodium		Alloy	200	Fermi
Experimental	Sodium		Oxide	20	Southwest
					Experimental Fast Oxide Reactor
Power	Organic	Organic	Alloy	46	Piqua

*Shut down.

discharged in the coolant. In the air-cooled reactors, for example, production of ⁴¹Ar with a 110-min half-life produces a significant source of radioactive material that is released to the atmosphere. In the water-cooled reactors at Hanford, some discharge of noble gases to the atmosphere occurs through stacks or from the open cooling basins as the dissolved gases separate from the water. In the X-10 reactor, a series of fuel-cladding failures caused contamination of the channels and ducts so that a continuous discharge of fission products occurred. This was remedied by filtering the exit coolant. A particular source of possible effluent for the graphite-moderated reactors is the production of ¹⁴C in the moderator and subsequent slow oxidation or exchange with the

moderator coolant. Present use of once-through cooling systems is minimal for reactors of any significant power level owing to the preference to operate at higher temperatures and pressures to conserve expensive coolant and to minimize discharges of radioactive materials.

The direct-cycle plants are best illustrated by the present boiling-water reactors or the superheat reactors, either thermal (in the present generation) or fast (in the future). In these plants the steam or superheated steam flows directly from the reactor to the turbine after which it is condensed and returned to the reactor. A continuous demineralization system is used to remove most of the fission or corrosion products in the coolant. In these direct-cycle plants, some of the gaseous and volatile

radioactive materials carried with the steam are carried with the gases from the turbine condenser and exhausted to the atmosphere. Experience has shown that the phase separation at the surface of the boiling water in a boiling-water reactor is an efficient method of decontaminating the steam for all except the noble gases. For normal operating conditions separation factors of 500 to 1000 have been estimated for iodine. The majority of the more troublesome fission products remain in the system to be removed by the ion exchangers. Somewhat more possibility of escape to the atmosphere occurs with the direct-cycle superheat plants because the coolant is steam and no phase change occurs. Even in the event of serious fuel-cladding failures in the present experimental plants, however, the release to the steam is lower than could be expected and has been controlled by changing the faulty fuel, treating the exit air, or lowering the operating temperature of the faulty fuel.

In the direct-cycle reactor, the primary system is completely contained, and any gases generated are withdrawn for storage and subsequent release. In pressurized-water reactors, the volume of noncondensable gases is reduced because the hydrogen and oxygen produced by radiolytic decomposition are dissolved in the water at the high pressure of the primary system. In the reactors employing liquid sodium as a coolant, the capacity of the sodium to dissolve and retain fission products even at the high temperatures involved has been found to be quite efficient. In particular the sodium appears to retain the radioiodines with high efficiency.

In any of these systems, leaks of the primary coolant with the associated dissolved fission products or activation products produce a potential source of airborne particulates and gases that must be removed by the air-cleaning system before release to the atmosphere. It is difficult to ensure that a water system will remain leakproof at high pressure, and any leakage will result in flashing of the steam with dissolved materials suspended in the air. The liquid-sodium and the heavy-water systems seem to have additional problems in this regard: the liquid-sodium systems because of the necessity of preventing chemical reaction with the air and water and the heavy-water systems because of the expense of the coolant.

Potential consequences of reactor accidents may range from the trivial to the extremely serious. The very large inventory of radioactive materials in a large operating reactor in close association with an almost unlimited source of heat poses the main concern.

There are two types of accidents that could result in major releases of fission products from the core. Smaller variants of these possibly could result in smaller releases.

1. A nuclear excursion in which the chain reaction goes out of control owing to a large, unexpected insertion of reactivity. Sufficient heat is generated to melt or even vaporize the fuel in a short time and thereby release the fission products.

2. A loss of coolant to the fuel after some period of operation so that the residual heat generation by the highly radioactive fission products causes the fuel to melt and release fission products from the core.

In the ultimate accident such mechanisms could result in complete melting of the core and the bulk release of fission products. A more likely case, however, would involve only a portion of the core or the melting of a single fuel bundle after removal from the reactor.

In modern reactors the principles of redundancy and inherently safe design are used to minimize the probability of an accident. For example, dual-control instrument systems are used to minimize the possibility of failure, and the reactors are designed to lose reactivity with increasing temperature. Redundant emergency coolant systems are also provided. In addition, a containment or confinement system surrounds the reactor so that any fission products escaping from the core will be contained in the building and the release to the atmosphere will be minimal.

The source term for a reactor accident can vary greatly, depending upon the type of accident and the severity. The time period over which release occurs can also vary considerably, depending upon the pressures involved and the heat sources, heat sinks, and coolant capacity available. In the extreme case where a containment is filled with fission products, sudden failure of the containment could result in a ground-level puff release. If the containment remains intact, a continuous release extending over an appreciable period of time could be expected.

Quantitative data on the fission products that might be released to the atmosphere are meager owing to the uncertainties of the physical conditions expected in a given accident and the lack of experimental data on the behavior of the isotopes. In estimating the release to the atmosphere, both the release from the fuel and the subsequent behavior of the released materials in such processes as agglomeration, impaction, electrostatic attraction, absorption, adsorption, and desorption should be considered. Unfortunately present knowledge permits only qualitative consideration of these processes, and even this is speculative because the environment in which the accident occurs is poorly defined.

An effort is currently being made to obtain experimental data concerning the release and deposition of fission products. The processes involved are complex, however, and accidental conditions are difficult to simulate. Current estimates can be considered only as useful generalizations until much more data have been obtained.

Some fission products may be released from fuel at temperatures below melting if the cladding is destroyed. In general, the noble gases and iodines will comprise the major portion of this release. If the fuel is melted, essentially all the noble gases and most of the iodines will be released. The volatile solids, such as cesium, tellurium, and ruthenium, under oxidizing conditions may be released to about the same extent as iodine. The more refractory fission products, such as strontium and the rare earths, may be largely retained in the melt. In an accident serious enough to volatilize the fuel, however, the bulk of the fission products may be released from the fuel.

Immediately after release the fission products will tend to cool, and particle sizes may increase by agglomeration. The nongaseous fractions will be subject to deposition on surfaces. The extent of the deposition will depend upon the exact pathway of escape, the flow velocities, the temperatures, the characteristics of the atmosphere, and similar factors. Investigation of these factors has been primarily directed at ^{131}I in large reactor facilities. Here it is noted that some fraction of the iodine will rapidly deposit on walls and other surfaces and a smaller fraction will deposit at a much

slower rate. Iodine appears in several combinations of chemical and physical form, and these probably change with time. These factors will produce a reduction in the quantities of material that may become airborne. However, the data are not available to estimate with any degree of certainty the quantity of material that will remain airborne following any reactor accident.

7-2.3.4 Separation Plants. After discharge from the reactor, the spent fuel still contains a significant quantity of ^{235}U plus quantities of ^{239}Pu in low-enrichment reactors and ^{233}U if thorium is present. Both ^{233}U and ^{239}Pu can be used as fuels if they are separated from the fission products and reworked into new fuel elements. This is accomplished by a chemical separation process in which either the uranium mixture is dissolved in acid and the uranium is then extracted with solvents or, in proposed processes, the uranium is volatilized by producing uranium hexafluoride and the plutonium and fission products are left to be treated separately.

Even though the fuel is stored for periods ranging from 100 days to 1 year or longer to provide time for radioactive decay, a large quantity of highly radioactive material is still present when the fuel reprocessing begins. During the initial step of dissolution or reaction to produce uranium hexafluoride, the noble gases and a part of the ^{131}I left after the decay period are released. The chief noble gas will be ^{85}Kr with a half-life of 10.76 years. Each ton of metal from a present-day power plant will contain about 5000 curies of ^{85}Kr that will be released to the atmosphere. If the cooling time is on the order of 100 days, significant quantities of ^{133}Xe with a 5.2-day half-life will also be released. Although the ^{85}Kr emits only a soft beta radiation with a minimum of gamma radiation, the ^{133}Xe radiation has a significant gamma component and can deliver external radiation to individuals in the environs.

Of the fission products released to the atmosphere, ^{131}I is of particular interest because of its rapid deposition and inclusion in the food cycles of the region. Since its half-life is 8 days, cooling periods of several hundred days permit time for it to essentially disappear by radioactive decay. If shorter cooling periods are used, various techniques are applied to

remove the iodine from the gaseous effluent before discharge to the atmosphere.

Tritium is also produced in fuel elements during power operations. More than 300 curies of ^3H per ton of fuel may be released during processing, and the bulk of it is discharged to the environs.

In addition to these radioisotopes, a general background of other fission products results from leaks and dusts that become airborne. Banks of high-efficiency particulate air (HEPA) filters are used in the ventilation system to maintain fission products at a reasonable level.

In some plants selected fission products are separated for use as radiation or radioisotope heat sources. Such special separations include ^{90}Sr , ^{137}Cs , and ^{144}Ce . The operations can also produce radioactive aerosols that must be controlled by filtration.

Other possible sources of airborne wastes are the storage tanks used for the waste fission products and the equipment used to immobilize the fission products by incorporating them in an insoluble medium. In some cases the heat generation in storage tanks due to the high fission-product concentration may be sufficient to cause boiling. Again, treatment of the exit air is necessary.

Generally, separations plants require massive shielding so that the accidents will not cause a loss of plant integrity or excessive exposure to employees. One potential accident of special concern is the redispersion of radioactive materials collected in one of the air-cleaning devices, e.g., a fire in a charcoal filter containing iodine or in an air filter containing other fission products.

7-2.4 Other Sources

The widespread use of radioisotopes in research, industry, and medicine has created a variety of potential sources ranging from a large number of applications involving small quantities of a few radioisotopes to a few large laboratories where many different radioisotopes are produced and used in large quantities. At each of these installations, at least potentially, a discharge of radioactive materials into the atmosphere can result in a multiple-source problem with low source strength. In general, controls on the amount emitted consist of air-treatment facilities and monitoring.

Shipments of radioisotopes or spent fuel might, in an accident, constitute another source of airborne radioactivity. Although shipping containers are designed to withstand most accidents, the use of air transportation or the potential for involvement of the shipment with large quantities of explosives poses the possibility that the shipping cask might be breached and a portion of the contents melted or vaporized.

The use of radioisotopes and reactors for auxiliary power sources and in propulsion systems has added a new dimension in accident potential because these systems often exist in unusual environments or operate under unusual stresses. The safety problems are further complicated by the uncertainties in predicting the dispersion of the material (particularly for space missions) and in evaluating the biological effects of the various fuel forms.

7-3 RADIOLOGICAL CONCEPTS

7-3.1 Radiations

The ionizing radiations of concern in potential problems with atmospheric waste disposal or a reactor accident are alpha radiation, beta radiation, and gamma radiation.

7-3.1.1 Alpha Radiation. Alpha radiations are particles that are physically identical with the nuclei of helium atoms and are ejected from the nucleus of certain radioactive atoms with a kinetic energy of 4 to 10 Mev. Alpha particles have a mass of approximately 4 amu (6.642×10^{-24} g) and a positive charge of two resulting in strong interactions with atoms encountered. In these interactions a portion of the alpha-particle energy is imparted to the electrons of the atoms of the material traversed with consequent ion-pair production (a negative electron and an associated positive ion). The large number of such ion pairs produced per unit of track length by the alpha particle (30,000 to 100,000 ion pairs per centimeter in air) cause a rapid loss of kinetic energy and, consequently, a short range. Alpha particles emitted from most radioactive materials will travel only 1 to 8 cm in air, depending upon their energies. Since alpha particles of these energies will generally be stopped by the inert surface layer of skin, alpha emitters present no problem of external radiation but can produce serious damage if emitted inside the body where such a protective layer does not exist.

7-3.1.2 Beta Radiation. Beta radiations are electrons emitted from the nucleus of a radioactive atom with an energy of 0.02 to 3.2 Mev which imparts a speed approaching that of light. Because of their lower mass and charge, ionizing interactions between beta particles and the atoms of other materials are more infrequent than for alpha particles (≤ 200 ionpairs per centimeter in air). The consequent lower rate of energy loss permits a longer range of up to several meters in air and several centimeters in tissue. External beta radiation is important because it will penetrate the protective layer of the skin and produce damage to the underlying sensitive tissues if the kinetic energy is above about 100 Kev. It will not penetrate to the deeper-seated organs of an animal as large as man; so damage is largely confined to the surface layers of the body, including exposed organs such as the eyes. The slowing of the high-speed beta-ray electrons in material (bremsstrahlung) gives rise to electromagnetic radiation, but generally such radiation is significant only with very intense and energetic beta-ray sources. Owing to the limited range of the beta radiation, the dose rate is sensitive to the local concentration, and wide variations can be encountered because of local shielding, pockets developed by increased deposition from eddies, or later translocation and concentration by runoff from a rain storm.

7-3.1.3 Gamma Radiation. Gamma radiations are electromagnetic rays emitted from the nucleus of radioactive atoms. They are indistinguishable from X rays, and both X rays and gamma rays are included in dose calculations. Interactions of gamma photons passing through matter are less frequent than for charged particles; consequently their penetrating ability is much greater. Such interactions occur at random with the drop-off in intensity of the primary beam following an exponential relation complicated by the reentry of certain scattered components into the beam. The relaxation length (distance to decrease the intensity by a factor of $1/e$) for typical gamma rays (~ 0.7 Mev) is on the order of 100 m for air, 50 cm for water, and 5 cm for lead. As a result the gamma-ray dose at any location is a function of the distribution of the radioactive material over a relatively large area or volume.

7-3.2 Radiation Units

Special units have been defined to permit measurement of radiation relative to the effects it produces. The damage to tissue is related to the amount of energy deposited by the radiation in the tissue in the form of ionization and excitation of the molecules with subsequent degradation to heat. Therefore many of the radiation units are closely related to energy deposition.

7-3.2.1 Rad. The rad is the unit of absorbed dose and is defined as the deposition of 100 ergs/g of any absorbing material [International Commission on Radiological Units and Measurements, 1962]. The amount of energy delivered to a body will depend on the nature and degree of the interaction between the radiation and the body irradiated. It is therefore possible to have different absorbed doses in different materials subjected to the same radiation flux as, for example, when the same masses of lead and tissue are exposed to a gamma flux. An obsolete unit frequently found in older literature is the rep, which was defined as the absorption of 93 ergs/g in soft tissue.

7-3.2.2 Roentgen. The roentgen (r) is the unit of exposure (formerly, exposure dose). The past definition gave the roentgen as the quantity of X or gamma radiation such that the associated corpuscular emission per 0.001293 g of air (1 cm^3 at S.T.P.) produces, in air, ions carrying 1 esu of electricity of either sign (International Commission on Radiological Units and Measurements, 1959). A recent report (International Commission on Radiological Units and Measurements, 1962), defined exposure as the quotient $\Delta Q/\Delta M$, where ΔQ is the sum of the electrical charges on all the ions of one sign produced in air when all the electrons liberated by photons in a volume element of air with a mass of ΔM are completely stopped in air. The symbol Δ indicates that averaging must be done in a volume that is small enough so that a further reduction in its size would not appreciably change the measured value of the ratio but is still large enough so that it contains many interactions. The unit of exposure is given by

$$1 \text{ r} = 2.58 \times 10^{-4} \text{ coulomb/kg}$$

Since the roentgen is a measure of the interaction of gamma radiation and air, the absorbed dose (in rads) will vary in different materials from the same exposure (in roentgens). With moderate-energy gamma rays (0.2 to 3 Mev), an exposure of 1 r will produce an absorbed dose in muscle of about 0.97 rads.

7-3.2.3 Dose Equivalent. The dose-equivalent concept is similar to the former RBE dose. All radiations do not produce identical biological effects for a given amount of energy delivered to tissue, i.e., for equal absorbed doses. In general, radiations with high specific ionization along their tracks will produce a greater effect, but the quantitative degree of difference will depend upon the biological change chosen for study. The relative biological effectiveness factor (RBE) is used to compare the effectiveness of absorbed doses of radiation delivered in different ways. It has been recommended (International Commission on Radiological Units and Measurements, 1962) that the term RBE be used only in radiobiological work and that the term quality factor (QF) be used for the linear-energy-transfer dependent factor for radiation-protection purposes. Where appropriate, provision is also made for other factors, such as the distribution factor. The dose equivalent (DE) is now defined as the product of the absorbed dose, the quality factor, the dose-distribution factor, and other necessary modifying factors. The concept of quality factor has a limited usefulness since the biological effectiveness depends on many factors, including the type and degree of biological damage, the absorbed-dose rate, the fractionation of the dose, and the conditions of delivery. It is not possible to determine an accurate value of the quality factor for a given radiation as compared to another which will describe all effects, such as lethality, carcinogenic action, damage to specific tissues, etc.

The dose equivalent is numerically equal to the product of the dose in rads and an agreed-upon conventional value of the quality factor which describes the effects of concern. The standard of comparison is X or gamma radiation having a linear energy transfer in water of 3.5 Kev/ μ and a rate of about 10 rad/min.

The unit of dose equivalence is the rem. In addition to the uncertainties introduced by the concept of quality factor, the rem assumes

values of the quality factor determined by convention rather than direct measurement. Such values are normally chosen on the basis of long-term effects that may occur over the lifetime of the individual and are therefore applicable to long-term effects rather than the more pronounced immediate effects from a large absorbed dose. For radiation from internal emitters, the following quality factors (QF) are used to convert from absorbed dose to dose equivalent:

Internal radiation	QF
β^- , β^+ , γ , X	
$E_{\max.} > 0.03$ Mev	1
$E_{\max.} < 0.03$ Mev	1.7
α	10
Fission fragment and α recoil	20
Neutrons from spontaneous fission	8

In the case of exposure to mixed radiations, the dose equivalent is assumed to be equal to the sum of the dose equivalents (in rems) of each of the radiations involved.

7-3.2.4 Curie. The curie is the special unit of activity (International Commission on Radiological Units and Measurements, 1962). It was earlier the unit of quantity of radioactive material. The activity is given by $\Delta N/\Delta T$, where ΔN is the number of nuclear transformations that occur in the quantity of material of concern in time Δt . The symbol Δ indicates that averaging must be done in a time interval that is short enough so that a further reduction in the interval would not appreciably change the measured value of the ratio but is still long enough so that it contains many disintegrations. One curie is that quantity of a radioactive nuclide in which the number of disintegrations (transmutations) per second is exactly 3.7×10^{10} .

The curie is related to the mass of a radioactive material through the half-life and the number of atoms per unit mass (atomic weight). The mass of material associated with 1 curie varies widely. For example, one curie of ^{24}Na (half-life, 14.8 hr) weighs 10^{-15} g, 1 curie of ^{226}Ra (half-life, 1622 years) weighs 1 g, and 1 curie of ^{238}U (half-life, 4.5×10^9 years) weighs 3×10^6 g. The relation between the curies and the mass of a specific radionuclide is given by

$$q = \frac{\lambda_R M_R N_0}{A (3.7 \times 10^{10})} \quad (7.14)$$

where q = number of curies

λ_R = disintegration constant (fraction of the atoms that decay per second) = $0.693/T_{1/2}$

$T_{1/2}$ = half-life of the radionuclide (sec)

M_R = mass of radionuclide (g)

N_0 = Avagadro's number: 6.03×10^{23} atoms/gram atomic weight

A = atomic weight of the radionuclide

Frequently there is some confusion about the use of the term "curie" when it is applied to mixtures of radionuclides. In this book the term will be used to describe the total disintegration rate of the mixture, i.e., the sum of the activities contributed by the specific radionuclides of the mixture when more than one radionuclide is involved. Many radionuclides emit more than one gamma ray per disintegration or a beta particle and one or more gamma ray per disintegration, and care must be exercised when determining the radiation flux from a source (in curies) or when determining the source (in curies) that produces a given flux.

The relation between the quantity of radioactive material (curie) and the radiation dose rate (rad/sec) is dependent upon the energy of the radiation emitted, the type of radiation emitted, the geometrical pattern between the radioactive material and the receptor, and the amount of absorbing material between the radioactive material and the receptor.

7-3.3 Radiation Measurements

Measurement devices used in radiation protection are designed to measure the radiation dose or dose rate or to measure the quantity of active material present by the rate at which radiation is emitted. Although a large number of measurement devices are available, the three in principal use are the ionization chamber, the particle counter, and the photographic film.

An ionization chamber in its simplest form consists of a pair of charged electrodes that collect ions formed within their electrical field. Many of the electrons actually entering the gas in the chamber are the result of secondary interactions in the chamber walls; consequently the chamber response is governed

by the wall material. Properly designed and calibrated chambers are usually used for dose or dose-rate measurements because they provide an indirect measure of the energy deposited in the chamber.

Counters are designed to record the passage of a single particle, primary or secondary, through the sensitive volume of the counter. Gas-filled counters may simply collect the ionization produced by the radiation and amplify it to produce a pulse. Amplification may be produced by the acceleration of the electrons and their subsequent collisions with molecules in the gas to produce a pulse proportional to, but greater than, the initial pulse (proportional counter); or a large amplification may be produced by providing conditions such that the pulse spreads through the entire sensitive volume and a complete discharge between the electrodes results (Geiger-Mueller counter). A scintillation counter registers electrical pulses from a phototube. These pulses are initiated by the small flash of light produced by the radiation interacting with the phosphor or crystals in the vicinity of the tube.

Counters are usually used to determine the activity present by measuring the number of particles or photons emitted by the radioactive material. Scintillation or proportional counters can be used to determine dose or dose rates if the equipment is designed to measure the energy delivered to the receptor. Geiger counters are frequently used as radiation detectors because of their sensitivity and can be used to measure dose rate if they are calibrated for the energy and type of radiation to be measured.

Photographic film will darken upon exposure to radiation and can be used to give a measure of the quantity of radiation present. Although the film response is energy dependent for gamma rays below 0.5 Mev, this can be corrected by filters to about 0.2 Mev. Film is frequently used in measuring personnel exposure and in other dose measurements where a long integration time period is required or a permanent record is desired.

7-3.4 Effects of Radiation

The effects of a given dose of radiation depend upon a large number of factors, including the magnitude of the absorbed dose, the type of radiation, the penetrating power of the radia-

tion, the radiosensitivity of the organism for the effect involved, the rate at which the dose is delivered, the portion of the organism irradiated, and the specific organs involved.

The qualitative aspects of radiation effects are well known from past experience with human exposure and extensive studies with animals, but the quantitative relation between a dose delivered under given conditions and the resulting biological change is not so well defined for man. Only a few of the effects will be discussed here in order to illustrate the magnitudes of doses involved.

A massive exposure delivered in a relatively short time can produce immediate or acute effects, which in their most severe form can cause the death of the organism. For man it is estimated that a X- or gamma-radiation dose of 400 to 500 rads delivered to a major portion of the body in a short time interval will result in the death of about 50% of those so exposed within a few months (Braestrup and Wyckoff, 1958). Most mammals have about the same degree of sensitivity to the dose required to give 50% lethality in 30 days (LD 50) ranging from perhaps 275 r for swine to 800 r for the rabbit (Patt and Brues, 1954). Sublethal exposure of humans to doses above 100 to 200 r will produce nausea, vomiting, diarrhea, and malaise with the symptoms developing earlier for the higher exposures (Glasstone, 1962). A loss of hair might occur about 2 weeks after an exposure of greater than 300 r (Nickson and Bane, 1959).

Skin reactions will develop if the dose is sufficiently high. Such effects are of primary importance in exposure to beta radiation or low-energy gamma radiation which can deliver very high doses to the skin with comparatively small doses to the remainder of the body. Doses of 400 to 1500 rads to the skin will result in an erythema (reddening) in several days and possible loss of hair in about 2 weeks (Nickson and Bane, 1959). The hair will return, but there is a possibility that it will be gray. At the higher end of this dose range, there may be dry scaling of the surface layers in about 3 weeks. Above 1500 rads an erythema will develop, and raw moist areas will form. Above 5000 to 7000 rads, these areas may develop into areas that heal very slowly or not at all.

Late effects of radiation exposure are those which occur some time after the exposure or

as a result of continued low-level exposure over a period of time. These effects are detected by statistical review of the incidence of the effect among those exposed compared to the incidence in a similar unexposed group; i.e., these effects are not an invariable result of exposure as with the acute effects but rather express themselves as an increase in probability of the individual's contracting a specific ailment. The degree of increase in probability depends upon the radiation dose, the time of exposure, the organs exposed, and other parameters.

The increased incidence of certain malignancies with radiation exposure has been well documented both in experimental animals and in man. The ingestion of radium by the dial painters in New Jersey between 1916 and 1925 claimed 41 victims over a period of about 20 years (Furth and Lorenz, 1945). Bone sarcomas developed in individuals, a few of whom had 20 μ g of radium or more in their bodies. The increased incidence of leukemia among the Japanese surviving the nuclear detonations over Hiroshima and Nagasaki (Wald, 1958) and among certain patients treated with radiation (Court-Brown and Doll, 1957) has indicated that leukemia can be produced by single exposures on the order of 100 r. Similar results have been noted in a study of American radiologists who received radiation over a period of time during the days before the institution of present protection methods (March 1944). Similarly numerous skin cancers have developed among early workers with X rays where the majority of the exposure was to the hands. Direct evidence on the production of malignancies by lower doses is not available, and it is not known whether this is due to the difficulty of detecting a small increase over the normal incidence or to a possible threshold value of dose below which the effect will not occur.

Radiation is but one of the mutagenic agents to which man is exposed. Such agents can produce changes in the germ cells which can be passed to children and may produce abnormalities in future generations (World Health Organization, 1957). These effects will not appear in the person irradiated but may result in harmful changes in the future populations. Since such changes result from intermixing of the genes, an important criterion of possible radiation damage in addition to the exposure to the individual is the number of individuals

exposed. An estimate of the average dose over the reproductive lifetime of the individual which is required to double the mutation rate is between 10 and 100 rads (United Nations, 1958).

Other effects that occur later include cataract formation and shortening of the lifespan. Cataracts have been noted from single clinical exposures of over 175 r (Merriam and Facht, 1958). With the radiation delivered over a period of 3 months, the lowest exposure associated with a cataract was 550 r. Life shortening from radiation exposure has been noted in animals where no specific injury was responsible for the death. Evidence for such an effect in man has been presented for the early radiologists (Warren, 1956) but has been questioned owing to the difficulty of obtaining an adequate control group for comparison (Lewis, 1957).

7-3.5 Radiation Limits

Shortly after the discovery of X rays in 1895, evidence of the harmful effects of indiscriminate exposure to radiation began to accumulate, and recommendations for limitations of exposure were made. In 1928 the International Commission on Radiological Protection (ICRP) was organized under the auspices of the Second International Congress on Radiology. In 1929 the Advisory Committee on X-ray and Radium Protection was organized to develop recommendations in the United States. Following World War II, this Advisory Committee was reorganized as the National Committee on Radiation Protection (NCRP) with representatives from professional societies and governmental agencies, as well as individual experts. The recommendations developed by this group have served as the basis for most radiation-protection programs and, later, for rules and codes adopted by the various regulatory agencies in the United States. In 1959 the Federal Radiation Council (FRC) was formed to advise the President on radiation matters and to provide guidance for all federal agencies in the formulation of standards of protection against radiation damage.

Each of the groups that have recommended exposure limits has written lengthy reports that include a discussion of the factors taken into consideration in arriving at the recommendations and limits. These reports must be

read fully and carefully to understand the recommendations as they were intended by the authors. Only deliberate study of the complete reports can provide the basis for judgment on the adequacy of a program for radiation protection. When limits are enumerated, they are intended as limits and not design criteria for an apparatus or activity. All groups have stressed the fact that there are multiple sources and modes of exposure and have assumed a linear relation between dose and effect. All groups have recommended keeping exposure to the minimum practical and have called for a balance of risk and benefit to justify those exposures which cannot be avoided. A balance of benefit and risk is difficult because they cannot be adequately reduced to a common denominator and considerable uncertainty is involved on both sides of the "scale". In the final analysis professional judgment is required, and the basis for the judgment should be sound.

7-3.5.1 Occupational Exposure. The original recommendations of the NCRP and the ICRP were aimed at controlling the exposures of individuals working with radiation and were primarily based upon effects that had been observed in early radiation workers. Following World War II a large body of information became available from the animal experiments performed in connection with the Manhattan Project. Review of these data, plus continued review of the extensive work since that time, has led to the present recommendations for the control of occupational exposure.

The maximum permissible doses (MPD) for radiation workers as recommended by the ICRP (International Commission on Radiological Protection, 1959) and NCRP (National Committee on Radiation Protection, 1958) are in basic agreement although there are some minor differences.

Organ	MPD (annual), rem
Gonads, red bone marrow, and whole body	5*
Skin, thyroid, and bone	30
Hands and forearms, feet and ankles	75
All other organs	15

*The cumulative dose of $D = 5(N-18)$ rem should not be exceeded. Here D (rem) is the cumulative dose and N (years) is the age of the individual.

These exposure levels are well below those observed to cause damage, and they are not expected to cause appreciable bodily injury to a person continuously so exposed during his working years. However, the assumption that the relation between risk and dose is linear leads to the conclusion that any radiation will result in an increase in the risk for the development of certain somatic and genetic effects.

Limitations for internal emitters are based upon the radiation dose delivered to specific organs or, in the case of bone-seekers, are based upon a comparison with the energy delivered by radium. Radium is the only internal emitter for which the available experience with man is adequate to establish a maximum permissible limit from direct evidence of damage. In the derivation of limits for internal emitters, the maximum permissible quantity in an organ is calculated from the organ size, the energy release in the organ, and, where appropriate, the distribution of the isotope in the organ (National Committee on Radiation Protection and Measurement, 1959). This quantity of radioactive material in the organ provides the basic limitation because it describes the radiation actually received by the organ. As guides to the conditions that could result in exposure of the organ to the maximum permissible dose, secondary numbers called maximum permissible concentrations of the radioisotope in air and water are derived by considering the uptake and translocation of the isotope in the various organs.

To provide a standard basis of calculation, the ICRP has defined a so-called "standard man" in terms of the intake of air and water, retention of particulates, and organ weights (International Commission on Radiological Protection, 1959a). All maximum permissible concentrations refer to this idealized model. It should be noted that this standard man is a hypothetical individual and that specific people vary significantly from this standard.

Use of standard-man values does not provide an estimate of the greatest dose that might be experienced by any specific individual (or increment) of the general population, but it does, generally, provide an overestimate of the doses that might be experienced by the average industrial worker.

The uncertainties in the metabolic parameters are such that the estimation of the dose

to individuals and the estimation of the resulting effects can be only order of magnitude at best. Important parameters of the standard man are given in Table 7.3.

Table 7.3—SELECTED PARAMETERS OF THE STANDARD MAN

Organ weights			
Total body			70,000 g
Skeleton:			
Without bone marrow			7,000 g
Red marrow			1,500 g
Yellow marrow			1,500 g
Contents of GI tract:			
Lower large intestine			150 g
Stomach			1,100 g
Small intestine			135 g
Upper large intestine			1,700 g
Liver			1,500 g
Lungs			700 g
Kidneys			300 g
Spleen			150 g
Testes			30 g
Thyroid			20 g
Water intake in food and fluids			2,200 g/day
Total air inhaled per day			2×10^4 cm ³

Distribution	Retention of particles*	
	Readily soluble compounds, %	Other compounds, %
Exhaled	25	25
Deposited in upper respiratory passages and later swallowed	50	50
Deposited in lungs	25†	25‡

Portion	Constants for GI tract		
	Mass of contents, g	Time of food arrival, hr	Time of food leaving, hr
Lower large intestine	150	13	31
Small intestine	1100	1	5
Upper large intestine	135	5	13
Stomach	250	0	1

*A new lung model is being developed to replace that described here. Current literature should be consulted for possible changes.

†Taken up into the body.

‡One-half eliminated from lungs and swallowed in first 24 hr. Remaining 12.5% retained in lungs with a half-life of 120 days except for plutonium and thorium for which the biological half-life is assumed to be one year and four years respectively.

With knowledge of the way the various radionuclides are assimilated by the body and intake values for the standard man assumed, one can calculate the concentrations of specific radionuclides in water or air which, if ingested or inhaled continuously by the individual during a

50-year working lifetime, would limit the critical organ exposure to the maximum permissible dose rate. For most radionuclides taken at a uniform rate over a 50-year period, the body will attain an equilibrium quantity of the radionuclide, i.e., balance between intake and elimination by both radiological decay and biological modes. However, certain long-lived radionuclides that are retained by the body for a long time (^{90}Sr , ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , etc.) do not attain equilibrium in 50 years, and the quantity accumulated in the body will continue to increase. In either case the total quantity of the radionuclide in the body when the critical organ is being exposed at the MPD rate is the maximum permissible body burden. The concentrations of the radionuclides in air and water, calculated as described above, are the maximum permissible concentrations in air (MPC)_a and water (MPC)_w, respectively. Typical values of the limits for several internal emitters for occupational exposure as derived by the NCRP and ICRP are given in Table 7.4 (National Committee on Radiation Protection and Measurement, 1959).

Table 7.4—NCRP LIMITS FOR SELECTED INTERNAL EMITTERS

Isotope	Critical organ	Body burden, μc	MPC (industrial worker, 168-hr week), $\mu\text{c}/\text{cm}^3$	
			Water	Air
^{89}Sr	Bone	4	10^{-4}	10^{-6}
^{90}Sr	Bone	2	10^{-6}	10^{-10}
^{105}Ru	GI		8×10^{-4}	2×10^{-7}
	Kidney	20	3×10^{-2}	3×10^{-7}
^{106}Ru	GI		10^{-4}	3×10^{-6}
	Kidney	3	4×10^{-3}	5×10^{-6}
	Lung		2×10^{-9}	2×10^{-9}
^{131}I	Thyroid	0.7	2×10^{-5}	3×10^{-6}
^{137}Cs	Total body	30	2×10^{-4}	2×10^{-6}
^{239}Pu	Bone	0.04	5×10^{-5}	6×10^{-13}

The critical organ in Table 7.4 is considered to be the organ that receives the highest radiation dose after the isotope is absorbed into the body and is therefore assumed to be the organ most likely to be damaged. In some cases, particularly when the isotope is in a form not readily assimilated into the body, the highest dose is received by the organs at the point of entry, i.e., usually the gastrointestinal tract or the lung. In such cases a maximum permissible concentration is derived on the basis of the estimated dose to these organs.

7.3.5.2 Exposure of Populations. Individuals not working with radiation can be exposed in a variety of ways: e.g., the radiation from cosmic rays or natural radioactive materials in the earth; medical or dental uses of X rays; television tubes; radium dials; fallout from nuclear detonations; waste disposal from nuclear plants; accidents with radiation sources. An estimate of the relative importance of some of these sources to the population as a whole is given in Table 7.5 (Parker, 1960).

Table 7.5—THIRTY-YEAR PER CAPITA DOSES IN THE UNITED STATES FROM RADIATION SOURCES OTHER THAN FALLOUT

Source	30-year dose	
	Gonad dose, rem	Bone dose,* rem
Natural radiation		
External		
Cosmic rays	0.9	0.9
Terrestrial radiation	2.1	2.1
Atmospheric radiation	0.06	0.06
Internal		
^{40}K	0.6	0.3
^{14}C	0.06	0.06
Rn-Th	0.06	0.06
Ra		1.2
Subtotal	~3.8	~4.7
Man-made radiation		
Diagnostic X ray	3.0†	1.4
X-ray therapy	0.36	†
Internal isotopes	<0.03	†
Luminous dials	0.03	†
Television	<0.03	<0.03
Atomic energy workers	0.003	0.003
Medical workers	0.03	†
Subtotal	~3.5	~1.4
Approximate total	7 to 8	5 to 7

*Taken here as the osteocyte dose. This principally affects the radium entry, which would be 0.05 rem for mean marrow dose.

†Additional studies on limited groups have given values on the order of 1.5 rems.

‡No estimate.

All groups that have made recommendations on radiation exposure of the general public have recommended values lower than those for occupational workers. A decrease by a factor of 10 for individuals in the vicinity of an area controlled because of the presence of radiation has been used by the NCRP (National Committee on Radiation Protection and Measurement, 1959) and the ICRP (International Commission on Radiological Protection, 1966). Further, the ICRP specifies that these limits

apply to the "appropriate critical group" within the general public which, because of age, diet, activities, etc., receives the highest dose in the particular environment. As an alternative to defining this group, the commission recommends applying an appropriate safety factor to the dose limits for the general public. The dose limit applied to the thyroid of children less than 16 years of age is 1.5 rem/year because of the damage susceptibility. A gonad dose limit of 5 rem in 30 years is specified for populations in consideration of genetic effects. The Federal Radiation Council (FRC) gives a radiation protection guide of 0.5 rem/year of whole-body radiation for individuals with an average of 0.17 rem/year for appropriate population groups to ensure that no individual exceeds 0.5 rem/year. In each case exposure from medical applications or natural background is omitted.

Controls on internal emitters must also consider the amount of radiation received by the gonads. The ICRP has suggested that the MPC of isotopes that irradiate the whole body or the gonads be reduced by 1/100, a factor that allows for concurrent external exposure when applied to whole populations. For other isotopes the commission recommends a reduction factor of 1/30.

period of time. Intake within this range must be evaluated in terms of the RPG, and, if necessary, appropriate positive control measures should be instituted.

Range limits for several isotopes as given by the FRC are listed in Table 7.6 (Federal Radiation Council, 1961).

7-3.5.3 Emergency Radiation Limits. The radiation limits given earlier are applicable to routine situations that are amenable to control of the exposure. Exposure to higher radiation levels does not mean that damage will result. In emergency situations, where the radiation rates are high, it may be impossible to control the exposure to any preset level either because the radiation is received before action is instituted or because action involving radiation exposure is required to save lives or otherwise control the situation. Firm limits to be followed in all such situations cannot be derived since any exposure under these conditions involves a judgment on the value of a given action as opposed to the possible damage that would result from the exposure. The NCRP, in a study of this situation, states: "It is obviously not possible to spell out every... circumstance; and it is certainly not the prerogative of any committee to assign relative values

Table 7.6—RANGES OF TRANSIENT RATES OF INTAKE FOR USE IN GRADED SCALE OF ACTIONS

Isotope	Range I, $\mu\text{c/day}$	Range II, $\mu\text{c/day}$	Range III, $\mu\text{c/day}$
^{226}Ra	$\leq 2 \times 10^{-6}$	2×10^{-6} to 2×10^{-5}	2×10^{-5} to 2×10^{-4}
^{131}I	$\leq 1 \times 10^{-5}$	1×10^{-5} to 1×10^{-4}	1×10^{-4} to 1×10^{-3}
^{90}Sr	$\leq 2 \times 10^{-5}$	2×10^{-5} to 2×10^{-4}	2×10^{-4} to 2×10^{-3}
^{89}Sr	$\leq 2 \times 10^{-4}$	2×10^{-4} to 2×10^{-3}	2×10^{-3} to 2×10^{-2}

The FRC (Federal Radiation Council, 1961) has published guides for normal use based on daily intake with three ranges of concern:

1. Range I would not be expected to result in a large fraction of the Radiation Protection Guide (RPG) for an appreciable fraction of the individuals. Only surveillance is needed.

2. Range II would not be expected to result in average exposures exceeding the RPG. Active surveillance and routine control are needed.

3. Range III exposures would be expected to exceed the RPG if continued for a sufficient

to high-priority tasks on the one hand, and to the radiation exposure involved in doing so on the other hand" (National Committee on Radiation Protection and Measurements, 1962). The NCRP defines a radiation emergency as: "... an accident, or other event out of the ordinary, that threatens to expose people to more than 25 r in 1 week. It is not intended to include maintenance and reconstruction activities following a peacetime accident after the hazard to the general population has passed. It is expected that most radiation accidents in peace-

time can be handled satisfactorily under the principles set forth in Handbook 59" (National Committee on Radiation Protection and Measurement, 1962).

Limits per se are not pertinent to emergency conditions during which the risk of acute exposure and immediate harm is present. Instead, such situations must be guided by individuals acquainted with the risks involved and the importance of various tasks.

The FRC in reviewing the problem of emergency conditions noted that "Protective actions are appropriate when the health benefit associated with the reduction in dose that can be achieved is considered sufficient to offset the undesirable factors associated with the action" (Federal Radiation Council, 1964). The FRC also provided guidance in applying Protective Action Guides (PAG). They proposed:

1. If the projected dose exceeds the PAG, protective action is indicated.
2. The amount of effort that properly may be given to protective action will increase as the projected dose increases.
3. The objective of any action is to achieve a substantial reduction of the dose that would otherwise occur—not to limit it to some prespecified value.
4. Proposed protective actions must be weighed against their total impact. Each situation should be evaluated individually. As the projected dose becomes less, the value of protective actions becomes correspondingly less.
5. The Protective Action Guide is based on the assumption that the occurrence, in a particular area, of environmental contamination that would require protective action is an unlikely event. Circumstances that involve either repetitive occurrence or in which there appears a substantial probability of recurrence within a period of one or two years would require special considerations. . . . In contemplating the possibility of a future event, it is necessary to consider not only the possible magnitude but also the probability that the event will occur.

The FRC then recommends the use of a value of 30 rads to the thyroid of an individual as the PAG for ^{131}I . As an operational technique, it is considered that the PAG will not be exceeded if the average dose to the thyroid of a suitable sample of the population does not exceed 10 rads. A suitable sample is considered to consist of children about 1 year of age using milk from a reasonably homogeneous supply.

7-4 EXTERNAL BETA RADIATION

Because beta radiation has a limited range (1 to 10 m in air), the beta dose depends mainly upon the concentration of radioactive materials in the immediate vicinity of the individual. Since any variations in the local concentration will result in a corresponding variation in the dose, the prior estimates of beta doses can be regarded only as order of magnitude estimates of the average dose, particularly in those cases in which the radioactive materials settle very close to, or on, the person.

7-4.1 Beta Dose from Passing Clouds

A cloud of radioactive materials that decay through beta emission will constitute a source of beta radiation. The importance of this source will depend upon the energies of the radiations emitted, the size of the cloud, the concentration in the cloud, and the wind speed.

7-4.1.1. Beta Dose from Infinite Clouds. Consider a spherical cloud with a uniform concentration of a beta-emitting isotope and a radius equal to the range of the beta particles. Such a cloud would be considered an infinite cloud for a receptor at the center because any additional beta-emitting material beyond the cloud dimensions would not alter the flux of beta particles to the receptor. For such a cloud the beta dose rate can be calculated from equilibrium considerations provided the receptor volume is small and does not perturb the radiation field.

For an infinite uniform cloud containing χ curies of beta radiation per cubic meter, the beta dose rate in air at the cloud center is given by Eq. 7.15:

$$\beta D'_{\infty} = \frac{\bar{E}_{\beta} (1.6 \times 10^{-6}) (3.7 \times 10^{10}) \chi}{(1293) (100)} \quad (7.15)$$

$$\beta D'_{\infty} = 0.457 \bar{E}_{\beta} \chi \quad (7.16)$$

where $\beta D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec)

$\bar{E}_{\beta}$ = average beta energy per disintegration (Mev/dis)

1.6×10^{-6} = number of ergs per Mev

χ = concentration of beta-emitting isotope in the cloud (curie/m³)

3.7×10^{10} = disintegration rate per curie (dis/sec curie)

1293 = density of air at S.T.P. (g/m³)

100 = energy absorbed per gram of absorbing material per rad (erg/g-rad)

If the air is at other than standard density, Eq. 7.16 can be corrected by the ratio of the air densities, thus

$${}_B D'_\infty = 0.457 \bar{E}_B \chi \left(\frac{\rho_0}{\rho} \right) \quad (7.17)$$

where ρ_0 = density of air at S.T.P. (g/m³) and ρ = density of air at reference conditions (g/m³). Although applicable throughout, this correction factor for air density will not be repeated in subsequent equations.

By the time a cloud or plume has reached the receptor, the dimensions of the cloud are generally on the order of tens of meters. Under these conditions the cloud can be considered infinite in extent for any point within the cloud, and the error introduced by this approximation will be well within the uncertainties caused by other factors.

For these conditions of infinite cloud size, if $\chi_{(x,y,z,t)}$ is the instantaneous concentration of the cloud at the receptor location at time t (sec) after the release, the external beta dose rate ${}_B D'_{\infty(x,y,z,t)}$ to the receptor at any time t after release is given by

$${}_B D'_{\infty(x,y,z,t)} = 0.457 \bar{E}_B \chi_{(x,y,z,t)} \quad (7.18)$$

where $\chi_{(x,y,z,t)}$ has been corrected for depletion by ground deposition and radioactive decay en route.

The total direct beta dose ${}_B D_{\infty(x,y,z)}$ due to cloud passage is given by

$${}_B D_{\infty(x,y,z)} = 0.457 \bar{E}_B \psi_{(x,y,z)} \quad (7.19)$$

where $\psi_{(x,y,z)}$ = concentration time integral (curie sec/m³) at the receptor location and is corrected for depletion en route

$$= \int_{t_a}^t \chi_{(x,y,z,t)} dt$$

t_a = time of cloud arrival (seconds since release)

t = time of cloud departure (seconds since release)

The meteorological formulas generally used to determine the cloud concentration or con-

centration time integral are developed for nonradioactive source material and do not include correction for ground deposition. When the source material is radioactive, correction factors must be included to account for the depletion by radioactive decay en route to the receptor. When a single radionuclide is involved, the correction is simply

$$\left(\frac{Q_x}{Q_0} \right)_R = \exp \left(-\lambda_R \frac{x}{\bar{u}} \right)$$

where $(Q_x/Q_0)_R$ = the reduction in quantity of radioactive material due to radioactive decay

x = downwind distance between the point of release and the receptor location (m)

$\bar{u}$ = average wind speed during transit (m/sec)

λ_R = radioactive-decay constant (sec⁻¹)

When the source term includes several radionuclides, a single exponential equation is inadequate, and a power function or other empirical decay factor may be required to describe the depletion by concurrent decaying of several radionuclides. In this case the correction factor may be indicated symbolically as

$$\left(\frac{Q_x}{Q_0} \right)_R = f_R(t)$$

The correction factor for depletion due to ground deposition (see Sec. 5-3.2.2) is similar to Eq. 5.48,

$$\left(\frac{Q_x}{Q_0} \right)_d = \exp \left[-\left(\frac{2}{\pi} \right)^{1/2} \left(\frac{v_d}{\bar{u}} \right) \int_0^x \frac{\exp(-h^2/2\sigma_z^2)}{\sigma_z} dx \right]$$

The total depletion correction term is the product of the two factors:

$$\frac{Q_x}{Q_0} = \left(\frac{Q_x}{Q_0} \right)_R \left(\frac{Q_x}{Q_0} \right)_d$$

In this chapter, in which meteorological dispersion formulas are used to calculate radiation dose values, the symbols for cloud concentration (χ) and cloud concentration time integral (ψ) at the receptor location are assumed to include the appropriate depletion correction factors.

The value of $\bar{E}_\beta$ in Eqs. 7.15 and 7.19 is the average kinetic energy of the beta particles released per disintegration. Beta particles are emitted from radioactive material with a continuous spectrum of energies ranging from zero to a maximum value characteristic of the specific material. The average of the beta-ray spectrum varies from about 0.25 to 0.45 of the maximum energy for negative electrons, depending upon the atomic number of the radionuclide and the maximum energy (Hine and Brownell, 1956). In view of the accuracy with which other parameters are known, it is usually adequate to consider the average beta energy for a specific radionuclide to be one-third of the maximum beta-ray energy. In a cloud composed of a number of radioisotopes, it is necessary to obtain an overall average beta energy by considering the fraction of each species present and its average energy of emission.

Equations 7.15 and 7.19 describe the beta dose rate and dose resulting from an essentially infinite cloud where media of different absorbing characteristics are not present. In the actual case the presence of the body in the cloud causes a perturbation to the radiation field. Owing to the very limited range of beta particles in tissue, the beta dose to a body surrounded by a beta source will generally be limited by geometrical considerations to irradiation from half the cloud. Protruding organs, such as the nose, ears, and fingers, may, however, receive radiation from greater than a 2π solid angle. For this reason the beta dose or dose rate at the surface of a body immersed in an infinite cloud will be about one-half the equilibrium beta dose (or dose rate) of the full cloud. In addition, the surface of the earth provides a further perturbation in the radiation field. The dose rate at the surface is only one-half of that in an infinite cloud but increases to the full infinite-cloud dose at an elevation equal to the maximum range of the beta particles involved. Therefore dose rates will vary from the head to the foot of an individual with the highest dose rate at the head. This factor varies from 0.5 at the ground to 1.0 at heights greater than the range of the beta radiation. This correction factor will not be included in subsequent formulas although specific corrections can be made for particular

elements of tissue if the radiation energy and height of concern are defined.

The surface body dose from beta emitters in the infinite cloud can be approximated as

$$\beta D_{\infty(x,y,z)} = (0.5) (0.457) \bar{E}_\beta \psi_{(x,y,z)} \quad (7.20)$$

$$\beta D_{\infty(x,y,z)} = 0.23 \bar{E}_\beta \psi_{(x,y,z)} \quad (7.21)$$

where $\psi_{(x,y,z)}$ is the concentration time integral for the cloud (curie sec/m³) corrected for depletion by ground deposition and radioactive decay en route. If the source (release of radioactive material) is known in terms of power units, e.g., $P_{\beta 1}$ (Mev/sec) or $P_{\beta 2}$ (erg/sec), the beta dose (rads) to the surface of the body from the passing infinite cloud can be given by

$$\beta D_{\infty(x,y,z)} = \frac{(0.5) (1.6 \times 10^{-6})}{1293 \times 100} \left(\frac{\psi_{(x,y,z)}}{Q} \right) P_{\beta 1} \quad (7.22)$$

$$\beta D_{\infty(x,y,z)} = 6.2 \times 10^{-12} \left(\frac{\psi_{(x,y,z)}}{Q} \right) P_{\beta 1} \quad (7.23)$$

where $[\psi_{(x,y,z)}/Q]$ is the time-integrated dilution factor (sec/m³) and $P_{\beta 1}$ is the source term (Mev/sec) emitted by the entire cloud at the time of release.

When the source term $P_{\beta 2}$ is in units of ergs per second, the equation for the beta dose becomes

$$\beta D_{\infty(x,y,z)} = 3.86 \times 10^{-6} \left(\frac{\psi_{(x,y,z)}}{Q} \right) P_{\beta 2} \quad (7.24)$$

Similar equations can be written for other beta power units, e.g., watts.

In Eqs. 7.20 through 7.24, the concentration and energy release rates depend upon the source characteristics and the amount of radioactive decay and ground deposition that has occurred between the time of release and the time of exposure. Whether the release occurs over a short period of time or is continuous, the change in quantity due to radioactive decay during the actual passage time of the cloud is generally small compared to the change during transit between the release point and the receptor. In this case a single correction term has been applied to ψ to correct for the radioactive decay.

In cases where the release consists of a mixture of radionuclides and the time of exposure to the radiation from a passing cloud is long with respect to the half-life of the radioactive material, it is desirable to correct for decay during passage. In this case the source term is not constant, and the concentration integration over time must include a term describing the time variation of the source, i.e., $\bar{E}_\beta \psi$ or $(\psi/Q) P_\beta$ must be replaced by $\int (\bar{E}_\beta \chi) dt$ or by $\int (P_\beta \chi/Q) dt$. This, in turn, may be evaluated from considerations of the individual radionuclides present by fitting an arbitrary mathematical function to the rate of decay and integrating or by calculating the dose rate at various times during cloud passage and averaging over the exposure time.

7-4.1.2 Depth Dose. The beta dose calculated in the preceding section is the dose received at the surface of the body or clothing. Before damage is produced, the radiation must penetrate the clothing and the normal inert layer of skin and reach the sensitive tissues underneath. A complete estimate of the beta-dose effects must evaluate the dose to the underlying tissues.

From measurements of the distribution of dose in air around a point source of beta radiation, Loevinger, Japha, and Brownell (1956) have derived an empirical equation that permits calculation of the dose as a function of the depth of penetration and energy of the beta radiation. The integration of this expression to give the dose above an infinite slab can be used to provide an estimate of the depth dose because the cloud will act as an infinite slab in contact with the body.

$$\beta D_d = \beta D_\infty \alpha \left\{ c^2 \left[3 - \exp \left(1 - \frac{\nu d}{c} \right) - \frac{\nu d}{c} \right] \times \left(2 + \ln \frac{c}{\nu d} \right) + \exp (1 - \nu d) \right\} \quad (7.25)$$

where the quantity in square brackets equals zero when $d \geq c/\nu$.

In Eq. 7.25 βD_d is the dose at a depth of d (g/cm^2 in the body) from a surface dose of βD_∞ , and c is an energy-dependent parameter with the following values:

$$\begin{aligned} c = 2 & \quad 0.17 \text{ Mev} < E_0 < 0.5 \text{ Mev} \\ = 1.5 & \quad 0.5 \text{ Mev} \leq E_0 < 1.5 \text{ Mev} \\ = 1 & \quad 1.5 \text{ Mev} \leq E_0 < 3 \text{ Mev} \end{aligned} \quad (7.25a)$$

where E_0 is the maximum beta energy emitted and α is a normalizing constant given by

$$\alpha = [3c^2 - (c^2 - 1)e]^{-1} \quad (7.25b)$$

The values of α for various values of c are

c	α
1	0.333
1.5	0.297
2	0.260
3	0.190

In Eq. 7.25 ν is an apparent absorption coefficient derived from measurements in polystyrene with a correction for the electron density of tissue. It is expressed in units of square centimeters per gram so that the product νd is dimensionless.

$$\nu = \frac{37.2}{(E_0 - 0.036)^{1.37}} \quad (7.25c)$$

An additional term to correct for anomalous spectra with certain beta emitters has been omitted in Eq. 7.25c because the correction is not of significance for the accuracy of these calculations.

The dose at a given depth in the tissue will be strongly dependent upon the energy of the beta particles emitted. For a mixture of different emitters, it is necessary to evaluate the dose from each emitter separately or to calculate for various energy groupings and weight the result by the fractional occurrence of each group. Figure 7.5 illustrates the dose to be expected at various depths in tissue from exposure to a cloud with a concentration time integral (ψ) of $1.0 \text{ curie sec}/\text{m}^3$ and with the beta energies shown. For a sustained concentration of $1.0 \text{ curie}/\text{m}^3$, Fig. 7.5 will give the dose rate in rads per second from an infinite cloud. The depth is expressed in milligram per square centimeter, a unit that expresses the mass available for absorbing radiation to the thickness given. For tissue with a density of one, a thickness of $100 \text{ mg}/\text{cm}^2$ corresponds to a linear depth of 1 mm.

The inert thickness of the epidermis varies over the body but is usually considered to be $7 \text{ mg}/\text{cm}^2$. For most calculations the surface dose given by Eq. 7.21 will estimate the dose

Fig. 7.5—Beta doses at various depths in tissue produced by an exposure of 1.0 curie sec/m³ from an infinite cloud containing a beta-emitting radionuclide.

for the unclothed part of the body to within the accuracy of the knowledge of the local concentration.

7-4.2 Beta Dose from Material Deposited on the Ground

During the passage of the cloud, a portion of the radioactive material will be deposited on the ground by dry deposition, rainout, or wash-out. This material can consist of any of the radioactive materials in the cloud except the

noble gases. In general, the pattern of deposition calculated from the dispersion equations will consist of a central line of high deposition with the amount per unit area decreasing on either side in a Gaussian distribution similar to the pattern of concentration in the cloud. In practice there will be localized dose rate variations due to runoff and puddling of rain water, increased deposition caused by eddies near obstructions, variable patterns from wind-shifts, deposition of large particles that cannot remain airborne, increased accumulation in oils or

greases, etc. The present knowledge of the detailed diffusion mechanisms for a particular terrain is not adequate to calculate these localized variations in any detail. The calculations of the radiation field above the ground provide the general area dose rates with the localized variations due to these factors superimposed upon this general field. In dry weather, differences in the rate of dry deposition due to particle sizes, physical form of the material, and other factors will cause fractionation; so the composition of the deposited material may vary with distance. In general, the beta dose from the separate fractions with different deposition constants must be evaluated separately and then summed.

The material deposited on the surface can be considered as a thin source that, because of the short range of the beta particles, is effectively infinite in the horizontal plane. Under these conditions the variation of dose rate above the ground is given by Loevinger, Japha, and Brownell (1956) as

$$\beta D'_{\omega(b,t)} = 0.0297 \nu \bar{E}_{\beta} \alpha \omega \left[c \left(1 + \ln \frac{c}{b\nu} \right) - \exp \left(1 - \frac{b\nu}{c} \right) \right] + \exp(1 - b\nu) \quad (7.26)$$

where the expression in brackets equals 0 when $b > c/\nu$. The term $\beta D'_{\omega(b,t)}$ is the beta dose rate (rad/sec) at height b (g/cm²) above an infinite plane upon which ω curies/m² of beta-emitting material has been deposited (1 cm³ of air weighs 1.293×10^{-3} g), ω is surface activity (curie/m²), and values of the other parameters are as given previously.

Where the cloud is a plume of radioactive material from the release of a single radionuclide at a constant rate of Q'_0 (curies/sec) from an elevated point h_d (m) above the ground plane, the concentration of the established plume at a location $(x,y,0)$, when corrected for depletion en route, is given in curies per cubic meter by

$$\bar{\chi}_{(x,y,0)} = \frac{Q'_0}{\pi \bar{u} \sigma_y \sigma_z} \left(\frac{Q_x}{Q_0} \right)_d \exp \left(-\lambda_R \frac{x}{\bar{u}} \right) \times \exp \left(-\frac{y^2}{2\sigma_y^2} - \frac{h^2}{2\sigma_z^2} \right) \quad (7.27)$$

The activity deposited on the surface, $\omega_{(x,y,0)}$ (curie/m²) is given by

$$\omega_{(x,y,0)} = \bar{\chi}_{(x,y,0)} v_d \int_0^{t_d} \exp(-\lambda_R t) dt \quad (7.28)$$

where v_d is the deposition velocity (m/sec) and t_d is time interval of discharge (sec). The net activity on the ground after the first part of the plume reaches the point $(x,y,0)$ will be given by

$$\omega_{(x,y,0)} = \bar{\chi}_{(x,y,0)} v_d \frac{[1 - \exp(-\lambda_R t_d)]}{\lambda_R} \quad (7.29)$$

At times after the last of the plume has passed the point, the deposited activity of a single radionuclide will again decay according to the simple exponential, and the activity per unit surface area will be given by

$$\omega_{(x,y,0,t)} = \bar{\chi}_{(x,y,0)} v_d \times \frac{[1 - \exp(-\lambda_R t_d)] \exp(-\lambda_R t)}{\lambda_R} \quad (7.30)$$

where t is the time interval (sec) after the last of the plume has passed the point $(x,y,0)$.

For the special case of an instantaneous release of Q_0 curies, i.e., $t_d \rightarrow 0$, the quantity $[1 - \exp(-\lambda_R t_d)] \rightarrow \lambda_R t_d$, and $Q_0 = Q'_0 \Delta t$, Eq. 7.30 becomes

$$\omega_{(x,y,0,t)} = \frac{Q_0 v_d (Q_x/Q_0)_d}{\pi \bar{u} \sigma_y \sigma_z} \exp \left[-\lambda_R \left(t + \frac{x}{\bar{u}} \right) \right] \times \exp \left(-\frac{y^2}{2\sigma_y^2} - \frac{h^2}{2\sigma_z^2} \right) \quad (7.31)$$

where t is the time interval (sec) after the cloud center has passed the point $(x,y,0)$.

Figure 7.6 presents the dose rate at several approximate distances above the ground (rounded off for simplicity) for a uniform deposition of 1.0 curie/m². For estimations of the total beta dose received during a given period of occupancy, the dose rate shown (with an appropriate correction for radioactive decay) should be multiplied by the time of occupancy or integrated over the appropriate time interval.

Note that the beta dose rate will vary when the material is deposited on surfaces other than the ground. For example, the dose rate from fission products on tall bushes, on buildings, or in regions of very rough terrain will differ from that calculated by Eq. 7.26, which is based upon a plane surface.

Fig. 7.6—Beta dose rate in air above a plane surface with a uniform contamination of 1.0 curie/m². Curves are labeled with the depth of air in terms of length and mass. The curves will apply to the dose rate in tissue if the density of tissue is substituted for the density of air.

7-4.3 Beta Dose Rates from Materials Deposited on the Body

Radioactive materials deposited directly on the skin can deliver relatively high beta doses because of their proximity to sensitive tissues. Although some gamma dose is usually associated with the beta dose, the longer range of the gamma radiation results in spreading the absorbed energy over a larger volume, whereas the short range of the beta particles results in intense energy deposition in the first few millimeters of the skin.

The physical form of the deposited material can range from a uniform coating over the entire surface to the random distribution of a few small very intensely radioactive particles, depending upon the characteristics of the material in the passing cloud. These single particles are usually found close to the point of discharge or, in the event of a contaminated surface, in dust resuspended by the action of the winds. Such particles act as discrete sources of radiation and provide very-high dose rates to the tissue in the immediate vicinity of the particle. One can estimate dose rates of 10^6 rad/hr at distances of 10μ from a 1μ c particle. The beta dose rate, however, drops rapidly with distance from the particle; so the total amount of tissue involved is very small. The significance of such estimated doses is not clear in determining the hazard because the available correlations with biological damage are poor. For this reason the equations for estimating the dose or dose rate from discrete particles are not included, and the reader is referred to the literature for material on the estimation of dose and effects (George, Healy, and Bustad, 1955; Loevinger, 1956; Passonneau, Brues, Hamilton, and Kisielski, 1952).

The amount of radioactive material deposited on the body will vary over the body surfaces. It is to be expected that the greatest deposition will occur on the hair and on any oily spots on the body. The feet and legs may also become contaminated if the individual moves around in the contaminated area. The estimation of the quantities of material deposited on the body can be made only with poor accuracy, but for an average estimate it can be assumed that the deposition velocity for the transfer of radioactive material from the passing cloud to the body is the same as for transfer of material to the

ground. The beta dose rate to the body can be calculated from Eq. 7.26, where b is the depth within the body rather than the height in air above the ground. In addition, the dose rate at various depths within the tissue can be obtained from Fig. 7.6 if the depth in tissue is expressed in units of mg/cm^2 .

Figure 7.7 presents the approximate beta dose rate that would occur through the $7\text{ mg}/\text{cm}^2$ protective layer of the skin immediately

Fig. 7.7—Approximate beta dose rate through the $7\text{ mg}/\text{cm}^2$ of protective skin layer from materials deposited on the body for various deposition velocities following an exposure (ψ) of $1.0\text{ curie sec}/\text{m}^3$ for various β energies.

after cloud passage as a result of deposition velocities ($v_d = \omega/\psi$; see Sec. 5-3 of Chap. 5) ranging from the low values expected for small particles (and perhaps organic compounds of iodine) in light winds to the much higher values that might apply for iodine in the vapor state in moderate winds. The beta dose rate for beta emitters of various energies and for a considerable range of deposition velocities can be obtained directly from the graph.

In estimates of beta doses, the shielding effects and protection provided by clothing may be taken into account. The maximum beta dose will be delivered to the exposed surfaces of the skin. Clothing will attenuate the beta radiation dose because of its shielding effect and the fact that the clothing will hold the deposited material away from the skin.

7-5 EXTERNAL GAMMA RADIATION

Although the principles and methods of calculating the dose from extended sources of gamma radiation are well-known and straightforward, the solution for actual cases is the most complicated of the cloud-dosage calculations. As a result, the gamma dose from a passing cloud can be evaluated with good accuracy only for specific conditions.

With extended sources, such as cloud or ground deposits, the gamma dose in air at a given point depends upon the location of the radioactive materials for distances up to several hundred meters. The radiation field is therefore more uniform than with beta radiation because local variations in the material distribution are averaged over a much larger area or volume.

The use of the equations developed in this section (and in other sections) should be approached with considerable caution because many assumptions and approximations are necessary to obtain numerical solutions. Many factors in a specific application may cause significant differences between the measured doses or dose rates and those calculated by the formulas presented.

7-5.1 The Point Source

Calculations of the gamma dose from an extended source start with consideration of the radiation received at the receptor from a differential area or volume that can be regarded as a point source. This radiation is then integrated over the entire source with account being taken of the geometry of the source, variation in concentration, attenuation by interactions of the photons with matter in the path between source and receptor, and scattering of radiation from material outside the direct path to the receptor.

For a point source the geometrical considerations provide an inverse-square decrease in the dose with distance. Since the total radiation emitted (minus that absorbed) will pass through a sphere of radius equal to the distance between source and receptor, the flux passing through any area at that distance can be determined if the initial emission is known.

Gamma photons are absorbed in matter by three discrete processes: (1) the photoelectric effect in which the total energy of the photon is passed to an orbital electron with consequent ejection of the electron from the atom, (2) Compton scattering in which the photon undergoes an elastic collision with an electron with consequent ejection of the electron from the atom and a scattered photon of lower energy, and (3) pair production in which the photon in the electric field of a nucleus or an electron is converted to an electron and a positron. The photoelectric effect occurs with low-energy photons and high-atomic-number materials. For lead the photoelectric effect is important up to energies of about 1 Mev; for aluminum, up to about 0.1 Mev. The Compton effect is of importance in the energy range above the photoelectric effect and is the dominant absorption process for intermediate gamma energies (1 to 5 Mev for lead, 0.1 to 5 Mev for aluminum). Pair production occurs at higher energies with a threshold of 1.02 Mev, which is the energy equivalent of the mass of the positron and electron pair at rest.

For a narrow, collimated beam falling on an absorber, the gamma radiation is absorbed according to

$${}_γD = {}_γD_0 \exp(-\mu x) \quad (7.32)$$

where ${}_γD$ is the gamma-ray dose (rad), ${}_γD_0$ is the gamma-ray dose (rad) that would be received without a shield of x thickness, and μ is the total gamma-ray absorption coefficient ($1/\text{m}$ or cm^2/g). The absorption coefficient, μ , is a function of the specific shield material and energy of the radiation. A shield of thickness $x = 1/\mu$ would attenuate the beam by a factor of $1/e$; this thickness is designated as the relaxation length. The product μx is frequently used as a dimensionless thickness of absorber. The units of μ may be reciprocal length (linear absorption coefficient) or reciprocal mass per unit area (mass absorption coefficient).

In the Compton scattering process, which predominates in the energy ranges and absorber atomic numbers of interest in calculations of cloud gamma dose, only a portion of the energy is absorbed and transferred to a short-range electron. The remainder is carried by scattered photons that, although removed from the beam under the conditions for which Eq. 7.32 is written, can contribute to the gamma dose received at some other point. The total cross section for this process is divided into two parts: μ_a represents the fraction of the total energy in the recoil electrons and thus represents the fraction of the energy absorbed near the site of the interaction and μ_s represents the fraction of the energy contained in the scattered photons. Thus $\mu = \mu_a + \mu_s$.

Contributions from scattered radiation in air or other materials not directly between the source and the receptor will cause the gamma dose calculated from Eq. 7.32 to be low for broad beams. Buildup factors to account for the increased radiation have been computed for several configurations and shielding materials (Goldstein and Wilkins, 1954). For air, Gamertsfelder (1960) has used a buildup factor of the form $(1 + k\mu x)$, where k is equal to $(\mu - \mu_a)/\mu_a$. This factor gives the correct value of energy absorption in an infinite radioactive medium but overestimates the scatter component from sources close to the receptor and underestimates the scatter from distant sources.

The gamma dose rate to air at a distance r meters from a point source of q (curies) can then be written as

$$\gamma D'_r = \frac{\mu_a q (3.7 \times 10^{10}) \bar{E}_\gamma (1.6 \times 10^{-6}) (1 + k\mu r) \exp(-\mu r)}{4\pi r^2 (1293) (100)} \quad (7.33)$$

$$\gamma D'_r = 0.0364 \frac{\mu_a q \bar{E}_\gamma (1 + k\mu r) \exp(-\mu r)}{r^2} \quad (7.33a)$$

where $\gamma D'_r$ = dose rate (rad/sec) at a distance r (m)

μ_a = energy absorption coefficient for air (1/m)

μ = total absorption coefficient for air (1/m)

$k = (\mu - \mu_a)/\mu_a$

$\bar{E}_\gamma$ = average gamma energy emitted at each disintegration (Mev/dis)

3.7×10^{10} = disintegrations per second per curie

1.6×10^{-6} = ergs per Mev

1293 = density of air at S.T.P. (g/m³)

100 = ergs per gram per rad

The gamma dose rate to tissue would be higher than that to air by the ratio of the electron densities in the two media. Over the energy range from 0.1 to 5.0 Mev, this ratio can be approximated as 1.11. The gamma dose rate to tissue is given by

$$\gamma D'_t = 0.0404 \frac{\mu_a q \bar{E}_\gamma (1 + k\mu r) \exp(-\mu r)}{r^2} \quad (7.34)$$

Values of the gamma absorption coefficients for air and the associated values of k are given in Fig. 7.8 as a function of gamma energy. If air density differs from the standard conditions of temperature and pressure used in developing the relationship, corrections must be made in applying these coefficients. The ratio (ρ_0/ρ) should be used as a correction factor for values of μ and μ_a .

7.5.2 Cloud Gamma Dose

For clouds with dimensions that are small compared to the range of the gamma radiation, a calculation of the gamma dose at a given point must take into account the radiation received from various parts of the cloud. The complexities of geometry, absorption, and buildup make a complete solution difficult and require the use of special integrals. Approximate solutions have been obtained by neglecting absorption or buildup. Holland (1956) has calculated the dose, assuming that all particles of the cloud pass the receptor with the mean wind velocity and isotropic turbulence so that $\sigma_y = \sigma_z$, where σ_y represents the standard deviation of width and σ_z the standard deviation of height for the Gaussian-shaped cloud. His calculations have been presented in the form of nomograms. Couchman (1960) has described a computer technique based on the Sutton form of the diffusion equations which provides for attenuation, buildup, and nonisotropic diffusion. Gamertsfelder (1960) has provided a general method for an isotropic cloud which will be used in the following discussion.

7.5.2.1 Infinite Cloud. If the dimensions of a homogeneous cloud of gamma-emitting material are large compared to the distance that the

Fig. 7.8 — Absorption coefficients and values of the buildup constant for air at S.T.P.

gamma rays travel, an equilibrium condition occurs similar to that noted for beta radiation in Sec. 7-4.1.1. Under these conditions the rate of energy absorption per unit volume is equal to the rate of energy release per unit volume, and the gamma dose rate to air will be given by Eq. 7.17 if $\bar{E}_\beta$ is replaced by $\bar{E}_\gamma$, the average gamma energy released per disintegration, and a correction factor is included for the ratio of electron densities of tissue to those for air (a factor of 1.11).

$$\gamma D'_0 = 0.507 \bar{E}_\gamma \chi \left(\frac{\rho_0}{\rho} \right) \quad (7.35)$$

where the dose rate is in rads per sec and (ρ_0/ρ) is the correction factor that can be applied when the air is at other than standard conditions. This factor will generally not be repeated in subsequent equations except in instances where it is required for clarity.

If the receptor is at ground level, the gamma dose calculated with Eq. 7.35 must be corrected for the fact that the radiation is received from only half the cloud owing to the presence of the ground (a factor of 0.5). For an actual cloud the assumption of an infinite cloud of gamma-emitting material can be made only after relatively long travel times when the di-

mensions of the cloud (σ) are large compared to $1/\mu$. When this is the case, the value of the gamma dose rate is given by

$$\gamma D'_{\infty(x,y,0,t)} = 0.25 \bar{E}_{\gamma} \chi_{(x,y,0,t)} \quad (7.35a)$$

where $\gamma D'_{\infty(x,y,0,t)}$ is the gamma dose rate (rad/sec) to a person located at $(x,y,0)$ at time t and $\sigma > 1/\mu$.

Equation 7.35a will yield the gamma dose rate for an established plume from the emission of gamma-emitting material at a constant rate of Q'_0 (curies/sec) if the plume concentration $\bar{\chi}_{(x,y,0)}$ (curies/m³) is used. If the concentration time integral $\psi_{(x,y,0)}$ (curie sec/m³) is used, the dose received during the cloud passage is given by

$$\gamma D_{\infty(x,y,0)} = 0.25 \bar{E}_{\gamma} \psi_{(x,y,0)} \quad (7.36)$$

Consistent with earlier definitions, the value of $\bar{\chi}$ and ψ must include the appropriate correction factors for depletion en route, i.e., $[(Q_x/Q_0)_d (Q_x/Q_0)_R]$.

7.5.2.2 Finite Cloud. Near the source of the plume, the gamma doses and dose rates will be less than those estimated by the infinite-cloud assumption. The following treatment of the smaller cloud is from unpublished work by Gamertsfelder (1960).

In Fig. 7.9 consider the gamma dose rate to a ground-level receptor at point $(x_1, y_1, 0)$ and at time t if the gamma dose rate is due to the presence of q curies in a volume element of a

cloud containing a total of Q curies and if the center of the cloud is moving with a velocity u (average velocity $\bar{u}$) in the $x-z$ plane and is located at the point $(\bar{u}t, 0, h)$ at the time t . For an isotropic cloud consisting of a single puff release, values of σ_{xI} , σ_{yI} , σ_{zI} , and σ_I are equal, i.e., $\sigma_{xI} = \sigma_{yI} = \sigma_{zI} = \sigma_I$. If perfect reflection at the ground plane is assumed, the gamma dose rate to a receptor located at $(x_1, y_1, 0)$ from that part of the cloud which is reflected from the ground is the same as that which would be received if the cloud passed through the ground plane. Since the cloud concentration is symmetrical with respect to the line from $(x_1, y_1, 0)$ to the cloud center at $(\bar{u}t, 0, h)$, the dose rate from the entire cloud at time t is obtained by replacing the incremental source term, q , in Eq. 7.34 with the instantaneous cloud concentration times the elemental volume $\chi_{(x,y,z)} dV$ (which is equivalent) and integrating the volume element $2\pi r \sin \phi d\phi r dr$ over the entire cloud volume. The gamma dose rate to the receptor from the cloud at time t is given by

$$\begin{aligned} \gamma D'_{(x_1, y_1, 0, t)} &= 0.0404 \mu_a \chi_{(x,y,z,t)} \bar{E}_{\gamma} \\ &\times \int_0^{\infty} \int_0^{\pi} \frac{(1 + k\mu r) \exp(-\mu r)}{r^2} \\ &\times 2\pi r^2 \sin \phi d\phi dr \end{aligned} \quad (7.37)$$

In terms of the reference system given in Fig. 7.9, the general equation for the instantaneous isotropic cloud concentration, $\chi_{(x,y,z,t)}$ (curies/m³) can be written

$$\chi_{(x,y,z,t)} = \frac{Q_0(Q_x/Q_0)}{(2\pi)^{3/2} \sigma_I^3} \exp \left[-\frac{(m^2 + r^2 - 2mr \cos \phi)}{2\sigma_I^2} \right] \quad (7.38)$$

Fig. 7.9—Coordinate system for cloud gamma-dose calculations.

where Q_0 is the total number of curies of the radionuclide in the cloud at the time of release and (Q_x/Q_0) is the fraction of original source material which has not been deposited on the ground or removed by radioactive decay en route.

Replacing $\chi_{(x,y,z,t)}$ in Eq. 7.37 by the equivalent in Eq. 7.38 gives

$$\begin{aligned} \gamma D'_{(x,y,z,t)} = & \frac{0.0404 \mu_a \bar{E}_\gamma Q_0 (Q_x/Q_0)}{(2\pi)^{3/2} \sigma_1^3} \\ & \times \int_0^\infty \int_0^\pi (1 + k\mu r) \exp(-\mu r) \\ & \times \exp \left[-\frac{(m^2 + r^2 - 2 \cos \phi)}{2\sigma_1^2} \right] \\ & \times 2\pi r^2 \sin \phi \, d\phi \, dr \end{aligned} \quad (7.39)$$

$$\begin{aligned} \gamma D'_{(x,y,z,t)} = & \frac{0.0404 \mu_a \bar{E}_\gamma Q_0 (Q_x/Q_0)}{(2\pi)^{3/2} \sigma_1} \int_0^\infty (1 + k\mu r) \exp(-\mu r) \\ & \times \left\{ \frac{\exp \left[-\frac{(m-r)^2}{2\sigma_1^2} \right] - \exp \left[-\frac{(m+r)^2}{2\sigma_1^2} \right]}{mr} \right\} dr \end{aligned} \quad (7.40)$$

If σ_1 and the cloud source-depletion correction factors for decay and ground deposition are considered to be constants when the center of the cloud is at a distance $\bar{u}t$ from the point of release for the effective time of cloud passage, the equation for the gamma dose may be written

$$\begin{aligned} \gamma D_{(x,y,0)} = & \frac{0.0404 \mu_a \bar{E}_\gamma Q_0 (Q_x/Q_0)}{2\pi \sigma_1} \int_0^\infty \int_0^\infty (1 + k\mu r) \exp(-\mu r) \\ & \times \frac{\exp \left[-\frac{(m-r)^2}{2\sigma_1^2} \right] - \exp \left[-\frac{(m+r)^2}{2\sigma_1^2} \right]}{mr} \, dr \, dt \end{aligned} \quad (7.41)$$

If I_1 and I_2 are defined as

$$\begin{aligned} I_1 = & \frac{\bar{u}}{4(2\pi)^{3/2} \mu \sigma_1} \int_0^\infty \int_0^\infty \frac{\exp(-\mu r)}{mr} \\ & \times \left\{ \exp \left[-\frac{(m-r)^2}{2\sigma_1^2} \right] \right. \\ & \left. - \exp \left[-\frac{(m+r)^2}{2\sigma_1^2} \right] \right\} \, dr \, dt \end{aligned} \quad (7.41a)$$

and

$$\begin{aligned} I_2 = & \frac{\bar{u}}{4(2\pi)^{3/2} \mu \sigma_1} \int_0^\infty \int_0^\infty \frac{\mu \exp(-\mu r)}{m} \\ & \times \left\{ \exp \left[-\frac{(m-r)^2}{2\sigma_1^2} \right] \right. \\ & \left. - \exp \left[-\frac{(m+r)^2}{2\sigma_1^2} \right] \right\} \, dr \, dt \end{aligned} \quad (7.41b)$$

then Eq. 7.41 may be written as

$$\gamma D_{(x,y,0)} = \frac{0.1616 \mu \mu_a \bar{E}_\gamma Q_0 (Q_x/Q_0) (I_1 + kI_2)}{\bar{u}} \quad (7.42)$$

Values of I_1 and I_2 were determined by graphical and numerical integrations. Specific values of I_1 and I_2 can be obtained from Figs. 7.10 and 7.11 for various values of $\mu \sigma_1$. In these figures values of I_1 and I_2 are presented as a function of $\mu \sigma_1$ and are given for various values of μs , where $\mu s = \mu(y^2 + h^2)^{1/2}$ and s is the distance between the receptor and the center line of the cloud path. Dashed lines show values of I_1 and I_2 as functions of fixed ratios between s and σ_1 and thereby permit interpolation of I_1 and I_2 for values of μs other than those given.

The values of I_1 and I_2 can be applied to the case of the continuous source if the instantaneous values of cloud width (σ_1) are replaced by values for the continuous plume (σ).

Equation 7.36 was developed for an isotropic single-puff release of Q_0 curies of radioactive material. If the release were to occur at a constant rate (Q_0) over a period of time (t_d) so that $Q_0 t_d = Q_0$, then the governing meteorological variables to determine the concentration time integral, ψ , would be $\bar{u}$, σ_y , and σ_z . These sigma values would be larger than the instantaneous puff values, σ_1 , because they would include the effects of meander of the plume at the receptor over the time interval of the release. Since the equation for the concentration time integral for the plume is identical in form to that for the puff release (see Chap. 3, Eq. 3.158), the only difference in the computed values of ψ would be due to the differences in the wind speed and the plume spread during the long and short time intervals. In both cases then the dose is proportional to ψ . Values for the continuous and instantaneous sigmas can be found in Chap. 4, Secs. 4-4.1.7, 4-4.3, and 4-10.3. The sigma values can be adjusted to the actual release

Fig. 7.10 — Values of the I_1 integral (see Eq. 7.41a). Values of σ or σ_1 can be used.

Fig. 7.11—Values of the I_2 integral (see Eq. 7.41b). Values of σ or σ_1 can be used.

interval over the range of a few minutes to a few hours by a power law such as is given in Chap. 2, Eq. 2.7.

The gamma dose rate ($\gamma D'$), which is the time derivative of the dose (γD), will be proportional to the time derivative (Q'_0) of the source (Q_0) if the plume has attained an equilibrium growth condition. Therefore Eq. 7.42 can be used to calculate the gamma dose rate from an established plume if the source release rate (Q'_0) is used in place of the total source (Q_0) and if sigma values appropriate to the continuous source are used.

$$\gamma D'_{(x,y,0)} = \frac{0.1616 \mu \mu_a \bar{E}_\gamma Q'_0 (Q_x/Q_0) (I_1 + kI_2)}{\bar{u}} \quad (7.43)$$

where $\gamma D'_{(x,y,0)}$ is the gamma dose rate (rad/sec) from the established plume to a receptor on the ground plane at a point x (m) downwind and y (m) perpendicular to the plume center line in the horizontal plane and Q'_0 is the source release rate at the time of release (curie/sec).

It is frequently more convenient to consider the power generated by the radioactive materials in the cloud at the point of interest. In this case the product $\bar{E}_\gamma Q_0$ can be replaced by the gamma power, P_γ , if suitable absorption coefficients can be assumed. Conversion of the constants in Eq. 7.42 gives

$$\gamma D_{(x,y,0)} = \frac{4.36 \times 10^{-12} \mu \mu_a P_{\gamma 1}}{\bar{u}} (I_1 + kI_2) \quad (7.44)$$

where $P_{\gamma 1}$ is the total energy release rate in the cloud by gamma emission at the time of cloud passage (Mev/sec), i.e., it is corrected for depletion en route, and γD is the gamma dose in rads. If $P_{\gamma 2}$ is in watts, Eq. 7.44 becomes

$$\gamma D_{(x,y,0)} = 27.3 P_{\gamma 2} \frac{\mu \mu_a}{\bar{u}} (I_1 + kI_2) \quad (7.45)$$

and the same depletion correction factors used before are applicable.

Values of $(I_1 + kI_2)$ can be computed as a single term, I_T , if the gamma energy and thus the absorption coefficients can be fixed. If it is assumed that the effective gamma energy in a cloud consisting of fission products is 0.7 Mev, then $\mu = 9.7 \times 10^{-3}$ and $\mu_a = 3.8 \times 10^{-3}$. The

values of $I_T = (I_1 + kI_2)$ for this condition are given in Fig. 7.12.

The preceding equations apply to an isotropic cloud, i.e., $\sigma_y = \sigma_x$ or $\sigma_y = \sigma_z$. Since many of the present studies of diffusion result in a non-isotropic-cloud formulation, particularly with inversions, a method of calculation is required. Gamertsfelder (1960) noted that if $\sigma_y > \sigma_z$ the value of I_T for $\sigma_y > \sigma_z$ and σ_z small with respect to the height of the cloud center line is the same as if $\sigma_y = \sigma_z$. For a cloud which is effectively infinite in height ($\sigma_z \approx 700$ m), however, the value of I_T should be reduced by the fraction σ_z/σ_y to account for the reduced concentration caused by the horizontal spread. Gamertsfelder suggested that a correction factor at intermediate distances can be obtained by drawing a curve with correction factor on the abscissa and σ_z on the ordinate scales. The value of the correction factor should be unity for 0.7-Mev gamma rays at $\sigma_z \leq 0.035h$, and the curve should smoothly approach the value of σ_z/σ_y at $\sigma_z = 700$ m. This curve is shown in Fig. 7.13.

An alternative approach to the problem of correcting for a nonisotropic cloud which gives comparable answers is to use an average value of σ given by

$$\sigma_{av.} = (\sigma_y \sigma_z)^{1/2} \quad (7.46)$$

7.5.2.3 Ratio of Finite to Infinite Cloud Dose. Since it is relatively simple to estimate the gamma dose or dose rate from a cloud large enough to be considered infinite, it is frequently more convenient to apply correction factors to the dose calculations for infinite clouds to determine the dose from a cloud of finite size. The concentration time integral at a point on the path of the center of a cloud of finite size is given by

$$\psi_{(x,0,h)} = \frac{Q_x}{\pi \bar{u} \sigma_y \sigma_z} \quad (7.47)$$

If the concentration of an infinite cloud is numerically equal to the concentration time integral of the finite cloud, i.e., $\chi_\infty = \psi$, and the exposure time to the infinite cloud is 1 sec, the ratio of the gamma doses from the two clouds can be calculated by using Eqs. 7.36, 7.42, and 7.46.

Fig. 7.13—Correction curves for gamma dose from nonisotropic clouds.

Fig. 7.14—Ratio of the gamma dose in a finite cloud to the gamma dose in an infinite cloud having the same concentration as the center line of the finite cloud for 0.7-Mev gamma photons.

Fig. 7.15—Ratio of off-center-line to center-line doses from finite size clouds containing radionuclides emitting 0.7-Mev gammas compared to Gaussian-concentration distribution.

three different gamma energies in Fig. 7.16. The spread in ratios is only a factor of 2 for energies that differ by a factor of 20.

7-5.2.4 Gamma Dose from Releases Under Capping Inversions. It is not uncommon to find that the vertical diffusion of the cloud released in a layer between the ground and an inversion base is restricted by the presence of the two boundaries. Just as the concentration is affected by the presence of the ground (the reflected component), it is also affected by the presence of the inversion layer. Capping inversions have been discussed in Chap. 3, Sec. 3-3.5, and values of the heights that might be expected have been presented by Holzworth (1964). Mounting evidence indicates that capping inversions occur

frequently over cities at night, as well as over other rather limited geographical areas.

The concentration and concentration time integral of a cloud released in the layer between the ground and a capping inversion will be greater than similar values calculated with equations based upon the uniform infinite atmosphere. The magnitude of this difference will depend upon the relative magnitudes of the height of the inversion layer, the height of release of the source material, and the distance downwind from the release point. The following treatment of the problem, which is similar to that of Hewson, Gill, and Bierly (1959), makes use of the reference system shown in Fig. 7.17.

Consider a source Q_0 released at a point h meters above the ground and below the base of the inversion, h_i . As the cloud moves downwind, the source is depleted by ground deposition and radioactive decay, i.e., $Q_x = Q_0 (Q_x/Q_0)$. An observer located at the center of the cloud and moving with the cloud at the average wind speed $\bar{u}$ (m/sec) would observe that the upper and lower cloud limits follow the paths indicated by the solid and dashed lines of Fig. 7.17, respectively. The path lengths computed with the reference points $(a, a', b, b', \dots)$ are indicated in Table 7.7.

If an arbitrary index $m = \theta$ is used for the first two ground reflections, i.e., a, a', b, b' , and

Table 7.7—HYPOTHETICAL CLOUD OBSERVATIONS UNDER CAPPING INVERSIONS

Reference points	Direction	Distance
Upper Cloud Limit		
b	↓	$(2h_i - h - \Delta h)$
b'	↑	$(2h_i - h + \Delta h)$
d	↓	$(2h_i - h - \Delta h) + 2h_i$
.	.	.
.	.	.
n	↓	$(2h_i - h - \Delta h) + (a - 1)2h_i$
n'	↑	$(2h_i - h + \Delta h) (a + 1)2h_i$
Lower Cloud Limit		
a	↓	$(h - \Delta h)$
a'	↑	$(h + \Delta h)$
c	↓	$(h - \Delta h) + 2h_i$
.	.	.
.	.	.
m	↓	$(h - \Delta h) + a2h_i$
m'	↑	$(h + \Delta h) + a2h_i$

Fig. 7.16—Ratio of gamma dose at center line in a finite cloud of gamma-emitting radionuclides to dose in an infinite cloud with the same center-line concentrations.

Fig. 7.17—Schematic drawing of the reference system used in the calculational model for the cloud concentration under a capping inversion.

$m = 1$ for the next two reflections, etc., the path lengths can be described in general terms:

Reference points	Distances
Lower Cloud	
(a, c, e, ... m)	$2mh_i + h - \Delta h$
(a', c', e', ... m')	$2mh_i + h + \Delta h$
Upper Cloud	
(b, d, f, ... n)	$(2m + 1)h_i - h - \Delta h$
(b', d', f', ... n')	$(2m + 1)h_i - h + \Delta h$

The concentration time integral at a point r meters from the center of an isotropic cloud can be described by

$$\psi_r = \psi_p \exp\left(-\frac{r^2}{2\sigma^2}\right) \quad (7.49)$$

where ψ_p is the concentration time integral of the cloud center line (curie sec/m³).

With the path lengths developed for the capping inversion, the concentration time integral

at a point x m downwind directly under the cloud path and Δh meters above the ground may be described by

$$\begin{aligned} \psi_{(x,0,\Delta h)} = \psi_p \sum_{m=0}^{\infty} \left\{ \exp - \left[\frac{2mh_i + h - \Delta h}{(2)^{1/2}\sigma_z} \right]^2 \right. \\ + \exp - \left[\frac{2mh_i + h + \Delta h}{(2)^{1/2}\sigma_z} \right]^2 \\ + \exp - \left[\frac{(2m+1)h_i - h - \Delta h}{(2)^{1/2}\sigma_z} \right]^2 \\ \left. + \exp - \left[\frac{(2m+1)h_i - h + \Delta h}{(2)^{1/2}\sigma_z} \right]^2 \right\} \quad (7.50) \end{aligned}$$

For the special case of the receptor at ground level, $\Delta h = 0$, the first and second exponential terms are equal and the third and fourth terms are equal; therefore Eq. 7.50 becomes

$$\begin{aligned} \psi_{(x,0,0)} = 2\psi_p \sum_{m=0}^{\infty} \left\{ \exp - \left[\frac{2mh_i + h}{(2)^{1/2}\sigma_z} \right]^2 \right. \\ \left. + \exp - \left[\frac{(2m+1)h_i - h}{(2)^{1/2}\sigma_z} \right]^2 \right\} \quad (7.50a) \end{aligned}$$

For the special case of the inversion layer at a very-high altitude, only the ground reflection is important, and Eq. 7.50 becomes

$$\psi_{(x,0,0)} = 2\psi_p \exp \left(-\frac{h^2}{2\sigma_z^2} \right) \quad (7.50b)$$

The terms $f(\sigma_z, h, h_i)$, f_1 , and f_2 are defined as

$$\begin{aligned} f(\sigma_z, h, h_i) = \sum_{m=0}^{\infty} \left\{ \exp - \left[\frac{2mh_i + h}{(2)^{1/2}\sigma_z} \right]^2 \right. \\ \left. + \exp - \left[\frac{2(m+1)h_i - h}{(2)^{1/2}\sigma_z} \right]^2 \right\} \quad (7.51) \end{aligned}$$

and

$$\begin{aligned} f_1(\sigma_z, h, h_i) &= \frac{h}{(2)^{1/2}\sigma_z} f(\sigma_z, h, h_i) \\ f_2(\sigma_z, h, h_i) &= \frac{h^2}{2\sigma_z^2} f(\sigma_z, h, h_i) \quad (7.52) \end{aligned}$$

When $(2)^{1/2}\sigma_z > h_i$, where the vertical equilibrium has been established by multiple reflections of the cloud between the two planes, Eq. 7.52 becomes

$$f_1(\sigma_z, h, h_i) = \frac{(\pi)^{1/2}h}{2h_i}$$

and

$$f_2(\sigma_z, h, h_i) = \frac{(\pi)^{1/2}h^2}{(2)^{1/2}\sigma_z h_i} \quad (7.52a)$$

When $(2)^{1/2}\sigma_z < h_i$, Eq. 7.52 becomes

$$f_1(\sigma_z, h, h_i) = \frac{h \exp(-h^2/2\sigma_z^2)}{(2)^{1/2}\sigma_z}$$

and

$$f_2(\sigma_z, h, h_i) = \frac{h^2 \exp(-h^2/2\sigma_z^2)}{2\sigma_z^2} \quad (7.52b)$$

With these functions the equations for the cloud concentrations become the following:

For an instantaneous point source released at h ,

$$\begin{aligned} \chi_{(x,y,0,t)} &= \frac{Q_x \sigma_{zi}}{\pi^{1/2} (2)^{1/2} \sigma_{xi} \sigma_{yi} h^2} f_2(\sigma_{zi}, h, h_i) \\ &\times \exp \left\{ - \left[\frac{(x - \bar{u}t)^2}{2\sigma_{xi}^2} + \frac{y^2}{2\sigma_{yi}^2} \right] \right\} \quad (7.53) \end{aligned}$$

and, when $(2)^{1/2}\sigma_{zi} > h_i$,

$$\begin{aligned} \chi_{(x,y,0,t)} &= \frac{Q_x}{2\pi \sigma_{xi} \sigma_{yi} h_i} \\ &\times \exp \left\{ - \left[\frac{(x - \bar{u}t)^2}{2\sigma_{xi}^2} + \frac{y^2}{2\sigma_{yi}^2} \right] \right\} \quad (7.53a) \end{aligned}$$

For a continuous point source with steady winds released at h ,

$$\begin{aligned} \bar{\chi}_{(x,y,0)} &= \frac{2Q'_x f_1(\sigma_z, h, h_i)}{\pi \bar{u} h (2)^{1/2} \sigma_y} \exp \left(-\frac{y^2}{2\sigma_y^2} \right) \\ &= \frac{2Q'_x \sigma_z f_2(\sigma_z, h, h_i)}{\pi \bar{u} h^2 \sigma_y} \exp \left(-\frac{y^2}{2\sigma_y^2} \right) \quad (7.54) \end{aligned}$$

and, when $(2)^{1/2}\sigma_z > h_i$,

$$\bar{\chi}_{(x,y,0)} = \frac{Q'_x}{(2\pi)^{1/2} \sigma_y \bar{u} h_i} \exp \left(-\frac{y^2}{2\sigma_y^2} \right) \quad (7.54a)$$

Note that Eqs. 7.54 and 7.54a yield the concentration time integral when Q_x is substituted for Q'_x and appropriate values of σ are used.

For a continuous source release over long time periods (the equations for concentration and dosage from sources operating over long time periods are developed in Sec. 7-5.2.5.),

$$\bar{X}_{(R,\theta,0)} = \frac{2Q'_R f_1(\sigma_z, h, h_1)}{(\pi)^{1/2} \bar{u} R \theta h} \quad (7.55)$$

and, when $(2)^{1/2} \sigma_z > h_1$,

$$\bar{X}_{(R,\theta,0)} = \frac{Q_R}{\bar{u} R \theta h_1} \quad (7.55a)$$

These equations can be used to determine the amount of radioactive material inhaled during cloud passage and to calculate the direct dose to persons from the cloud passage. Values of $f_1(\sigma_z, h, h_1)$ and $f_2(\sigma_z, h, h_1)$ can be obtained from Figs. 7.18 and 7.19, where they are shown

graphically as a function of σ_z/h for various ratios of h_1/h .

The equations previously developed for the direct dose from passing clouds can be written in terms of $f_1(\sigma_z, h, h_1)$ and $f_2(\sigma_z, h, h_1)$.

For the gamma dose from passage of a finite size cloud (see Eq. 7.42) where $(2)^{1/2} \mu \sigma_z > 10$, the values of I_1 and I_2 can be given accurately as

$$I_1 = I_2 = \frac{\exp(-s/2\sigma_1^2)}{2\mu\sigma_1^2} \quad (7.56)$$

and the equation for the gamma dose can be written

$$\gamma D_{(x,y,0)} = \frac{0.1616 Q_0 (Q_x/Q_0) \exp(-s^2/2\sigma_1^2)}{\bar{u} 2\sigma_1^2} \quad (7.57)$$

Fig. 7.18—Values of $f_1(\sigma_z, h, h_1)$ as a function of σ_z/h for various ratios of h_1/h . (See Eq. 7.52).

Fig. 7.19—Values of $f_2(\sigma_z, h, h_i)$ as a function of σ_z/h for various ratios of h_i/h . (See Eq. 7.52).

where values of μ_a and μ are not involved.

In terms of the functions previously derived, this can be written

$$\gamma D_{(x,y,0)} = \frac{0.1616 Q_x \bar{E}_\gamma \exp(-y^2/2\sigma_y^2)}{\bar{u} h^2} \times \frac{\sigma_z}{\sigma_y} f_2(\sigma_z, h, h_i) \quad (7.58)$$

In a similar manner the gamma dose from a prolonged release of Q'_d (curies/sec) for t_d (sec) to a receptor in the sector wherein the release may be averaged over the sector width θ (for a fuller discussion of this, see Eqs. 7.60, 7.61, and 7.62) can be written

$$\gamma D_{(R,\theta,0)} = \frac{0.2865 Q'_d t_d \bar{E}_\gamma}{\bar{u} R \theta h} f_1(\sigma_z, h, h_i) \quad (7.59)$$

This equation for a capping inversion yields results that are slightly high for cases where $\mu h_i < 2$, but, if $\mu h_i > 2$, the results are high by only a few percent and no correction is necessary.

7-5.2.5 Average Gamma Dose from Continuous Releases over Long Time Periods. During routine release of radioactive effluents from stacks, the gamma dose received in the vicinity over a long period of time is of interest for comparison with the radiation limits, which frequently permit averaging of the dose over time intervals of up to 1 year. Estimates of this dose prior to carrying out a disposal program permit the establishment of emission limits, which, in turn, indicate the plant features needed to meet these limits.

The average dose over such a long time period must consider the effects of shifting winds, as well as the more normal parameters for estimating turbulent diffusion. The frequency with which the wind blows in a given direction is given by the conventional wind rose, usually to an 8- or 16-point compass or to the nearest 10°. For an estimate of long-term average doses, the frequency of occurrence of each atmospheric stability class must be provided to augment the wind rose.

If it is assumed that a source of radioactive material is continuously emitted at a constant rate of Q'_0 (curie/sec) at a height h (m) above the ground plane for a time period t_d (sec), the equilibrium concentration in the plume at a point (x, y, z) is given by Eq. 3.115 in Chap. 3.

$$\bar{X}_{(x,y,z)} = \frac{Q'_x}{2\pi\bar{u}\sigma_y\sigma_z} \exp\left(-\frac{y^2}{2\sigma_y^2}\right) \times \left\{ \exp\left[-\frac{(z-h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(z+h)^2}{2\sigma_z^2}\right] \right\}$$

where $Q'_x = Q'_0 (Q_x/Q_0)$ is the activity release rate for a specific radionuclide corrected for depletion en route to the receptor and the coordinate system is the same as that used previously. The last exponential term in the equation represents the reflected component, i.e., that portion of the plume which has diffused to

the ground plane and diffused back into the plume. For calculations of the direct gamma dose rate from the plume, the term for the reflected component will be retained in this form which, essentially, disregards the presence of the ground plane as a barrier but correctly describes the location of the reflected component source contribution with respect to a ground-level receptor.

The following treatment of the problem is based upon the work of Gamertsfelder (1960). Figure 7.20 shows the reference system used in the following discussion. If the direction of the emission shifts continuously and uniformly over a horizontal angle θ (radians) during the time period of concern, the concentration within the sector can be averaged in the horizontal direction across the sector. Integrating Eq. 3.115 in the y -direction from $-\infty$ to $+\infty$ and averaging the concentration over the width of the sector with R as the radial equivalent of x yields

$$\bar{X}_{(R,\theta,0)} = \frac{Q'_R}{(2\pi)^{1/2}\sigma_z\bar{u}R\theta} \left\{ \exp\left[-\frac{(z-h)^2}{2\sigma_z^2}\right] + \exp\left[-\frac{(z+h)^2}{2\sigma_z^2}\right] \right\} \quad (7.60)$$

where $\bar{X}_{(R,\theta,0)}$ is the average concentration in the plume at a point R meters radially from the re-

Fig. 7.20 — Reference system for long-term dosage equations.

lease point and z meters above the ground plane (curies/m³) and θ is the angular width of the sector within which the plume is dispersed (radians).

In cases where the distance between the source and the receptor is sufficiently great so that the change in concentration with distance is relatively small over the relaxation length of the gamma rays, the plume, averaged over the sector, can be considered to be comprised of a series of infinite horizontal plane sources. The plane sources vary in concentration according to a Gaussian distribution centered about the plume center plane, i.e., centered about the height of release.

The gamma dose rate to tissue located a distance r (m) from a source increment of q curies was given by Eq. 7.34.

If a volume element of the source is located at a height z (m) above the ground plane and a horizontal distance L (m) from the receptor, the source is a distance $(L^2 + z^2)^{1/2}$ from the receptor. With this reference system shown in Fig. 7.20, a volume element of source in the plane is $2\pi L dL dz$. Substituting $(L^2 + z^2)^{1/2}$ for r in Eq. 7.34 and the expression for $\bar{X}_{(R,\theta,0)}$ of Eq. 7.60 for the incremental source q , the equation for the gamma dose rate from the plume can be written as

$$\begin{aligned} \gamma D'_{(R,\theta,0)} &= \frac{0.0404 \mu_a \bar{E}_\gamma Q'_x}{(2\pi)^{1/2} \bar{u} R \sigma_z \theta} \int_0^\infty \int_0^\infty \frac{1}{L^2 + z^2} \\ &\times \left[\left[1 + k\mu (L^2 + z^2)^{1/2} \right] \left\{ \exp \left[-\frac{(z-h)^2}{2\sigma_z^2} \right] \right. \right. \\ &\left. \left. + \exp \left[-\frac{(z+h)^2}{2\sigma_z^2} \right] \right\} \exp \left[-\mu (L^2 + z^2)^{1/2} \right] \right] \\ &\times 2\pi L dL dz \quad (7.61) \end{aligned}$$

If I_1 and I_2 are defined as

$$\begin{aligned} \bar{I}_1 &= \frac{1}{(2)^{1/2} \sigma_z} \int_0^\infty \int_0^\infty \frac{1}{L^2 + z^2} \left\{ \exp \left[-\frac{(z-h)^2}{2\sigma_z^2} \right] \right. \\ &\left. + \exp \left[-\frac{(z+h)^2}{2\sigma_z^2} \right] \right\} \exp \left[-\mu (L^2 + z^2)^{1/2} \right] L dL dz \\ &= \frac{1}{(2)^{1/2} \sigma_z} \int_0^\infty \left\{ \exp \left[-\frac{(z-h)^2}{2\sigma_z^2} \right] + \exp \left[-\frac{(z+h)^2}{2\sigma_z^2} \right] \right\} \\ &\times \left\{ \int_{\mu z}^\infty \frac{\exp \left[-\mu (L^2 + z^2)^{1/2} \right]}{\mu (L^2 + z^2)} d[\mu (L^2 + z^2)^{1/2}] \right\} dz \quad (7.61a) \end{aligned}$$

$$\begin{aligned} \bar{I}_2 &= \frac{1}{(2)^{1/2} \sigma_z} \int_0^\infty \int_0^\infty \frac{1}{(L^2 + z^2)^{1/2}} \left\{ \exp \left[-\frac{(z-h)^2}{2\sigma_z^2} \right] \right. \\ &\left. + \exp \left[-\frac{(z+h)^2}{2\sigma_z^2} \right] \right\} \exp \left[-\mu (L^2 + z^2)^{1/2} \right] \mu L dL dz \\ &= \frac{1}{(2)^{1/2} \sigma_z} \int_0^\infty \left\{ \exp \left[-\frac{(z-h)^2}{2\sigma_z^2} \right] \right. \\ &\left. + \exp \left[-\frac{(z+h)^2}{2\sigma_z^2} \right] \right\} \exp (-\mu z) dz \quad (7.61b) \end{aligned}$$

Equation 7.61 can be written

$$\gamma D'_{(R,\theta,0)} = \frac{0.2865 \mu_a Q'_x \bar{E}_\gamma}{\bar{u} R \theta} [\bar{I}_1 + k\bar{I}_2] \quad (7.62)$$

Values of $\bar{I}_1$ and $\bar{I}_2$ are shown graphically in Figs. 7.21 and 7.22 as a function of $\mu\sigma_z$ for various release heights. The dashed lines permit interpolating for release heights not given specifically. Equation 7.62 and other equations derived from the model will yield values that somewhat overestimate the gamma dose rate (or dose) if $\mu r \theta < 2$ but yield quite accurate values when $\mu r \theta \geq 2$. The gamma dose rate can be multiplied by the time period of discharge, t_d (sec), to calculate the dose to the receptor. For a specific gamma energy, the term $[\bar{I}_1 + k\bar{I}_2]$ can be combined into a single term $\bar{I}_T$. Figure 7.23 presents the values of $\bar{I}_T$ for 0.7-Mev photons.

Equation 7.62 can be written as shown in Eq. 7.59 for the gamma dose from a prolonged release under a capping inversion. For a fuller discussion see Sec. 7-5.2.4 and Eq. 7.59.

Equation 7.62 can be used to calculate the long-term direct gamma dose to receptors in the general vicinity of a point where a source of radioactive material is released continuously at a uniform rate over an extended period of time. The direct gamma dose from such a source to an individual located at $(R, \theta_1, 0)$, a distance R (m) from the point of release in a sector of width θ (radians) at ground level, can be calculated by evaluating Eq. 7.63.

$$\begin{aligned} \gamma D_{(R,\theta_1,0)} &= \frac{0.2865 \Delta t}{\theta R} \left[\sum_{j=1}^l \sum_{j=1}^m \sum_{k=1}^n \frac{P_{\sigma_{z(j,k,\theta_1)}} Q'_{R(i,j,k)}}{\bar{u}_j} \right. \\ &\left. \times \mu_{a_i} \bar{E}_{\gamma_i} (\bar{I}_1 + k\bar{I}_2)_{(i,j,k)} \right] \quad (7.63) \end{aligned}$$

Fig. 7.21 — Values of the $\bar{I}_1$ integral (see Eq. 7.61a).

Fig. 7.22 — Values of the $\bar{I}_2$ integral (see Eq. 7.61b).

where $Q_{R(i,j,k)}$ = source term (release rate) of isotope type i corrected for depletion by ground deposition and radioactive decay en route to receptor location

$$= Q_0' \left[\exp \left(-\lambda_i \frac{R}{\bar{u}_j} \right) \right] (Q_x/Q_0)_{d,j,k}$$

(curie/sec)

$(Q_R/Q_0)_{d(j,k)}$ = depletion factor for ground deposition

$$= \exp \left[- \left(\frac{2}{\pi} \right)^{1/2} \frac{v_d}{\bar{u}_j} \right] \times \int_0^R \frac{\exp \left(- \frac{h^2}{2\sigma_{zk}^2} \right)}{\sigma_{zk}} dR$$

or equivalent Sutton form

$(\bar{I}_1 + k\bar{I}_2)_{i,j,k}$ = function of μ_i , σ_{zk} , and $\bar{u}_j$

$\bar{u}_j$ = average of wind speed range j (m/sec)

σ_{zk} = standard deviation of plume concentration in vertical plane for stability class k (m)

$P_{\sigma_{z(j,k,\theta_1)}}$ = joint probability of wind speed and atmospheric stability into sector θ_1

Δt = exposure time interval (sec)

θ = angular width of sector θ_1 (radians)

R = distance from release point (m)

At long distances from the point of release, σ_z becomes large so that the cloud becomes uniform in concentration and can be regarded as an infinite cloud for which the gamma dose

Fig. 7.23 — Values of the $\bar{I}_T$ integral for 0.7-Mev gamma radiation.

rate is given by Eq. 7.35a with the assumption that the value of χ used includes the reflected component of the cloud.

A receptor located on the line that defines the division between sectors would receive an intermediate dose between those doses calculated for the two sectors. In practice the doses for each of the sectors are calculated at several radial distances over the range of interest. The results are plotted on log graph paper (dose vs. distance); the radial distances in each sector are determined for several selected dose values (usually differing by factors of 10^1 or $10^{0.5}$), and these locations are plotted on circular coordinate paper for the center-line rays in each sector. Points representing the location of common dose values are then connected with a smooth curve to represent the dose isopleths.

When the many approximations in developing the mathematical models and difficulty in obtaining input data are considered, more exact methods are not warranted.

7-5.3 Gamma Dose from Radioactive Materials Deposited on the Ground

A passing cloud of radioactive particles or halogens will deposit material on the ground by dry deposition, washout, or rainout. The radioactive materials on the ground will result in a gamma-radiation dose for a period of time after the cloud has passed. The ground pattern of these materials can be estimated by the methods indicated in other chapters. The persistence of the dose depends upon the radio-

active decay of the materials involved and the degree of cleanup by natural forces or by decontamination efforts.

7-5.3.1 The Infinite Plane Source. When the contamination is uniformly spread over a wide area, the intensity of the radiation is given by Eq. 7.64 for various distances above the ground.

$$I = \frac{S}{2} E_i(\mu r) + \frac{kS \exp(-\mu r)}{2} \quad (7.64)$$

where I is the gamma-ray intensity (photon/m² sec), $E_i(\mu r)$ is the exponential integral $\int_{\mu r}^{\infty} [\exp(-\mu r)/r] dr$, and S is the source strength per unit area of the plane (photon/m² sec). The exponential integral, $E_i(\mu r)$, occurs in many shielding problems and is given in tabular form by Jahnke, Emde, and Lösch (1960). Equation 7.64 can be converted to dose rate units to give

$$\gamma D'_{\infty} = \frac{(I) (\bar{E}_{\gamma}) (\mu_a) (1.6 \times 10^{-6}) (1.11)}{(1293) (100)}$$

$$\gamma D'_{\infty} = 1.37 \times 10^{-11} I \bar{E}_{\gamma} \mu_a \quad (7.65)$$

If I is replaced by its equivalent as given in Eq. 7.64 and one photon of average energy $\bar{E}_{\gamma}$ is assumed to be emitted per disintegration, the gamma dose rate is given by

$$\gamma D'_{\infty} = 0.254 \omega \bar{E}_{\gamma} \mu_a \left[E_i(\mu b) + \frac{k \exp(-\mu b)}{2} \right] \quad (7.65a)$$

where $\gamma D'_{\infty}$ is the gamma dose rate (rad/sec) to a receptor located b meters above an infinite plane uniformly contaminated, and ω is the number of curies per square meter on the ground. The constant in the buildup factor has been divided by two to provide an approximate correction for the primary radiation that enters the ground and is scattered to the receptor with a much lower efficiency than that which enters the air.

Figure 7.24 presents the dose rate at several heights above the ground for radiation with energies from 0.1 to 2 Mev as obtained from Eq. 7.65.

Fig. 7.24—Gamma dose rate at heights of 1, 10, and 100 m above ground as given for infinite plane source of 1.0 curie/m².

7-5.3.2 The Finite Plane Source. If radioactive materials are deposited from a plume, either by dry deposition or by rainout or washout, the deposition pattern will consist of a maximum at the center line with the deposition decreasing with distance from the center line according to the Gaussian distribution of concentration in the cloud. This pattern can be described by

$$\omega_{(x,y,0,t)} = \omega_{(x,0,0,t)} \exp\left(-\frac{y^2}{2\sigma_y^2}\right) \quad (7.66)$$

where $\omega_{(x,0,0,t)}$ is the deposition in curies per square meter at distance x on the cloud center line at a specific time and $\omega_{(x,y,0,t)}$ is the deposition at a distance y from the center line at the same time. Since the value of ω will vary as a function of time owing to radioactive decay or decontamination efforts, the equation for ω must include these considerations.

Gamertsfelder (1960) has derived an equation to give the gamma dose rate to tissue at a height b above the center line of the pattern.

Fig. 7.25 — Reference system for equations involving deposition patterns of finite size.

From the equation for a point source, Eq. 7.34, and the coordinate system given in Fig. 7.25, the dose rate is given by

$$\begin{aligned} \gamma D_{(x_1, 0, b, t)} &= 0.0404 \mu_a \bar{E}_\gamma \omega_{(x_1, 0, 0, t)} \\ &\times \int_{-\infty}^{\infty} \int_0^{\infty} \frac{[1 + (k/2) \mu (l^2 + b^2)^{1/2}] \exp[-\mu(l^2 + b^2)^{1/2}]}{l^2 + b^2} \\ &\times \exp\left(-\frac{y^2}{2\sigma_y^2}\right) dx dy \end{aligned} \quad (7.67)$$

where $\gamma D_{(x_1, 0, b, t)}$ is in rads per second at time t and $k/2$ is used in the buildup factor to allow for loss of scattered radiation from that part of the primary radiation that starts in a downward direction. With G_1 and G_2 defined as

$$G_1 = \frac{1}{(2)^{1/2} \mu \sigma_y} \int_{-\infty}^{\infty} \int_0^{\infty} \frac{\exp[-\mu(l^2 + b^2)^{1/2}] \exp(-y^2/2\sigma_y^2)}{l^2 + b^2} \times dx dy \quad (7.68)$$

$$G_2 = \frac{1}{(2)^{1/2} \pi \mu \sigma_y} \int_{-\infty}^{\infty} \int_0^{\infty} \frac{\mu \exp[-\mu(l^2 + b^2)^{1/2}] \exp(-y^2/2\sigma_y^2)}{(l^2 + b^2)} \times dx dy \quad (7.68a)$$

Then

$$\begin{aligned} \gamma D_{(x_1, 0, b, t)} &= 0.36 \omega_{(x_1, 0, 0, t)} \\ &\times \sigma_y \mu \mu_a \bar{E}_\gamma \left(G_1 + \frac{k}{2} G_2\right) \end{aligned} \quad (7.69)$$

For cases where the cloud is large with respect to the range of the gamma radiation ($\mu \sigma_y > 10$), the ground concentration is essentially uniform in all directions and the value of

$\exp(-y^2/2\sigma_y^2)$ can be taken as unity for all values of y . Under these conditions the values of G_1 and G_2 can be written as

$$\begin{aligned} G_1 &= \frac{1}{(2)^{1/2} \mu \sigma_y} \int_0^{\infty} \frac{\exp[-\mu(l^2 + b^2)^{1/2}]}{(l^2 + b^2)} l dl \\ &= \frac{1}{(2)^{1/2} \mu \sigma_y} E_1[-\mu(l^2 + b^2)^{1/2}] \end{aligned} \quad (7.70)$$

$$\begin{aligned} G_2 &= \frac{1}{(2)^{1/2} \mu \sigma_y} \int_0^{\infty} \frac{\mu \exp[-\mu(l^2 + b^2)^{1/2}]}{(l^2 + b^2)^{1/2}} l dl \\ &= \frac{1}{(2)^{1/2} \mu \sigma_y} \int_{\mu b}^{\infty} \frac{\exp(-\mu b) d(\mu b)}{(2)^{1/2} \mu \sigma_y} \end{aligned} \quad (7.71)$$

The exponential integral, $E_1[-\mu(l^2 + b^2)^{1/2}]$, is as discussed earlier. These values for G_1 and G_2 provide the upper limits of the integrals. Values for smaller finite deposition patterns were obtained by Gamertsfelder by numerical evaluation of the integrals and are given in Figs. 7.26 and 7.27.

The ratio of the gamma dose rate above the center line of a deposition pattern smaller than an infinite plane to the dose rate above an infinite plane can be obtained from Eqs. 7.65 and 7.69. Figure 7.28 indicates this ratio for 0.7-Mev gamma photons. For most deposition patterns of interest, the correction factor for cloud size is comparatively small. Calculations for other energies indicate that Fig. 7.28 gives reasonable estimates of the ratio for energies up to 2 Mev but the difference may be 5% to 10% for the small pattern widths when the energy is as low as 0.1 Mev.

These equations provide an estimate of the gamma dose rate for a given amount of radio-

Fig. 7.26 — Values of the G_1 integral (see Eq. 7.68).

Fig. 7.27—Values of the G_2 integral (see Eq. 7.68a).

Fig. 7.28—Ratio of the gamma dose rate from a Gaussian pattern of finite size to that from an infinite plane pattern with deposition equal to the center line of the finite Gaussian pattern.

active material on the ground at a specific time. If radioactive decay or weathering is of importance during the period of exposure, the total dose received must be corrected for the decrease in dose rate. In the simplest case, i.e., a release of a single isotope and subsequent deposition, the decrease can be described by a single exponential, and the dose rate at any given time (t sec) after the cloud has passed the receptor can be written as

$$\gamma D'_{(x_1, 0, b, t)} = 0.36 \omega_{(x, 0, 0, 0)} \sigma_Y \mu \mu_a \bar{E}_\gamma \times \left(G_1 + \frac{k}{2} G_2 \right) \exp(-\lambda_R t) \quad (7.72)$$

where $\omega_{(x, 0, 0, 0)}$ is the center-line deposition based upon the source at the time of release and λ_R is the removal constant ($0.693/T_{1/2}$ with $T_{1/2}$ in seconds). Integrating from the cloud arrival time to the time at which the exposure estimate is desired gives the gamma dose equation for the finite deposition pattern.

$$\gamma D_{(x_1, 0, b)} = 0.36 \omega_{(x_1, 0, 0, t)} \sigma_Y \mu \mu_a \bar{E}_\gamma \times \left(G_1 + \frac{k}{2} G_2 \right) \left[\frac{1 - \exp(-\lambda_R t)}{\lambda_R} \right] \quad (7.73)$$

In Eq. 7.73 $\gamma D_{(x_1, 0, b)}$ is in rads, and t is the time interval in seconds after arrival over which the dose estimate is desired.

For more complex rates of removal, such as a power function, similar integrations over time may be carried out, but the resulting integrals will probably not be as easily solved and tables of integrals or numerical solutions may be needed.

7.5.4 Protection by Structures

The computational methods presented have assumed that the receptor is in the open, usually in the center of a large flat plain. In most cases there will be some reduction of dose if the receptor is in a building. Although the integrals for estimating the dose received inside a building have not been evaluated exactly, some estimates of the protection factor expected can be obtained by assuming simple geometrical conditions for which analytical solutions can be found. The following methods of estimating the protection factors afforded by structures have

been developed from methods suggested by Decker (1964).

7-5.4.1 Cloud Gamma Dose. A structure can provide protection both by the geometrical effect of limiting the distance of approach of the cloud and by shielding afforded by the walls. For geometrical protection it is assumed that the radioactive material does not enter the building (see, however, Sec. 7-5.4.3).

A rough estimate of the protection factor afforded by various structures can be obtained by assuming the cloud to be hemispherical in shape and uniform in concentration throughout and by using a hemisphere of radius a to approximate the building. Under these conditions the dose rate in the building will be given by

$$\gamma D' = (\text{constant}) B_x \exp(-\mu_x w) \times \int_a^{r_1} \frac{(1 + k\mu r) \exp(-\mu r) 2\pi r^2}{4\pi r^2} dr \quad (7.74)$$

where w = thickness of shielding afforded by building walls (g/cm²)

μ_x = mass absorption coefficient for wall material (cm²/g)

r = distance from receptor to elemental cloud volume (cm)

r_1 = cloud radius (m)

a = building radius (m)

μ = absorption coefficient for air (Fig. 7.8) (cm²/g)

B_x = dose buildup factor (dimensionless)

constant = dose-rate conversion factor

The dose buildup factor is that appropriate for the shielding material and the energies and directions of radiation falling on the shield.

Integrating Eq. 7.74 and substituting $B_x = (1 + k_x \mu_x w)$ gives

$$\gamma D' = \frac{\text{constant}}{2\mu} (1 + k_x \mu_x w) \exp(-\mu_x w) [\exp(-\mu a) - \exp(-\mu r_1) - k(1 + \mu r_1) \exp(\mu r_1) + k(1 + \mu a) \exp(-\mu a)] \quad (7.75)$$

If the building is not present, a and w equal zero, and Eq. 7.75 becomes

$$\gamma D' = \frac{\text{constant}}{2\mu} [1 - \exp(-\mu r_1) - k(1 + \mu r_1) \times \exp(-\mu r_1) + k] \quad (7.76)$$

The protection factor (P.F.) provided by the building is then the ratio of Eqs. 7.75 and 7.76

$$\begin{aligned} \text{P.F.} &= (1 + k_x \mu_x w) \exp(-\mu_x w) \\ &\times \left\{ [\exp(-\mu_a) - \exp(-\mu r_1) - k(1 + \mu r_1) \right. \\ &\times \exp(-\mu r_1) + k(1 + \mu_a) \exp(-\mu a)] \\ &\times [1 - \exp(-\mu r_1) \\ &\left. - k(1 + \mu r_1) \exp(-\mu r_1) + k]^{-1} \right\} \quad (7.77) \end{aligned}$$

The term in the braces, $\{\}$, expresses the protection factor from the geometrical considerations, whereas the term outside the braces expresses the shielding factor. These two factors have been evaluated separately.

Figure 7.29 presents the geometric protection factor for buildings up to 50 m in effective radius and for clouds both infinite in extent and with a radius of 100 m. (Note that the 50-m building occupies 12.5% of the 100-m radius cloud; no correction was made for this.) It is apparent that the geometrical protection is insignificant for most residential buildings in a very large cloud but becomes somewhat more important as the cloud becomes smaller.

The values given in Fig. 7.29 are for an individual in the center of the building. If he is against one wall, the reduction in dose would be approximately one-half the reduction for a hemisphere twice as large. Thus, from Fig. 7.29, the reduction factor in the center of a 10-m-radius building for an infinite cloud is 0.96. If the receptor were against one wall, the reduction would be one-half that for a 20-m building, or $1.0 - 0.075/2 = 0.963$.

The shielding effectiveness was evaluated with the mass absorption coefficient for most construction materials considered to be about the same as that of air for the energies of in-

terest. The mass absorption coefficients are given in Table 7.8 for materials and elements commonly used in construction (Wyckoff, 1959).

The attenuation through various thicknesses of material expressed as the dimensionless quantity μw was calculated from

$$\text{P.F.} = B_x \exp(-\mu_x w) \quad (7.78)$$

where B_x is the buildup factor for radiation passing through a thickness w of a material with an absorption coefficient μ_x . The factor B_x varies with the energy of the radiation, the atomic number of the shielding material, and the direction of the radiation falling on the shield. Figures 7.30 and 7.31 illustrate the buildup factors for various thicknesses of water, aluminum, and iron for 0.5- and 2.0-Mev photons originating as a point source and as a plane monodirectional source. The value for a point source in air as computed by $(1 + k\mu w)$ is given for comparison. In the present problem one would expect the source to be between the point source and the monodirectional plane source with a fraction of the photons falling on the shield degraded in energy to a value below that of the initial energy.

In view of these factors and the simplifying assumptions already made, the buildup factor for air was used in estimating the shielding-protection factor. This will underestimate the protection factor for low-shielding thickness of high atomic numbers but may overestimate it for thicker structures of low atomic numbers.

Figure 7.32 gives the protection factors for various thicknesses of shields for 0.3-, 0.7-, and 2-Mev gamma photons. For illustration, a building with 8-in.-thick brick walls and roof would have a shielding thickness of about 20 cm. For brick with a density of 1.9, the shielding mass thickness would be 38 g/cm^2 . For 0.7-Mev

Table 7.8—MASS ABSORPTION COEFFICIENTS FOR AIR AND ELEMENTS FOUND IN CONSTRUCTION MATERIALS

Energy, Mev	Mass absorption coefficients, μ_x , cm ² /g						
	Air	Concrete	Iron	Calcium	Silicon	Aluminum	Oxygen
0.1	0.151	0.169	0.344	0.238	0.172	0.161	0.151
0.3	0.106	0.107	0.106	0.109	0.107	0.103	0.107
0.5	0.0868	0.0870	0.0828	0.0876	0.0869	0.0840	0.0870
0.6	0.0804	0.0804	0.0762	0.0809	0.0802	0.0777	0.0806
0.8	0.0706	0.0706	0.0664	0.0708	0.0706	0.0683	0.0708
1.0	0.0635	0.0635	0.0595	0.0634	0.0635	0.0614	0.0636
1.5	0.0517	0.0517	0.0485	0.0518	0.0517	0.0500	0.0518
2.0	0.0445	0.0445	0.0424	0.0451	0.0447	0.0423	0.0445

Fig. 7.29 — Apparent dose reduction (protection factor, P. F.) afforded by the geometrical configuration of buildings.

gamma photons, with an absorption coefficient of $0.075 \text{ cm}^2/\text{g}$ the thickness in μw units would be 2.85 or the estimated protection factor from shielding would be about 0.33. For buildings of lighter construction, such as most wooden residences, the shielding is negligible.

Most buildings have relatively thin roofs; so the shielding from material overhead is negligible. The overall protection factor can be estimated by considering the fraction of the solid angle subtended by the roof and the walls and applying the appropriate shielding and geometrical corrections. Figure 7.33 gives the fraction of a hemisphere which is subtended by the roof for buildings of the given ratio of effective diameter to height.

From these data one can estimate the order of magnitude of dose reduction expected according to structure. For the usual wooden residence, about 20 by 20 by 10 m, only the geometric factor would be of any significance. From Fig. 7.29 this would amount to 93% to 94% of the outside dose for an individual in the center of the ground floor. If the walls are of 8-in.-thick

Fig. 7.30 — Buildup factors for 0.5-Mev gamma radiation.

brick, about 70% of the solid angle would be shielded and about 30% open (Fig. 7.33). Thus in an infinite cloud an individual in the center of the first floor would be exposed to $0.93 (0.7 \times 0.33 + 0.3) = 0.5$ of the cloud dose outside the building. If the individual were on the second floor halfway up the building (diameter/height ratio = 4), the angle subtended by the roof would be 0.55, and his protection factor would be $0.93 (0.45 \times 0.33 + 0.55) = 0.65$. Actually the geometric protection factor would not be as great as 0.93 in the latter case because the effective radius is smaller. If the individual were directly under the roof, the building would offer little or no protection. As a further illustration, the protection factor offered in the center of the ground floor of a large industrial building 100 by 100 by 20 m with 8-in.-thick brick walls against a large cloud emitting 0.7-Mev photons would be [assuming a geometrical protection factor of 0.86 (35-m radius) and a roof fraction of 0.63]

$$\text{P.F.} = 0.86 (0.37 \times 0.33 + 0.63) = 0.65$$

In addition, the presence of floors or equipment would increase the protection factor.

The foregoing methods lay no claim to accuracy, but they do illustrate the approximate protection available from a passing cloud. Except for small clouds it appears that the ordinary residence provides little reduction in dose and that for large clouds only massive buildings with complete shielding are effective.

Fig. 7.31—Buildup factors for 2.0-Mev gamma radiation.

7-5.4.2 Radiation from Deposited Materials. The radiation dose rate at a height b above a uniformly contaminated area is given by Eq. 7.65a. If the building provides a clean area around the receptor of radius a (m), the dose rate from the material outside the clean area is given in units of rads per second by

Fig. 7.32—Shielding protection factors.

$$\gamma D' = 0.254 \omega \bar{E}_\gamma \mu_a \left[E_i (\mu \sqrt{a^2 + b^2}) + \frac{k}{2} \exp (-\mu \sqrt{a^2 + b^2}) \right] \quad (7.79)$$

The ratio of Eq. 7.79 to Eq. 7.65a then gives the geometrical protection factor

$$P.F. = \frac{E_i (\mu \sqrt{a^2 + b^2}) + (k/2) \exp (-\mu \sqrt{a^2 + b^2})}{E_i (\mu a) + (k/2) \exp (-\mu a)} \quad (7.80)$$

Figure 7.34 presents the reduction in dose at 1 m above grade and at 4 m above grade as compared to the dose 1 m above an infinite plane contaminated to a uniform level with radioactive materials emitting 0.7-Mev gamma rays. It can be seen that the reduction in dose on the second floor of an unshielded building is negligible.

This treatment ignores any deposition on the roof and sides of the building and the resulting radiation dose. This dose can be approximated by considering the roof to be a disk source at a distance b_1 meters from the receptor. If the shielding of the roof and floor materials is ignored, this dose rate is given by

$$\gamma D' = 0.254 \omega \bar{E}_\gamma \mu_a \left[E_i (\mu b_1) - E_i (\mu \sqrt{b_1^2 + a^2}) + \frac{k}{2} \exp (-\mu b_1) - \frac{k}{2} \exp (-\mu \sqrt{b_1^2 + a^2}) \right] \quad (7.81)$$

Fig. 7.33—Fraction of hemisphere subtended by roof of building with various ratios of diameter to height.

where a is the diameter of the roof. The ratio of Eq. 7.81 to Eq. 7.65a will give the geometrical protection factor.

$$P.F. = \left[E_i (\mu b_i) - E_i (\mu \sqrt{b_i^2 + a^2}) + \frac{k}{2} \exp(-\mu b_i) - \frac{k}{2} \exp(-\mu \sqrt{b_i^2 + a^2}) \right] \times \left[E_i (\mu b) + \frac{k}{2} \exp(-\mu b) \right]^{-1} \quad (7.82)$$

Equation 7.82 was solved with account taken of the ratio of the dose received at various distances from the roof to the dose received 1 m above a uniformly contaminated infinite plane emitting 0.7-Mev gamma rays. These results are given in Fig. 7.35.

For an illustration of the protection afforded, consider a wood frame house 20 m square and 10 m high. From Fig. 7.34 the dose at the center of the ground floor will be about 40% of the infinite plane dose. At the ground floor the roof is 10 m away with a diameter of 20 m. From Fig. 7.35 this will contribute an additional 7% for a total of 47%. On the second floor where the roof is 3 to 5 m away, the roof will contribute 20%, and the dose from the surrounding

Fig. 7.34—Geometrical protection from ground deposit of 0.7-Mev gamma radiation.

Fig. 7.35—Radiation received from deposit on roof given as a fraction of dose at 1 m above an infinite plane for 0.7-Mev gamma radiation.

area is about the same. For this floor the protection factor drops to about 60% of the dose 1 m above an infinite plane. These calculations assume that the roof is contaminated to the same level as the surrounding area and that no radioactive material is adhering to the walls.

If the same house is made of 8-in.-thick brick, the radiation from the ground outside is reduced by a factor of 0.33 from Fig. 7.32, and the protection factor becomes about 0.2 on the ground floor and about 0.33 on the second floor.

7.5.4.3 Contamination Inside a Building. The previous geometrical protection factors are based upon the assumption that the building provides a noncontaminated area or volume around the individual. Most buildings unless specifically designed to be airtight have changes of air from leaks in the structure resulting from pressure

differences caused by wind velocities or by temperature differences in heated buildings. If windows or doors are open, interchange of air with the outside can be relatively rapid.

If a building is surrounded by a cloud of average concentration $\bar{q}$, the amount of activity, dq , entering is given by

$$dq = V\bar{q}dt - \frac{Vq}{s} dt \quad (7.83)$$

where V is the ventilation rate in cubic meters per time interval and s is the volume of the building (q/s is the concentration in the building). This equation assumes that the activity entering is uniformly mixed through the building upon entry. Solving Eq. 7.83 and using the condition that $q/s = 0$ at zero time gives the concentration in the building as

$$\frac{q}{s} = \bar{q} \left[1 - \exp \left(-\frac{V}{s} t \right) \right] \quad (7.84)$$

If V/s is one change per hour (a reasonably tight building with low winds), the concentration inside will build to 10% of that outside in 6 min, to 50% in 40 min, and to 90% in 2.3 hr. If $V/s = 6$, the concentration inside will build to 10% of that outside in 1 min, to 50% in 7 min, and to 90% in 23 min.

Since the total dose received depends upon the integral of the concentration vs. time curve, it is of interest to compare this integral inside the building with that outside for a transient cloud. If the time of cloud passage is taken as t_1 (with the time of cloud arrival as t_0), then the integrated concentration is given by

$$\left(\frac{q}{s} \right)_f = \int_0^{t_1} \bar{q} \left[1 - \exp \left(-\frac{V}{s} t \right) \right] dt + \int_0^{\infty} \left(\frac{q}{s} \right)_{t_1} \exp \left(-\frac{\eta V}{s} t \right) dt \quad (7.85)$$

where $(q/s)_{t_1}$ is the concentration in the building after the cloud passage. The ventilation rate after cloud passage is taken as $\eta V/s$, where η is a factor describing possible changes in the ventilation rate due to changes in wind speed or to deliberate actions, such as opening windows or turning on ventilation.

Solving and substituting for (q/s) at t_1 from Eq. 7.84 gives

$$\left(\frac{q}{s} \right)_f = \bar{q} \left\{ t_1 - \frac{s}{V} \times \left[1 - \exp \left(-\frac{V}{s} t_1 \right) \right] \left(1 - \frac{1}{\eta} \right) \right\} \quad (7.86)$$

Note that if the ventilation rate remains the same, $\eta = 1$, the integral concentration inside becomes $\bar{q}t_1$ which is the same as that outside. If the ventilation rate decreases (as, for example, by diminishing wind speed), the integral concentration inside is greater than the concentration time integral outside. Equation 7.86 indicates that a defensive measure against a transient cloud would consist in closing window and doors and sealing cracks during cloud passage and then increasing the ventilation rate as rapidly as possible after the cloud passage. This may be impractical since it depends upon a knowledge of the exact time of cloud arrival and the exact time when the bulk of the cloud has passed.

In practice the concentration inside may be lower than outside, depending upon the physical and chemical characteristics of the contaminant, since the incoming air as it passes through narrow cracks or small holes may deposit some materials because of inertial forces or chemical affinity. Note that in this case there will be localized areas of higher surface contamination. Megaw (1962), for example, measured the concentrations of ^{132}I as vapor inside and outside a hut with ventilation rates of 1.8 to 5.5 changes per hour and found integral concentrations inside ranging from about 20% to 70% of those outside. When the windows were open at one end of the hut, the ventilation rate increased to 5 to 8 changes per hour, and the integral concentrations inside were about 70% of those outside. Iodine vapor is known as a very strong depositing material; so the influence of deposition mechanisms would be expected to be strong.

Megaw also studied the penetration into the same hut of Aitken nuclei produced by burning paraffin-soaked sand. For ventilation rates ranging from 0.8 to 2.5 changes per hour, the integral concentrations inside ranged from 34% to 78% of those outside.

Although the above treatment indicates that high concentrations inside the building are possible, it is expected that the deposition inside buildings will be considerably less than that outside, both due to lower turbulence levels in-

side and due to the restriction on the source by the limited rate of entry to the building. The experiments by Megaw again confirm this expectation since the deposition inside was only a few percent of that outside. The deposition velocity outside ranged from 0.3 to 3.1 cm/sec and that inside ranged from 0.1 to 0.24 cm/sec.

7-6 DOSE FROM INHALED MATERIALS

Inhaling radioactive materials can result in radiation doses to (1) the lungs and upper respiratory tract from isotopes deposited in the passages, (2) the gastrointestinal tract (GI tract) from isotopes originally deposited in the respiratory tract and later removed by ciliary action and swallowed, and (3) other organs in which the isotopes may be deposited after they are absorbed into the bloodstream from either the lung or GI tract.

The radiation dose to each organ will depend upon the physical and chemical characteristics of the radionuclides in the cloud and the distribution and retention of these materials in the body. In general, the physical and chemical characteristics of the routine radioactive wastes normally emitted from a facility can be determined with reasonable accuracy, but the materials from an accidental release will not be well characterized in advance. Even with accurate characterization the metabolic parameters involved, such as retention in the lung, solubility in body fluids, and translocation to other organs, are not known well enough to do more than provide an estimate of the radiation dose involved. The primary mathematical models and parameters for the calculation of the doses from these internal sources has been set by the Internal Dose Subcommittees of the National Committee on Radiation Protection (National Committee on Radiation Protection and Measurement, 1959) and the International Commission on Radiological Protection (International Commission on Radiological Protection, 1959a).

7-6.1 Intake, Retention, and Translocation of Inhaled Radioactive Materials

The total amount of radioactive material inhaled by a person during cloud passage is determined by the product of the cloud concentra-

tion time integral ψ (curie sec/m³) times the breathing rate B' (m³/sec) during cloud passage. Equations for ψ were developed in Chap. 3, and modifications to incorporate the effect of radioactive decay were developed earlier in this chapter.

The breathing rate for the standard man is given as 10 m³/8-hr work day and 20 m³/24-hr day. These are 3.48×10^{-4} m³/sec and 2.31×10^{-4} m³/sec, respectively. During times of extreme stress the breathing rate can be greater, and a breathing rate of 5×10^{-4} m³/sec is frequently assumed for relatively short periods of time. The breathing rate selected depends upon the details of the situation considered.

Two biological factors are of importance in assessing the overall dose to the lung: the fraction of the inhaled material that is deposited and the time interval during which the deposited material is retained in the lung. The deposition and site of deposition, whether in the upper or lower pulmonary tract, will vary with particle size, shape, and density. Different amounts and different patterns of deposition will occur, depending upon whether an individual breathes through his nose or his mouth. The time of retention in the lung will depend upon the site of deposition and the solubility of the material in body fluids. Material deposited in the upper respiratory tract will be quickly cleared by ciliary action; that deposited deep in the lung will be removed primarily by solution into the bloodstream or, in some cases, by transfer of the particle across the lung alveoli. Standard assumptions concerning the fate of soluble and insoluble radioactive particles in the lung have been made by the ICRP (International Commission on Radiological Protection, 1959a) to facilitate dose estimates. These assumptions, which are given in Table 7.3, should be used only when specific data on the particular material of interest are not available.

As indicated in Table 7.3 inhaled material that is soluble in body fluids will be translocated via the GI tract and through the bloodstream to other organs. The fraction of a soluble radionuclide which is translocated to the critical organ (generally the body organ that subsequently receives the highest dose from the material) is given by

$$f_a = (0.25 + 0.5 f_1) f_2' \quad (7.87)$$

where f_a is the fraction of a soluble radionuclide which is translocated to a critical organ, f_1 is the fraction of the radionuclide passing from the gastrointestinal tract to the blood, and f_2' is the fraction of radionuclide passing from the blood to the critical body organ.

When f_2' is not known, it is generally replaced by f_2 , the fraction of the total body radionuclide burden in the critical organ.

When the material inhaled is either insoluble or slightly soluble, a portion of the GI tract or the lung is usually the critical organ. When better data for the inhalation of specific radioactive insoluble dust particles is not available, the value of f_a is assumed to be 0.12 in the case of the lungs. Some of the inhaled radioactive material is swallowed and irradiates the GI tract. If a portion of the GI tract is the critical tissue, the value of f_a is given as 0.62 for insoluble material and 0.5 for soluble material. Values of the various biological factors have been tabulated by the ICRP (International Commission on Radiological Protection, 1959a).

Although in many instances it is difficult to predict the biological fate of the source material released from a facility, radioiodine in most forms will be quickly absorbed from the lungs into the blood stream and translocated to the thyroid. The noble gases do not enter into the body metabolism but may be dissolved to some extent in body fluids or fat. Since the physical and chemical characteristics of the particles are generally unknown, the common practice is to make two dose calculations: one based on the assumption that the radioactive material is soluble and the other based upon the assumption that the material is insoluble.

7-6.2 Dose to a Body Organ from Internal Radiation

The dose rate to any body organ from internal irradiation is given by

$$D' = (qf_2) \frac{(3.7 \times 10^{10}) \bar{E} (1.6 \times 10^{-6})}{100 M} \\ = 5.92 \times 10^2 (qf_2) \frac{\bar{E}}{M} \quad (7.88)$$

where D' = dose rate to the organ from internal irradiation (rad/sec)
 M = mass of the organ (g)

q = total quantity of radioactive material in the total body (curie)

f_2 = fraction of the total body radionuclide burden in the critical organ

qf_2 = quantity of radioactive material in the critical organ (curie)

3.7×10^{10} = disintegration rate (dis/sec) for 1 curie

$\bar{E}$ = energy absorbed in the organ from one disintegration (Mev)

1.6×10^{-6} = number of ergs per Mev

100 = number of ergs per gram per rad

The values of $\bar{E}$, the average energy absorbed in the organ per disintegration, can be taken as equal to the average energy released per disintegration for alpha and beta emitters because the ranges of these particles are small compared to the organ dimensions. For this reason nearly all the energy is released in the organ containing the radioactive material. For beta particles the average energy can be assumed to be one-third of the maximum energy, and the resulting error will be well within other uncertainties. If more precise information is needed, measured or calculated values can be found in the literature (Hine and Brownell, 1956). For most gamma photons the range in tissue will be considerably greater than the organ dimensions; so only a portion of the energy released will be absorbed in the organ. This fraction can be estimated by assuming the organ to be approximately spherical. Under these conditions the fraction absorbed will be approximately

$${}_a\bar{E}_\gamma = {}_0\bar{E}_\gamma [1 - \exp(-\mu_a r_0)] \quad (7.89)$$

where ${}_a\bar{E}_\gamma$ = the average gamma energy absorbed per disintegration

${}_0\bar{E}_\gamma$ = the average gamma energy emitted per disintegration

μ_a = the energy absorption coefficient for the gamma photon energy

r_0 = the effective diameter of the organ

For many of the calculations involving small organs, the gamma contribution to the dose can be neglected if mixed radiation is present. Total values of the average energy absorbed by various organs per disintegration for specific radionuclides are tabulated in the ICRP report (International Commission on Radiological Protection, 1959a).

The dose delivered to an organ by internal irradiation depends upon the retention of the radioactive material in the organ over the time period of concern. For the total dose it is necessary to integrate the quantity of radioactive material in the organ over the period of time of concern with proper allowance for accumulation, radioactive decay, and biological elimination. In the following discussion the intake is assumed to occur in a short time interval, e.g., exposure to a relatively small cloud from a puff release. Subsequently it will be shown that this assumption is not restrictive in most cases.

$$D = 5.92 \times 10^2 \frac{\bar{E}}{M} \int_{t_A}^{t_2} (qf_2) dt \quad (7.90)$$

where D is the dose to the organ (rad), t_A is the time at which the irradiation starts (sec) ($t_A = x/\bar{u}$ for a puff release), and t_2 is the time to which the total dose is to be calculated (sec). (This assumes a negligible transit time for the radionuclide between intake and deposition in the organ.) The value of (qf_2) at a given time after uptake depends upon both the elimination by radioactive decay and the elimination by biological processes.

$$\int_{t_A}^{t_2} (qf_2) dt = (qf_2)_0 \int_{t_A}^{t_2} f_R(t) f_B(t) dt \quad (7.91)$$

where $(qf_2)_0$ is the quantity of radioactive material in the organ initially (curie), $f_R(t)$ is the function chosen to describe the radioactive decay of the mixture in the organ, and $f_B(t)$ is the function chosen to describe the rate of biological elimination of the mixture from the organ.

In Eq. 7.91, $f_B(t)$ is frequently described by a single exponential function of time. For most radionuclides such an exponential is taken to describe only the long-term elimination with the assumption that the majority of the radiation dose is due to this long component. For many radionuclides the retention may be more accurately described by a series of several exponentials to describe different compartments or by a power function fitted to the retention curve over the period of interest (Healy, 1956).

The value of $f_R(t)$ for a single radionuclide is a single exponential function. For a group of radionuclides, the value of $f_R(t)$ can be represented over some time periods by a power function of the time. The well-known Way-Wigner

equation for decay of fission products following a nuclear detonation or critical accident involving a short burst in which the fission products decay as $t^{-1.2}$ is a power function describing the gross decay of the mixture (Way and Wigner, 1948). For very long periods of reactor operation and decay times that are relatively short compared to the time of operation, the gross decay of the fission mixture is frequently approximated as $t^{-0.2}$. For the actual time of operation and fractionation chosen, several power functions can frequently be fitted to the gross-decay curve.

From these combinations of methods for expressing the radioactive decay and the biological elimination, it is possible to produce a number of different methods for calculating the quantity of radioactive material in the organ. The particular function chosen will depend upon the information available and the type of calculation to be performed. If both the retention and the radioactive decay are expressed by a single exponential,

$$\int_{t_A}^{t_2} (qf_2) dt = (qf_2)_0 \int_{t_A}^{t_2} [\exp(-\lambda_R t) \times \exp(-\lambda_B t)] dt \quad (7.92)$$

where λ_R is the radioactive decay constant and equals $0.693/T_R$ (1/sec), λ_B is the biological elimination constant and equals $0.693/T_B$ (1/sec), and T_R and T_B are the radioactive and biological half-lives (sec).

Substituting $\tau = (t_2 - t_A)$ and integrating gives

$$\int_0^{\tau} (qf_2) dt = \frac{(qf_2)_0}{\lambda_B + \lambda_R} \times \{1 - \exp[-\tau(\lambda_B + \lambda_R)]\} \quad (7.93)$$

where τ is the time interval for which the cumulative dose is evaluated (sec).

If, compared to the human lifetime, the residence time in the body is short owing to biological elimination or radioactive decay, then the total number of disintegrations in the organ (or dose instead of dose rate when Eq. 7.90 is involved) can be obtained by letting τ equal infinity.

$$\int_0^{\infty} (qf_2) dt = \frac{(qf_2)_0}{\lambda_B + \lambda_R} \quad (7.94)$$

If both the retention and the radioactive decay are described by power functions, the integral can be expressed as

$$\int_{t_A}^{t_2} (qf_2) dt = (qf_2)_i \int_0^{\tau} t^{-n} t^{-m} dt \quad (7.95)$$

where n is the slope of the logarithmic plot of radioactivity vs. time, m is the slope of the logarithmic plot of retention vs. time, and $(qf_2)_i$ is the quantity of radioactive material in the critical organ after exposure ceases. In this case

$$\int_{t_A}^{t_2} (qf_2) dt = \frac{(qf_2)_i \tau^{[1-(n+m)]}}{1 - (n + m)} \quad (7.96)$$

If the radioactive decay is expressed as a power function and the retention as an exponential, then the integral can be expressed as

$$\int_{t_A}^{t_2} (qf_2) dt = \int_0^{\tau} t^{-n} \exp(-\lambda_B t) dt \quad (7.97)$$

In Eq. 7.97 the integral reduces to a gamma function if $n < 1$. If τ is taken as infinity, the integral is a complete gamma function. For evaluation of a time interval, τ , less than infinity, the integral is an incomplete gamma function and may be evaluated from Pearson's tables (Pearson, 1951). For values of $n > 1$, the integral must be evaluated by other means.

7-6.2.1 Critical Organ Dose Due to Finite Inhalation Exposure Time. A fraction of the radioactive material inhaled will be absorbed into the blood stream from either the lung or the GI tract and transferred to other organs, e.g., a significant fraction of the radioiodine absorbed into the blood will be deposited in the thyroid and a significant fraction of radiostrontium will be deposited in the bone. Experiments on the inhalation of water-insoluble silver iodide and strontium sulfate have indicated that the translocation process for these materials is as efficient as if the iodine or strontium were present as water-soluble compounds (Willard and Bair, 1958).

For an individual exposed to a uniform concentration of a nuclide for a finite period of time, the change in the quantity of the nuclide in the critical organ during the period of exposure is given by

$$d(qf_2) = B' \bar{\chi} f_a dt - (qf_2) \lambda dt \quad (7.98)$$

where B' is the breathing rate (m^3/sec), f_a is the fraction of inhaled material that is translocated to the critical organ, and λ equals $\lambda_R + \lambda_B$ (sec^{-1}).

The cumulative dose in rads from the onset of plume inhalation until the exposure ceases is obtained by integrating the amount in the critical organ over the time of exposure and converting to dose.

$${}_1D = \frac{5.92 \times 10^2 \bar{E}}{M\lambda} f_a B' \bar{\chi}_{(x,y,0)} \times \left[\tau - \frac{1 - \exp(-\lambda\tau)}{\lambda} \right] \quad (7.99)$$

where ${}_1D$ is the dose from onset of plume inhalation until exposure ceases (rad) and τ is time interval of exposure (sec).

If the exposure to the plume ceases after the time interval of τ seconds, the irradiation of the critical organ will continue owing to the burden in the critical organ at that time. The dose received in the time interval of τ_1 seconds immediately following the exposure is obtained by substituting the body burden at the end of the exposure period into Eq. 7.88 and integrating.

$${}_2D = \frac{5.92 \times 10^2 \bar{E}}{M\lambda^2} f_a B' \bar{\chi}_{(x,y,0)} \times [1 - \exp(-\lambda\tau)] [1 - \exp(-\lambda\tau_1)] \quad (7.100)$$

where ${}_2D$ is the dose received in time interval following exposure (rad) and τ_1 is the time interval following exposure (sec). The total cumulative dose delivered from the time that exposure to the plume started until some time after the end of exposure, i.e., $T = (\tau + \tau_1)$ seconds, is the sum ${}_1D + {}_2D$ (rads).

$$D_{co} = \frac{5.92 \times 10^2 \bar{E}}{M\lambda^2} f_a B' \bar{\chi}_{(x,y,0)} \{ \lambda\tau - \exp(-\lambda\tau_1) + \exp[-\lambda(\tau + \tau_1)] \} \quad (7.101)$$

where D_{co} is the cumulative dose from the beginning of exposure to the plume until some time after the end of exposure (rad).

There is one observation concerning this development that deserves special consideration. When the irradiation time τ_1 (time interval after cloud exposure) is long compared to the effective half-life of the radioactive material, i.e.,

$\lambda\tau_1 \rightarrow \infty$, the total dose to the critical organ is given

$$D_{co} = \frac{5.92 \times 10^2 \bar{E}}{M\lambda} f_a B' \psi_{(x,y,0)} \quad (7.102)$$

where D_{co} is the total dose when irradiation time is long compared to half-life (rad) and $\psi_{(x,y,0)}$ equals $\bar{\chi}_{(x,y,0)}\tau$. The total dose (dose commitment) to the critical organ depends upon the amount of radioactive material inhaled without regard to the time required to inhale the amount.

7-6.2.2 Lung Dose. When the radioactive material inhaled is insoluble and the other characteristics of the material are unknown, the ICRP values for the standard man are used (see Table 7.3). In the following development for a short exposure time, a breathing rate of $5 \times 10^{-4} \text{ m}^3/\text{sec}$ (30 liters/min) will be used. This is higher than the breathing rate of the standard man to allow for a heightened respiratory rate during periods of exertion.

The quantity of insoluble material (in curies) deposited in the lungs immediately after cloud passage is given by

$$\begin{aligned} (qf_2)_0 &= (5 \times 10^{-4}) (0.25) \psi_{(x,y,0)} \\ &= 1.25 \times 10^{-4} \psi_{(x,y,0)} \end{aligned} \quad (7.103)$$

where 5×10^{-4} is the breathing rate (m^3/sec).

The ICRP standard man allows for a decrease of 50% in the lung burden in the first day but does not specify the rate of this decrease or the form of the function. It will be arbitrarily assumed that the lung burden consists of two components, each consisting initially of 12.5% of the amount inhaled (to give a total initial retention of 25%). The long-lived component is eliminated with a half-life of 120 days; the short-lived component is eliminated with a half-life such that only about 1% of the initial quantity is present at the end of 1 day ($T_b = 4 \text{ hr}$). Note that there will be additional exposure not accounted for in these equations from the radioactive material deposited in the nasal passages and the upper bronchi. Since this material will be removed rather rapidly by ciliary action and swallowed, the total dose received by these organs is taken to be less than the dose received by the lung.

It should be recognized that the preceding assumptions concerning the behavior of material in the lung represent a standardized model that is valuable for estimating the order of magnitude of the expected dose. Under some circumstances the deposition in the areas of the bronchi where the air flow changes direction can result in localized doses that are higher than the average dose to the lung. This is recognized as an important factor in the long-term exposure to the relatively short-lived daughters of radon among uranium miners.

There is some concern about the possible effects of one or a few intensely radioactive particles that may penetrate to the lung. Although the total quantity of radioactive material carried by this mechanism may be small enough so that the dose averaged over the entire lung appears insignificant, the actual dose in the immediate vicinity of these few particles may be very high (National Academy of Science—National Research Council, 1961). At the present time it is not possible to state whether these particles represent a greater or a smaller hazard than the same amount of material uniformly distributed through the lung. It may also be noted that defects in the ciliary mechanism may permit a particle to be retained in the upper respiratory tract and, perhaps, produce localized damage. Such defects do occur in a reasonable proportion of the populace. There is also considerable evidence that the transfer of particulate material to the lymph nodes results in a very high concentration in a relatively small mass of tissue (Thomas, 1965). Although available evidence would not justify basing damage estimates on the dose to the lymphatic system, neither can the dose to the lymphatic system be ignored.

The total retention in the lung can then be written as the sum of two exponentials

$$\begin{aligned} (qf_2) &= (qf_2)_0 [\exp(-\lambda_1 t) \\ &\quad + \exp(-\lambda_2 t)] f_R(t) \end{aligned} \quad (7.104)$$

where t = time interval since inhalation (sec)

λ_1 = elimination constant of the short component ($0.693/4 \text{ hr}^{-1}$)

$$= 4.8 \times 10^{-5} \text{ sec}^{-1}$$

λ_2 = elimination constant of the long component ($0.693/120 \times 24 \text{ hr}^{-1}$)

$$= 6.7 \times 10^{-8} \text{ sec}^{-1}$$

$f_R(t)$ = radioactive decay function

For individual radionuclides that decay by a single exponential, the total dose to the lung is then given by combining Eqs. 7.88, 7.92, 7.103, and 7.104 and allowing 50% of the material from Eq. 7.104 in each component.

$$D_L = \frac{\bar{E}\psi_{(x,y,0)}}{M} \left\{ \frac{3.7 \times 10^{-2}}{\lambda_R + \lambda_1} [1 - \exp - (\lambda_R + \lambda_1)\tau] + \frac{3.7 \times 10^{-2}}{\lambda_R + \lambda_2} [1 - \exp - (\lambda_R + \lambda_2)\tau] \right\} \quad (7.105)$$

where D_L is the total dose to the lung (rad), τ is the time interval after cloud arrival over which the cumulative lung dose is desired (sec), and M is the mass of lung (g).

Equations 7.88, 7.97, 7.103, and 7.104 can be used to calculate the dose when the radioactive decay of the gross mixture of radioactive materials is described by a power function.

$$D_L = \frac{3.7 \times 10^{-2} \bar{E}\psi_{(x,y,0)}}{M} \left[\int_0^{\tau_2} t^{-n} \exp(-\lambda_1 t) dt + \int_0^{\tau_2} t^{-n} \exp(-\lambda_2 t) dt \right] \quad (7.106)$$

7-6.2.3 Dose to the Gastrointestinal Tract from Inhalation of Insoluble Radioactive Materials. ICRP standard man model is given in Table 7.3. When insoluble radioactive material is inhaled, 62.5% of the material is eliminated through the GI tract (50% deposited in the upper respiratory tract and later swallowed, plus one-half of the 25% initially deposited in the lungs). The 62.5% eliminated through the GI tract clears the respiratory system in 24 hr. If the exact rate of elimination from the respiratory system were defined, the dose to various portions of the GI tract could be obtained by integrating the dose rate over the time of passage through any portion of the GI tract. The mass content and residence time for various portions of the GI tract are given in Table 7.3 (International Commission on Radiological Protection, 1959a). The portion of the GI tract that receives the maximum dose will obviously depend upon the specific biological removal model assumed for the lung and the radiological half-life of the radioactive material. If the radioactive half-life of the material is relatively long, the dose to the lower large intestine (LLI) will be controlling because of the longer residence time in that

portion. For short-lived material the greater dose may be received by the upper portions of the GI tract.

Since the exact model for the clearing of material from the respiratory system is not given, a number of calculational methods are possible. For the purpose of demonstrating a calculational technique, it will be assumed that the clearing will be exponential with a biological half-life of 4 hr. A single radionuclide will be considered, and the inhalation will be assumed to occur over a short period of time.

The amount of insoluble radioactive material inhaled and initially deposited in the respiratory system is given in curies by

$$(qf_2)_0 = 0.75 B' \psi_{(x,y,0)} \quad (7.107)$$

The amount of material subject to removal via the GI tract is

$$(qf_2)_1 = 0.625 B' \psi_{(x,y,0)} \quad (7.108)$$

The amount of material in this component remaining in the respiratory system is given by

$$(qf_2)_2 = 0.625 B' \psi_{(x,y,0)} \exp(-\lambda t) \quad (7.109)$$

where λ equals $\lambda_B + \lambda_R$, λ_B is the biological elimination constant, and t is the time interval after intake by inhalation (sec). (Note that when $T_B = 4$ hr and $\lambda_B = 4.81 \times 10^{-5} \text{ sec}^{-1}$ virtually all of the material is removed from the respiratory system in 24 hr. Figure 7.36 may be helpful in following this development. It indicates the quantity of material in various locations as a function of time, but, for simplicity, the decay corrections are not shown.) The total amount of material (in curies) swallowed in the time interval t is

$$(qf_2)_s = 0.625 B' \psi_{(x,y,0)} [1 - \exp(-\lambda t)] \quad (7.110)$$

Since this component is insoluble in body fluids, the quantity that entered the lower large intestine (LLI) will be the amount swallowed corrected for radioactive decay en route, i.e., 13 hr,

$$(qf_2)_{LLI} = 0.625 B' \psi_{(x,y,0)} \exp(-4.67 \times 10^4 \lambda_R) \times [1 - \exp(-\lambda \tau)] \quad (7.111)$$

Fig. 7.36 — Reference system for GI-tract dose considerations for inhaled insoluble materials.

where τ is the time interval (sec) after material enters LLI.

Material entering the LLI is retained 18 hr; therefore, if a continuous elimination is assumed, the amount in curies eliminated is given by

$$(qf_2)_{\text{Elim.}} = 0.625 B' \psi_{(x,y,0)} \exp(-6.48 \times 10^4 \lambda_R) \times \{1 - \exp[-(\tau - 6.48 \times 10^4) \lambda]\} \quad (7.112)$$

where $\tau \geq 6.48 \times 10^4$ sec. The amount in the LLI is obtained by subtracting Eq. 7.112 from Eq. 7.111 to give

$$(qf_2)_{\text{LLI}} = 0.625 B' \psi_{(x,y,0)} \left(\exp(-4.67 \times 10^4 \lambda_R) \times [1 - \exp(-\lambda \tau)] - \exp(-6.48 \times 10^4 \lambda_R) \times \{1 - \exp[-(\tau - 6.48 \times 10^4) \lambda]\} \right) \quad (7.113)$$

Integrating Eq. 7.113 with respect to time gives

$$\int_0^\infty (qf_2)_{\text{LLI}} dt = 4.05 \times 10^4 B' \psi_{(x,y,0)} \times [\exp(-4.67 \times 10^4 \lambda_R) - \exp(-6.48 \times 10^4 \lambda_R)] \quad (7.114)$$

where $t \geq 50$ hours, $\tau \rightarrow \infty$, and the result is in terms of curie seconds.

The dose to the LLI from inhalation can be calculated by substituting Eq. 7.114 into Eq. 7.88.

$$D_{\text{LLI}} = \int_0^\infty D' dt = 5.92 \times 10^2 \frac{\bar{E}}{M} \times \int_0^\infty (qf_2)_{\text{LLI}} dt \quad (7.115)$$

$$D_{\text{LLI}} = 1.97 \bar{E} \int_0^\infty (qf_2)_{\text{LLI}} dt \quad (7.115a)$$

where $M = 150$ g (content of LLI) and irradiation of the intestine walls is assumed to occur from one side only (0.5 geometry). As stated at the onset, this is but one method of solution and is intended only as a point of departure for better calculational methods.

7-7 DOSE FROM INGESTED MATERIALS

The deposition of radioactive materials from the atmosphere can result in general contamination of an area so that, in addition to the direct radiation, radiation can be received from the later ingestion of water or of foodstuffs grown in the area. This type exposure is often the one that provides the lowest limit for the release of wastes during normal operating conditions and the one that requires countermeasures over the greatest area for accident conditions, e.g., the confiscation of milk produced in the area around the reactor following the Windscale release of ^{131}I .

Such contamination can result either by direct deposition of the radioactive materials on the foodstuff or by incorporation of the radioactive material into the nutrients of a food chain. Because of the many uncertainties in the deposition process and the chemical and biological features of the food chains, direct estimates of the potential contamination are difficult and uncertain. For this reason much of the available information relates to experience with individual radioisotopes in a given environment, and generalization of the experience is available in only a few cases. Thus only general consideration will be given to dose estimates from these chains.

7-7.1 Contamination of Vegetation

Consideration of the mechanisms of contamination of vegetation is of importance because the vegetation may serve directly as a source of diet for man or it may serve as an entry into a food chain for another source of diet, e.g., the movement of iodine from contaminated pasture to the milk of cows.

7-7.1.1 Direct Deposition. The contamination of plants due to direct deposition on the foliage is a transitory problem in the sense that only a single crop is affected by this process and the

contamination is reduced by radioactive decay, weathering, plant growth (and consequent dilution by uncontaminated vegetation), and, eventually, harvesting or a new cycle of growth. Materials deposited in this fashion provide a direct path of radiation without the time delay inherent in the root uptake and plant metabolism. As a result, this deposition is of importance for relatively short-lived radioisotopes that would be largely eliminated by radioactive decay before significant incorporation into the plant through the soil-uptake route. The importance of this mechanism for introducing radioactive materials into the diet will vary with the season of the year, i.e., it will be of less importance in some areas during the winter when crops are not growing and milk cows are usually fed with stored feed.

Available data on the distribution and retention of radioisotopes on vegetation are largely based upon studies of debris from nuclear weapons; so the data represent the behavior of the chemical and physical forms characteristic of this debris, as well as, in many cases, the integrated effects of deposition over a long time period. The behavior of this material depends upon the type of vegetation. It has been noted that the shape of wheat inflorescences tends to maximize particle entrapment (Eisenbud, 1963). About 2% of the ^{90}Sr deposited during the time the heads of the wheat plant were exposed was retained in the grain, and the whole crop retained about 3% of the deposited material (National Academy of Sciences—National Research Council, 1963). Less than 0.1% of the applied strontium was found in the wheat grain if the surface deposition occurred before the heads of the wheat developed. Rye grass grown in flats absorbed directly 23% of the ^{90}Sr .

Certain chemical forms of some radioisotopes can be absorbed through the plant foliage. About 4% of the strontium and 2% of the ruthenium applied to tomatoes is absorbed through the intact skin of the fruit (National Academy of Sciences—National Research Council, 1963). Gaseous ^{131}I will adsorb on plant leaves; about 35% to 40% penetrates the leaf epidermis, and about 2% moves to other parts of the plant. Stable iodine will not reduce the ^{131}I adsorption but does reduce the translocation (Selders and Hungate, 1956). The plant species is of importance in the absorption of elements. Wheat plants absorbed

85% of the applied strontium and 93% of the cesium, whereas cabbage absorbed only 5% to 6% of either of these elements.

The effectiveness of weathering as a mechanism for removing deposited materials has been estimated by the Federal Radiation Council (1965): "There is a wide range of measured values for the effective half-time of radionuclides on grass, most of which are close to 14 days." Unfortunately, since no references are given to the sources of these measured values and the types of contaminant are not indicated, it is difficult to assess the overall applicability of this half-time. It is noted, in the NAS-NRC report (National Academy of Sciences-National Research Council, 1963) that the washing effect of rain can reduce the foliar intake of strontium from material sprayed on the leaves by a factor as large as 6 and can reduce foliar intake of cesium to a lesser extent. It is not indicated whether a similar reduction occurs in the total contamination including adsorbed isotopes. Over 50% of the iodine in foliar contamination is removed by washing. Washing can also remove ions from within the plant. Sodium, potassium, and manganese are readily leached; calcium, magnesium, sulfur, potassium, and strontium are moderately leached; and iron, zinc, phosphorus, and chlorine are leached with difficulty. Thus some reduction in the amount of radioactive materials on the vegetation may occur during food preparation from such processes as washing, peeling, husking, and cooking.

A possible mechanism for uptake of materials not deposited directly on the foliage exists where there are root mats as in long-established pastures. Here the ^{90}Sr may be deposited directly on the roots and held long enough for some absorption to occur.

If the deposition occurs as a result of a single release, as in an accident, the vegetation contamination occurs in a relatively short time. Under conditions in which the vegetation serving directly as a source of food for man is harvested continually (as in a truck farm or a home garden), the uptake in the body can be estimated by using the uptake factor given by the ICRP (International Commission on Radiological Protection, 1959a) for water. (Note that this uptake factor may be different from that of water owing to the possible difference in chemical species of the radioisotope and the differ-

ence in chemical compounds associated with the radioisotope.) In this case the change in organ content can be written as

$$d(qf_2) = M_e f_w c_g \exp(-\lambda_v t) dt - (qf_2)(\lambda_R + \lambda_B) dt \quad (7.116)$$

where qf_2 = organ burden (μC)

M_e = quantity eaten (g/day)

f_w = fraction of ingested isotope that is deposited in the critical organ

c_g = concentration of isotopes in foodstuff ($\mu\text{C/g}$)

λ_v = fraction of isotope removed from the vegetation by decay, weathering growth, etc., (1/day)

λ_R = radioactive disintegration constant (1/day)

λ_B = fraction of isotope removed from the critical organ per day by biological processes (1/day)

Solving Eq. 7.116 with the condition that qf_2 equals zero at time zero (time is measured after ingestion starts) gives

$$qf_2 = \frac{M_e f_w c_g}{\lambda_R + \lambda_B - \lambda_v} \{ \exp(-\lambda_v \tau) - \exp[-(\lambda_R + \lambda_B)\tau] \} \quad (7.117)$$

where τ is the time interval after ingestion begins (days).

Equation 7.117 describes the quantity of radionuclide in the organ at any time after the start of ingestion. Substituting Eq. 7.117 for (qf_2) in Eq. 7.88 and integrating yields the equation for the total dose received over any time interval τ :

$$D = \frac{51 \bar{E} M_e f_w c_g}{M (\lambda_R + \lambda_B - \lambda_v)} \left\{ \frac{1 - \exp(-\lambda_v \tau)}{\lambda_v} - \frac{1 - \exp[-(\lambda_R + \lambda_B)\tau]}{\lambda_R + \lambda_B} \right\} \quad (7.118)$$

where the values of λ_v , λ_R , and λ_B are in units of 1/day.

For the special case in which removal from both the organ and the vegetation is by radioactive decay only, i.e., $\lambda_v = \lambda_R$ and $\lambda_B = 0$,

$$qf_2 = M_e f_w c_g \tau \exp(-\lambda_R \tau) \quad (7.119)$$

Substituting Eq. 7.119 for (qf_2) in Eq. 7.88 yields the dose received over the time period τ when removal is by radioactive decay only:

$$D = \frac{51 \bar{E} M_e f_w c_g}{M \lambda_R^2} [1 - \exp(-\lambda_R \tau)] \times (\lambda_R \tau - 1) \quad (7.120)$$

where λ_R is in units of 1/day.

The dose rate to the GI tract can be estimated by comparing the total intake of the radioactive material ($M_e c_g$) with the total intake of water at the MPC calculated by the NCRP to give 300 mrem/week to the GI tract (National Committee on Radiation Protection and Measurement, 1959). The total dose over a period of time can be estimated by integrating the resulting dose rate over the time of interest.

If the crop is harvested as a whole, the quantity remaining on the edible portions at the time of harvesting will depend on the rate of removal by weathering, growth, etc. and the time since the contamination occurred. This quantity will then remain constant except for radioactive decay. The quantity of isotope accumulated in the critical organ of an individual eating M_e grams of this crop per day will then be

$$(qf_2) = \frac{M_e f_w c_g \exp(-\lambda_v \tau_1)}{\lambda_R + \lambda_B} \times \{1 - \exp[-(\lambda_R + \lambda_B) \tau_2]\} \quad (7.121)$$

where c_g is the initial deposition ($\mu\text{C/g}$), τ_1 is the time interval between deposition and harvest, and τ_2 is the time over which the crop is eaten.

With a continuous release, as in waste disposal, the rate of deposition when averaged over a long time period can be taken as essentially constant. Under these conditions the contamination levels in the foodstuffs will be given by

$$c_g = \frac{f v_d \bar{X}_{(x,y,0)}}{M_g (\lambda_R + \lambda_v)} \{1 - \exp[-(\lambda_R + \lambda_v)t]\} \quad (7.122)$$

where f = fraction of the deposited material which is on the edible portion

v_d = deposition velocity (m/sec)

M_g = mass of edible portion per unit area (g/m^2)

t = time over which the deposited material is accumulated (days), i.e., effective time of formation of the edible portions

The quantity in the critical organ of an individual eating the food can then be calculated for continuous intake over the contamination period or after harvesting as in the short-period release.

7-7.1.2 Uptake from the Soil. Soils can become contaminated from airborne materials by several processes:

1. Direct deposition on the soil.
2. Removal of material initially deposited on vegetation by weathering.
3. Seasonal cycling of vegetation cover.
4. Irrigation by water contaminated by direct fallout or by runoff from a contaminated area.

Once on the soils, the fate of the radioactive material depends upon the physical nature of the material and the chemical behavior of the radioisotopes. In general, soluble materials will be adsorbed to a variable degree by soil particles. This results in the gradual leaching of such materials as strontium or cesium through the soil. For example, ^{90}Sr from worldwide fallout was located primarily in the upper 2 in. of uncultivated soil during 1954 and 1955 (National Academy of Sciences-National Research Council, 1963).

The plant uptake will depend on the concentration of the radioisotope near the roots and on the availability of the radioisotope in ionic form. Of the long-lived fission products, strontium and cesium are most strongly absorbed. Since cesium is more tightly bound to the soils than strontium, it is usually regarded as less likely to be absorbed by plants. The uptake in a particular situation, however, can vary, depending upon the local conditions. Cesium uptake, for instance, is increased by the presence of ammonia or potassium because these ions free the adsorbed cesium from the soil (U. S. Atomic Energy Commission, 1961). In addition, bean plants that were grown under conditions dry to the wilting point contained five times the concentration of ^{137}Cs in stems when compared with those grown with the soil moisture kept at near field capacity.

Measurements of the uptake indicate that about 1% of the applied strontium and less than 0.1% of the other elements are taken up by single crops of plants; however, uptake of up to 4% to 8% has been noted in controlled pot ex-

periments. When expressed as the ratio of plant-tissue concentration to soil concentration, factors for fallout constituents have been reported as 0.05 for the alkaline earth group and 0.009 for the rare earths in barley. The average maximum uptake of cesium in experiments appears to be about 0.1% of the applied dose. However, in a complex situation represented by long-continued worldwide fallout from nuclear debris, the ratio of ^{137}Cs to ^{90}Sr in Chicago foodstuffs was higher than in the contaminating mixture even for root vegetables and during periods when the rate of fallout was comparatively low (Hardy and Rivera, 1964).

Strontium tends to accumulate in the above-ground portions of plants with the greatest concentration usually found in the older leaves. Cesium is distributed more uniformly through the plants. About 10% of all cesium found in plants is in the grain of wheat and oats.

The root system of the plant plays an important role in uptake. As a result, the depth of the contamination in the soil in relation to the root will also affect the uptake. Since grasses have a shallow root system, they will tend to absorb radioisotopes occurring near the surface. In a grass-clover mixture the strontium content was reduced 70% by plowing under the surface contamination. In other crops, cultivation tends to increase strontium uptake. Little variation in the availability of strontium has been observed over a period of years, and the uptake of strontium per unit of plant weight is fairly constant. The uptake of cesium increased under prolonged cropping, possibly because of some potassium depletion in the soil.

The organ burden of an individual eating such crops can be estimated from Eq. 7.121 and the estimated contamination level of the crop. The dose to the organ can then be obtained from Eq. 7.88.

Special dose considerations are frequently given to ^{90}Sr because of its metabolic similarity to calcium. Both strontium and calcium are incorporated into new bone in the same ratio as they exist in blood. The dose to the mineral portion of the bone and that to the marrow surrounded by the mineralized bone then depend upon the ^{90}Sr concentration in the calcified bone. The ratio of strontium to calcium in the blood, however, is not the same as the ratio of strontium to calcium in nature or even in the diet

since there is some discrimination against the strontium. This discrimination is expressed as the Observed Ratio (OR) and is the ratio of the Sr/Ca in bone to the Sr/Ca in diet. Values of the OR are quoted as about 0.1 for the fetus, 1 at birth, 0.5 at the age of one year, and 0.25 for the adult (National Academy of Sciences—National Research Council, 1964). It is estimated that $10^{-3} \mu\text{c}$ of ^{90}Sr per gram of calcium uniformly distributed in adult bone would result in a dose of about 2.7 rads/year to the mineral bone and about 0.7 rads/year to the bone marrow (Federal Radiation Council, 1965). The dose to mineral bone of the fetus and infant may be about one-half of the value for adults because of the lower calcification although the dose to the bone marrow will be about the same as for the adult.

These concepts are applicable primarily to the ^{90}Sr intake over a long period of time. For short-term exposures, about 50% of the deposited radiostrontium is localized in areas of active bone formation, and the remainder is diffusely distributed in the bone in lower concentrations.

7-7.2 Contamination of Animal Products

Animals living in a contaminated region frequently will ingest radioactive materials to a much greater extent than humans. A portion of some of these radioisotopes will appear in the products used by man, such as meat, milk, eggs, etc. Chains of this nature that appear to be of the most importance are

There is some evidence that eggs may be of importance in such chains, particularly for ^{131}I .

Information on these chains and their importance has been developed both from experience and by experiment (Dunster, Howells, and Templeton, 1958); United Nations, 1962). Milk containing ^{131}I was the controlling factor in the environment following the Windscale release.

Here a deposition of about $1 \mu\text{C}/\text{m}^2$ resulted in about $0.09 \mu\text{C}/\text{liter}$ of ^{131}I milk.

Knapp (1963) from a study of the ^{131}I milk concentrations resulting from nuclear fallout during periods of low and high deposition rates indicated that the concentration in milk is related to the quantity in the soil and the current rate of deposition by

$$\begin{aligned} \text{Milk concentration } (\mu\text{C}/\text{liter}) &= \frac{1000}{9} \\ &\times (\text{total fallout on ground}) \\ &+ 2600 \times (\text{deposition in the last month}) \end{aligned}$$

where the fallout is expressed in curies per square mile. The first term expresses the uptake in vegetation from material previously deposited and the second term describes the strontium present from current direct deposition.

If data are available on the uptake and retention (or excretion in milk) from a single administration of a radioisotope, the quantities in meat or milk can be estimated. For example, for a single contaminating event with ^{137}Cs followed by continued ingestion of the vegetation by cows, the average concentration in meat will be given by

$$\frac{dc_m}{dt} = \frac{M_e c_v f_w}{M_m} \exp(-\lambda_w t) - c_m \lambda_B \quad (7.123)$$

where c_m = concentration of radioactive material in the animal's muscle ($\mu\text{C}/\text{g}$)

M_e = mass of food ingested by the animal per day (g/day)

c_v = concentration of cesium in the animal's food ($\mu\text{C}/\text{g}$)

f_w = fraction of the radioactive material eaten that goes to the muscle

M_m = total mass of the animal muscle (g)

λ_w = weathering constant for vegetation (1/day)

λ_B = elimination constant for the muscle (1/day)

Solving Eq. 7.123 gives

$$\begin{aligned} c_m &= \frac{M_e c_v f_w}{M_m (\lambda_B - \lambda_w)} [\exp(-\lambda_w t) \\ &\quad - \exp(-\lambda_B t)] \quad (7.124) \end{aligned}$$

7-7.3 Contamination of Water

Direct deposition into a stream or body of water is usually of minor importance compared with deposition on land because of the dilution and the relatively rapid removal of the contaminated portion by flushing. In special cases significant quantities of radioisotopes may deposit in sources of domestic water.

A special problem is the contamination of water supplies by runoff from land contaminated with reactor fission products by direct deposition or by decontamination efforts following an accident. For example, if 1 sq mile of land was contaminated so that the gamma reading at 1 m above the ground was about 5 r/hr, decontamination disposal to a stream with a flow of 5000 cu ft/sec would result in an average concentration of 10^{-6} to $10^{-5} \mu\text{C}/\text{cm}^3$ over a 1-month period.

Potential doses from the direct ingestion of water can be estimated by comparison to the maximum permissible concentration in water given by the NCRP (National Committee on Radiation Protection and Measurement, 1959) with appropriate allowances made for the exposure time compared to the half-life of the contaminating radioisotope.

Food chains are again of importance in many bodies of water because some elements are concentrated strongly in the lower organisms and in fish or wild fowl that feed on these organisms (Davis, Perkins, Palmer, Hanson, and Cline, 1958). This may make fish or wild fowl a source of significant radiation dose if they are eaten by people, even though the water may be within maximum permissible limits for drinking purposes. In addition, this provides a mechanism for retaining some contamination in a body of water even after the major contaminated portion of the stream has passed.

Detailed studies have been made of several radioisotopes in particular environments (Dunster, 1958, Couchman, 1960, and Dunster, Howells, and Templeton, 1958), but these data have not been generalized to permit overall estimation of the effects in any environment. Estimates of the potential effects can be made only after detailed study of the data available and their application to the environment in question.